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Nuclear Fuel Cycle Risk Assessment: Descriptions of Representative Non-Reactor Facilities

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Sections 15-19

Prepared by K. J. Schneider, Coordinator

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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15.0 FUEL REPROCESSING

The purpose of a fuel reprocessing plant is to separate, purify, and prepare for recycle the uranium and plutonium present in the spent fuel discharged from nuclear power reactors.

15.1 SUMMARY

The LWR fuel reprocessing facility described in this section is representative of current solvent extraction technology with the addition of waste processing and environmental control systems whose designated descriptions are based on conceptual and experimental studies. The other two facilities are a UF₆ conversion facility and a PuO₂ conversion facility. The overall characteristics and bases for the LWR fuel reprocessing plant are specified as follows:

- a single plant capable of reprocessing spent LWR fuel at 1500 MTHM per year charged to the reactor. Plutonium and uranium are separated from the spent fuel, purified, and converted to PuO₂ and UF₆, which are stored in product shipping containers.
- one plant capable of receiving spent fuel (enriched uranium dioxide and mixed oxide, both from LWRs) with two basic products (PuO₂ and UF₆). Only enriched UO₂ fuel is used in the fuel cycle currently and it is emphasized in this description.
- the reprocessing facility receives spent LWR fuel from commercial BWRs and PWRs.
- the burnup of the LWR fuel, initially enriched to 3.2 percent, is 33,000 MWd/MgHM which is achieved in 880 full power days.
- the facility operates at full capacity, and reprocesses 5 MgHM per day for 300 days per year and 24 hours per day for 40 years.
- the facility processes fuel at an annual rate that represents 45 GWy of electrical production.
- the facility accommodates spent LWR fuel with an initial enrichment of up to 5 percent ²³⁵U or Pu equivalent.
- the facility produces 810 canisters of solidified high-level and intermediate level waste, 180 canisters of processed cladding hulls, and about 10,000 drums of low-level waste per year.
- The plant requires an operational staff of 500.

The fuel reprocessing facility receives and stores spent LWR uranium fuels in a water-filled pool. The spent fuel is transferred from the storage pool, and the fuel rods are separated from the end fittings and are chopped into

short lengths. The fuel is dissolved in nitric acid, and the fission product and transplutonic element contaminants are separated from the fuel materials by a liquid-liquid solvent extraction (Purex) process. Plutonium and uranium are separated from each other and purified as nitrate solutions. The plutonium is converted to the solid oxide form utilizing an oxalate precipitation process followed by calcination. The containerized plutonium is stored onsite prior to shipping offsite as needed for fabrication into mixed oxide fuel elements. The uranium is converted by calcination, hydrogen reduction, hydrofluorination, and fluorination to UF_6 . The UF_6 is recycled to an enrichment facility (see Section 12) for reintroduction into the fuel cycle.

The high-level liquid wastes, which contain essentially all of the fission products and the transplutonium elements as well as about 0.5 percent of the plutonium and uranium from the spent fuel, are stored in water-cooled tanks for up to 5 years before they are solidified for transfer to a federally designated repository. This waste solidification facility features a continuous liquid-fed joule-heated melter which produces a glass waste form subsequently incorporated in steel canisters which are stored onsite prior to transfer. Other solid wastes (containing about 0.2 percent of the uranium), which include the fuel cladding materials, general process trash, failed equipment, and radioiodine removal waste, are packaged and stored onsite prior to shipment for ultimate disposition.

The process off-gases receive extensive treatment to minimize the release of radioactive materials. The noble gases and the water vapor containing most of the tritium in the spent fuel are released directly to the atmosphere with no treatment. Airborne effluents contain all of the krypton, xenon, and ^{14}C , about 70 percent of the tritium as the oxide, and less than 0.1 percent of the radioiodine from the original fuel. Less than $2E-8$ percent^(a) of the other fission products, of the uranium, and of the plutonium from the spent fuel are released in the airborne effluents. Contaminated liquids containing small amounts of radioactivity are evaporated for release as airborne effluents.

The predominant radioactive airborne effluents emanating from the processing are water vapor and mixed oxides of nitrogen. Carbon dioxide and sulfur dioxide are among the contaminants released from the combustion of fossil fuels for onsite steam generation. Also, carbon monoxide, hydrogen sulfide, hydrogen fluoride, ammonia, hydrogen, and fluorine are released from facility stacks daily. Radioactive materials of process origin are not released in any facility liquid effluent. Extensive recycle of process liquids is employed to reduce the amount of liquid effluents. Nonradioactive contaminants discharged in the liquid effluent amount to several thousand kg per day.

A summary of safety-related information is presented below; information for UF_6 conversion was presented in Section 11.

(a) $2E-8 = 2 \times 10^{-8}$; form used throughout this report.

Confinement Levels

The number of confinement levels varies with the amount and type of radioactivity present in a particular area or equipment. For high activity areas such as first-cycle solvent extraction or plutonium oxide precipitation, three levels of confinement are provided: the equipment, the cell or canyon, and the building shell. Pressure differentials are maintained by the ventilation system such that air flow is from outdoors into the building, into the operating area, into the canyon or cell, and into process equipment. Adequate confinement of cell and area air volumes is provided by filtration and treatment in the ventilation systems.

Remote Operations

All operation in spent fuel handling, dissolving, first-cycle solvent extractions and high-level waste operations require remote operations and maintenance. Second- and third-stage extraction, most of the waste handling operations, plutonium storage, and plutonium conversion and packaging require remote operations, but allow contact maintenance. Some of these operations are conducted in cells with glove box walls; gloves are used only when the equipment is shut down and most radioactive material has been removed. Contact operations are possible in $\text{UO}_2(\text{NO}_3)_2$ to UF_6 conversion and low-level waste handling. Glove boxes are used for some of these operations to provide needed confinement of particulate matter.

Waste and Effluent Streams

The values given below for waste and effluent streams are based on the radioactivity in spent fuel 5 years after the end of irradiation. It is expected that spent fuel may not be reprocessed until aged for 5 years or more in the foreseeable future and the high-level waste may not be solidified until 10 years or more after reactor discharge.

The radioactive waste streams consist of canisters of HLW and ILW incorporated in glass, containers of cladding hulls packed in sand, and drums of LLW immobilized in concrete. The annual production of HLW/ILW in glass is 4.5 E5 kg (145 m^3 , 6.5 E8 Ci). These canisters are eventually transported to a geologic repository after a period of onsite storage. The cladding waste in sand is produced at an annual rate of 5.7 E5 kg (125 m^3 , 1.0 E7 Ci); this waste is also stored onsite prior to shipment to a geologic repository. About 10,000 drums (4.2 E7 kg , 3000 m^3 , 3000 Ci) of LLW incorporated in concrete are produced annually; LLW TRU wastes are eventually sent to a geologic repository and non-TRU wastes are placed in a shallow-land burial ground.

The effluent streams from the facilities consist of gaseous effluents released from stacks and cooling towers, and non-radioactive liquids released to a holding pond. Radioiodine is the only gaseous element intentionally removed from the reprocessing off-gas stream. Liquid streams that contain radioactivity are recycled within the plant. Solids in the liquid discharge are nonradioactive and arise from recirculating cooling water treatment, etc. The gaseous discharges contain radioactive gases and particulates (fission products and actinides). About 42 kg of radioactive gases (1.8 E7 Ci) are

released per year; the principle component by weight and radioactivity is ^{85}Kr . In total, about 8700 kg of fission product gases are released per year. About $5.1 \text{ E}8 \text{ kg}$ ($6.0 \text{ E}3 \text{ Ci}$) of particulate fission products, and about $1.7 \text{ E}6 \text{ kg}$ ($1.7 \text{ E}4 \text{ Ci}$) of actinides are also released yearly with the gaseous effluents. About $6.0 \text{ E}7 \text{ kg}$ of nonradioactive gases (e.g., NO_x , CO_2 , SO_2 exclusive of air) and about $7.2 \text{ E}8 \text{ kg}$ of water vapor are released annually. The annual discharge of water is $3.8 \text{ E}9 \text{ kg}$ containing $4.8 \text{ E}5 \text{ kg}$ of nonradioactive solids.

Inventories

Very rough estimates of the inventories of significant amounts of radioactive and/or hazardous material at various locations in the plant are listed below.

- spent fuel pool--2700 LWR fuel assemblies, 1350 MTHM
- dissolver--6800 ℓ , 10.5 kg fissile
- feed adjustment tanks--20,500 ℓ , $\sim 114 \text{ kg}$ fissile
- first cycle extraction column--630 ℓ , $\sim 0.8 \text{ kg}$ fissile
- first cycle plutonium stripper--3500 ℓ , 2.4 kg fissile
- second and third cycle plutonium extraction columns--450 ℓ ,
 $\sim 3.1 \text{ kg}$ plutonium
- second and third cycle plutonium stripping columns--4200 ℓ ,
 $\sim 2.9 \text{ kg}$ plutonium
- plutonium nitrate storage--NA ℓ , NA kg plutonium
- plutonium nitrate concentrator--NA ℓ , $\sim 24 \text{ kg}$ plutonium
- plutonium oxalate precipitation--75 ℓ , $\sim 4.5 \text{ kg}$ plutonium
- plutonium oxide calciner-- $\sim 8 \text{ kg}$ plutonium
- plutonium oxide blender--40 kg plutonium oxide
- plutonium canister--10 kg plutonium oxide
- HLW canister storage--4000 canisters
- Cladding hulls storage--800 containers
- LLW storage--20,000 drums

Hazardous Materials

The following hazardous materials are stored in tanks and are used onsite; their use is indicated:

- anhydrous ammonia as a source of hydrogen for reduction of UO_3 to UO_2
- anhydrous HF as a source of fluorine and for conversion of UO_2 to UF_4
- fluorine for fluorination of UF_4 to UF_6
- kerosene as make-up diluent for PUREX extractant and incineration
- nitric acid (68 weight percent) for dissolving spent fuel and for various scrub and rinse operations
- sulfuric acid (93 weight percent) for pH control of recirculating cooling water, and dichromate and pH control of offsite water discharge
- chlorine for sanitary water treatment
- TBP for solvent extraction makeup
- KOH solution for various scrubbing processes
- propane as fuel for gas-fired processes.

The majority of these materials is in storage outside of process buildings; only for kerosene is a large source stored inside the reprocessing facility but its hazard is reduced by being mixed with TBP.

Unit Operations

A spectrum of unit operations are employed throughout the three major facilities on the reprocessing plant site. Those associated with major inventories of radioactive and/or hazardous material are listed by location below:

- fuel storage - solids handling, liquid pumping, ion exchange
- dissolver - solids dissolution, solids filtration
- first cycle extraction - organic/aqueous solvent extraction, aqueous stripping, liquid pumping
- second and third cycle plutonium solvent extraction - organic/aqueous solvent extraction, aqueous stripping, liquid pumping
- HLLW concentration - evaporation

- HLW solidification - solids handling, solids melting
- plutonium nitrate storage - liquid pumping, evaporation
- plutonium oxalate precipitation - concentration, precipitation, filtration, liquid pumping
- plutonium oxide production - drying, calcining, screening, grinding, mixing, solids handling, packaging
- hydrofluorination of UO_2 - solids handling, high temperature chemical reaction, liquid vaporization
- fluorination of UF_4 - solids handling, high temperature chemical reactions
- fluorine production - liquid vaporization, electrolysis.

Accident Driving Forces

Accident driving forces present at locations containing significant amounts of radioactive or hazardous materials are listed below.

- plutonium product storage and concentrator - 50 to 150 C, >500 g fissile
- plutonium calciner - >150 C, >500 g fissile
- HLW melter - >150 C, >500 g fissile
- HF storage - high chemical reactivity
- HNO_3 and H_2SO_4 storage - high chemical reactivity
- fuel storage pool - cask or fuel element drop, 25,000 kW
- plutonium cells - blender or canister drop, >500 g fissile
- waste storage - canister or drum drop
- plutonium facility - container drop, >500 g fissile
- HLLW storage - 19,500 kW, >500 g fissile
- dissolver - >500 g fissile
- PUREX process - >500 g fissile.

Reprocessing technology is almost 40 years old and is continually evolving in government-owned facilities in the United States and in other countries. Alternative processes that have been studied (see Section 15.6.1) need the

confidence obtained from demonstrations to provide a solid basis for extrapolation. It will be necessary to consider additional environmental control systems (as described in Section 15.6.2) to meet air quality standards for radioactive krypton and carbon in the 1990s. The process descriptions are based primarily on material in PNL-2286 (Schneider and Kabele, 1979) and in the Hanford PUREX Draft Environmental Impact Statement (U.S. DOE, 1981).

15.2 MAINLINE PROCESS DESCRIPTION

The mainline process is divided into three categories for convenience: 1) the process by which the uranium and plutonium are recovered in highly purified nitrate solutions, 2) the process by which the purified uranium is converted from nitrate solution to uranium hexafluoride, and 3) the process by which the purified plutonium is converted from nitrate solution to plutonium dioxide. The first process is performed in the main separations facility, the second process in a separate UF_6 facility, and the third in a separate plutonium product facility. A simplified flow diagram of these is shown in Figure 15.1.

15.2.1 The Uranium/Plutonium Recovery Process

The reference reprocessing facility uses the Purex process, which has been in large-scale use for over 20 years, and is currently used, with minor variations, by most reprocessing plants now operating throughout the world. A detailed overall process flow sheet is presented in Figure 15.2, which can be referred to for the remaining descriptions of this subsection. A representative process is described. The same basic operations could be conducted with different equipment and with different concentration of acid than those stated.

Fuel Assembly Receiving

The irradiated fuel assemblies arrive at the facility in shielded casks. The cask and carrier are monitored for exterior radioactive contamination to determine if any leakage has occurred, and are washed to remove outside road dirt. The cask environment is sampled and analyzed before the cask is removed from the carrier and vented. The fuel-containing cavity is flushed to remove radioactive contaminants that would unnecessarily contaminate the fuel storage pool.

Fuel Assembly Storage

The vented cask is moved by the cask handling crane to the cask unloading pool. The head bolts are loosened before lowering it to the bottom of the pool. Under water the cask lid is removed and the contained fuel is unloaded and placed in storage sleeves. These sleeves are then moved to the fuel storage pool for retention until the fuel is scheduled for reprocessing.

Fuel Assembly Shearing and Dissolution

The fuel assemblies are remotely transferred from the storage pool to the feed mechanism of the mechanical shear. After the end fittings have been

FIGURE 15.1. Overall Mainline Processes for the Fuel Reprocessing Plant

FIGURE 15.2. Mainline Process Flow Diagram for the Main Separations Facility

removed from the fuel assembly, the fuel rods are chopped into segments 5 to 12 cm long to expose the fuel to the dissolvent. The fuel segments fall into the dissolver containing 3 to 8M nitric acid and a small concentration of gadolinium nitrate, which serves as a poison to prevent an unwanted nuclear criticality event. The uranium, plutonium, and fission products are almost completely dissolved in the acid.

The undissolved cladding materials and accompanying hardware of stainless steel and Inconel remain in the dissolver basket. The dissolver solution is centrifuged to remove fine solids, which are sent to the high-level waste treatment system after washing and slurring from the centrifuge. The clarified dissolver solution is transferred to tanks to be sampled for U and Pu, and the acid concentration is adjusted to 2 to 3M nitric acid before the solution is fed to the solvent extraction process. The cladding hulls are rinsed with dilute nitric acid, monitored for residual fissile material, packaged in the remote maintenance and scrap cell, and transferred to the interim waste storage area prior to shipment to the repository.

Solvent Extraction, Partitioning, and Stripping of Plutonium and Uranium

The adjusted solution is fed to the first solvent extraction column where it is contacted countercurrently with a solution of tributyl phosphate (30 percent TBP) in normal paraffin hydrocarbon diluent (dodecane). The lighter organic solution preferentially extracts the tetravalent plutonium and hexavalent uranium, leaving most of the fission products and the other actinides in the aqueous acid solution. The organic phase is scrubbed with 2M nitric acid in the top of the column for additional fission product decontamination.

The aqueous acid solution leaving the bottom of the column contains about 99.9 percent or more of the nongaseous fission products, essentially all of the transplutonium elements and about 0.5 percent of the uranium and plutonium; it is sent to a high-level waste concentrator.

The organic solution from the pulsed column is then joined by organic raffinates (used organic streams) from the plutonium purification sections and passes through a U/Pu partitioning column where tetravalent plutonium is reduced to the less extractable trivalent state. This valence change enables the plutonium to be stripped into the aqueous 0.5M nitric acid solution containing the reductant. The organic uranium-containing solution passes through the final first cycle pulsed column where the uranium is stripped into acidified water (about 0.01M HNO_3).

Second Uranium Solvent Extraction and Concentration

The aqueous strip solution containing the uranium is concentrated from 0.3M to 1.5M uranium and adjusted to about 0.3M nitric acid. It is then sent to the second uranium solvent extraction cycle where it is preferentially extracted by another TBP/organic solution in a pulsed column. Before leaving the column, the organic solution is scrubbed with 1.3M nitric acid solution, which removes additional fission products. Hydroxylamine nitrate and hydrazine are also added to the scrub solution to remove residual plutonium by chemical reduction to the less extractable trivalent state.

Uranium is stripped from the organic solution in the subsequent pulsed column, using acidified water (0.01M nitric acid). This solution is concentrated in an evaporator from 0.4M uranium to 1.5M uranium. Finally, the concentrated uranium solution from the second cycle is passed through silica gel beds to remove residual traces of the fission products zirconium and niobium. Uranyl nitrate product solution is analyzed on a batch basis and piped to storage tanks in the separate onsite facility for conversion to UF₆.

Second Plutonium Solvent Extraction

Plutonium in the aqueous stream leaving the partitioning column in the first cycle is reoxidized to the tetravalent state with nitrogen dioxide or sodium nitrite and sent to the second plutonium solvent extraction cycle. Here it is preferentially extracted into TBP/organic solution in another pulsed extraction column. In the top portion of the same column, the organic stream is scrubbed with 1.0M nitric solution to remove most of the residual fission products. The organic product stream then passes through a strip column where tetravalent plutonium is transferred to an aqueous stream of dilute (0.01M) nitric acid.

Third Plutonium Solvent Extraction and Concentration

The extraction-scrubbing sequence is repeated in a third plutonium cycle for further removal of fission products. The plutonium feed is again oxidized by nitric acid to the tetravalent state. The plutonium is subjected to another solvent extraction, scrub and strip operation in two columns as in the previous purification operation. The plutonium nitrate solution is finally washed with organic diluent to remove traces of TBP.

Final plutonium nitrate solution concentration from 60 g/l to 200 g/l is accomplished in a concentrator made of titanium. The plutonium product solution is analyzed and stored in critically safe, slab-shaped tanks until it is transferred to the plutonium product facility for conversion to PuO₂.

Solvent Cleanup and Recovery

The organic solvent stream from the first (partition) cycle is washed successively with dilute aqueous solutions of sodium carbonate, nitric acid, and sodium carbonate to remove organic degradation products by extraction or precipitation. Precipitated solids are removed by filtration. Fresh TBP or diluent is added as required to maintain the 30 percent TBP concentration and the total solvent inventory. The solvent is then reused in the first cycle and in the plutonium cycles. The organic solvent stream from the second uranium cycle is treated similarly, except the second alkaline wash is omitted. This solvent is kept in a separate system.

Secondary Product Recovery by Solvent Extraction

The aqueous waste (raffinate) streams from the plutonium and uranium cycles are treated with nitrogen dioxide or sodium nitrate for valence adjustment to Pu (IV) and U (VI) and passed through a secondary recovery solvent

extraction cycle. Here they are processed in a pulse column where most of the residual uranium and plutonium are recovered by extraction into a TBP organic solution. The uranium and plutonium are recycled back to the first (codecontamination) cycle. The aqueous raffinate stream is concentrated in a low-level process waste concentrator, the bottoms from which are transferred to the high-level waste concentrator.

The overall recovery process produces uranium and plutonium product streams containing about one part in ten million of the fission products originally present in the spent fuel. This purity translates to a radioactivity level in recovered uranium of about twice that of natural uranium. Other separation factors are given in Table 15.1. The radioactivity levels in the various processing areas range from very high levels that require artificial cooling to remove the heat from the radioactive decay to levels low enough to permit direct personal contact.

15.2.2 Uranium Nitrate/Uranium Hexafluoride Conversion

Uranium nitrate solution is converted to uranium hexafluoride in the representative fuel reprocessing UF_6 facility. A detailed UF_6 facility process flowchart is shown in Figure 15.3 where the sources of effluents from the various process steps are also presented. This information is presented in greater detail in Section 11 of this report.

Uranyl Nitrate Receiving and Storage

The UF_6 facility receives nitrate solution recovered from spent fuel in the adjacent main separations facility. The solution is received in an accountability tank where it is measured, sampled, and then transferred to the storage tanks.

Uranyl Nitrate Calcination

The 40 weight percent (350 g/l) uranyl nitrate solution is pumped from storage tanks to a steam-heated thermosyphon reboiler where water is removed to form uranyl nitrate hexahydrate (UNH) containing 78.5 percent uranyl nitrate. The evaporated water is condensed and returned to the main separations facility for recycle. The UNH is then continuously calcined to uranium trioxide (UO_3) in a bed of UO_3 fluidized by superheated steam at $315^\circ C$. By denitrating in steam, the nitrate values in the calciner vapors are condensed as 45 weight percent nitric acid which is returned to the main separations facility for reuse.

Uranium Trioxide Reduction

A controlled discharge of UO_3 is withdrawn from the bed and is prepared as feed material by milling and screening to a uniform particle size followed by chemical activation with the addition of small amounts of H_2SO_4 . The UO_3 is continuously converted to uranium dioxide (UO_2) by reduction with hydrogen in a fluidized bed. The hydrogen is obtained by dissociation of ammonia.

TABLE 15.1. Typical Decontamination Factors for Fission Products (FP) and the Transuranics (TRU) from U and Pu in the Representative Fuel Reprocessing Plant

Separation Step	Approximate DF ^(a)
<u>Fuel Dissolution</u>	
Noble gases from Pu and U	4
Radioactive carbon from Pu and U	2
Iodine from Pu and U	1.5
All other FPs, TRUs from Pu and U	0
<u>1st Cycle Solvent Extraction</u>	
FPs, TRUs from U and Pu	4
U from Pu and Pu from U	4
<u>2nd Cycle Solvent Extraction</u>	
FPs, TRUs from U	2
FPs, TRUs from Pu	2
U from Pu and Pu from U	2
<u>3rd Cycle Solvent Extraction</u>	
FPs, TRUs from U	1
FPs, TRUs from Pu	1
<u>Cumulative</u>	
FPs, TRUs from Pu and U	7
<u>Waste</u>	
Pu from waste	2.1
U from waste	2

(a) Decontamination factor here is the quantity of FPs and/or TRUs associated with the U and Pu from the original irradiated fuel after the separation step compared to that before the separation step expressed as a log factor (exponent of 10).

FIGURE 15.3. UF₆ Facility Flow Chart

Uranium Dioxide Fluorination

The UO_2 is reacted with hydrogen fluoride (HF) continuously in a bed fluidized by HF to form uranium tetrafluoride (UF_4). After milling and blending, the UF_4 solids from hydrofluorination are fed to the fluorinator, which contains a bed of inert granular material (alumina) fluidized with gaseous fluorine. Fluorine is produced onsite by electrolysis of anhydrous HF in an electrolyte of molten potassium bifluoride. Solid UF_4 is converted to gaseous UF_6 which, along with excess F_2 and by-product HF, is filtered and passed to cold traps for UF_6 collection.

The bulk of the UF_6 is condensed to a solid in a primary cold trap. Secondary and tertiary traps operate at lower temperatures and remove additional UF_6 . Product UF_6 is removed from the cold traps intermittently as a liquid by heating the traps. The UF_6 is transferred through surge tanks into approved 14-ton UF_6 storage and shipping cylinders where it is sampled, cooled, and solidified before being moved to the storage pad.

15.2.3 Plutonium Nitrate/Plutonium Oxide Conversion

Plutonium nitrate solution is converted to plutonium oxide powder in the plutonium product facility. The conversion process consists of continuous precipitation of plutonium oxalate followed by filtration and calcination to plutonium dioxide (PuO_2). This process has been used for over 20 years in various nuclear installations. Two parallel conversion lines (i.e., precipitation through product packaging) are provided, each capable of furnishing the total production capacity (100 kg per day). A simplified block flow diagram of the process is shown in Figure 15.4.

Plutonium Precipitation

Plutonium nitrate solution is transferred in batches from plutonium nitrate storage to feed preparation tanks. The nitric acid concentration is adjusted to 3M by adding concentrated nitric acid, the plutonium valence is controlled to +4 by adding hydroxylamine nitrate. The plutonium concentration is approximately 150 g/l. The adjusted product feed and oxalic acid (1M) streams flow continuously to a precipitator vessel where they are mixed and precipitation commences. The slurry overflows the precipitator vessel to successive digestion vessels to allow crystal growth.

Plutonium Oxalate Filtration and Calcination

The slurry is filtered continuously on a rotary vacuum drum filter. The precipitate is then dried and calcined in a rotary screw calciner at temperatures up to 750°C. Countercurrent airflow at reduced pressure removes the gaseous products from the calciner.

Plutonium Dioxide Screening, Blending, and Packaging

The plutonium oxide powder is screened; oversize particles are size reduced and recycled. The total product is blended to achieve product uniformity. The oxide is then sampled and packaged in canisters which are

FIGURE 15.4: Simplified Flow Diagram of the Plutonium Product Facility

decontaminated, weighed, and placed in special containers that hold 32 kg of PuO_2 in four canisters. These containers are placed in a storage facility with a capacity of 40,000 kg; the storage facility also has forced cooling capability to remove heat from ^{238}Pu decay. This special container is later enclosed in a licensed, protective overpack for offsite shipment.

Secondary Product Precipitation and Filtration

The filtrate is held in a surge vessel to allow time for delayed precipitation and then refiltered in a polishing or secondary filter after adding more oxalic acid. These solids from the polishing filter are returned to the vacuum filter by backwashing with nitric acid.

Filtrate Treatment

The filtrate (about 2.4M HNO_3 , 0.1M $\text{H}_2\text{C}_2\text{O}_4$) is concentrated in a distillation column. Excess oxalic acid is destroyed in the boiling 12M HNO_3 bottoms. If necessary, potassium permanganate is added to assist in destroying excess oxalate in the concentrate in the sample tank. Then the concentrate is sampled for accountability and transferred to the third solvent extraction step in the main process building for recovery of plutonium. The distillate is combined with condensate from the plutonium calcination off-gas system, sampled for accountability, and transferred to the main separations facility for concentration in the low-activity waste concentrator.

15.3 DESCRIPTION OF EFFLUENT PROCESSING AND WASTE MANAGEMENT

Effluents in this study are materials that are deliberately discharged to the environment. These materials differ from wastes in that wastes are controlled or disposed of with the intent of isolating them from man's environment. This section describes the systems that treat and handle the effluents and wastes at the representative fuel reprocessing facility.

15.3.1 Effluent Processing

All of the liquid streams leaving either the plutonium product facility or the UF_6 facility are piped back to the main separations facility for liquid treatment, and no contaminated liquid effluents are released from these facilities. The main separations facility does not release radioactive liquid effluents. It discharges liquids such as blowdown from the cooling towers and boilers, demineralizer regeneration wastes, and single-pass cooling water discharges. These are all nonradioactive discharges under normal operations and are discharged to a retention pond. The liquids are retained in the pond before their discharge. Comprehensive sampling and analysis is done to assure that no radioactive liquid effluents are released to the environment via these discharges.

Airborne Effluents

The control of radioiodine, nitrogen oxides, and radioactive particulates in the effluent air is a major concern for processes in the main separation facility. In the plutonium product facility, minimizing release of plutonium

particulates is the primary concern. Minimizing the release of uranium particulates and gaseous fluoride is a major concern in the UF₆ facility. Each of the three processing areas has its own separate airborne effluent treatment system as described below:

Main Separations Facility

The airborne effluent treatment system for the main process off-gas system (DOG and VOG) is shown in Figure 15.5. The main process off-gas system starts with the dissolver off gases, which pass through a condenser and a knockout (de-entrainment) pot. The gases are joined by various other process off gases and then pass through a scrubber to remove radioiodine using HNO₃-Hg(NO₃)₂ scrub (Rogers 1980), and a water absorber to recover NO_x as nitric acid for reuse. NO_x scrubbers are installed on all operations generating NO_x for NO_x recovery for acid recycle and for environmental control.

The remaining process gases are added to this stream of gases. These combined gases pass through another condenser, another knockout pot, another HNO₃-Hg(NO₃)₂ iodine scrubber, a heater, and seven parallel banks of filter units. Each unit has a prefilter, a silver zeolite iodine adsorber, and two final stages of HEPA filters in series.

The treated gases are discharged to the main (100-m) stack. The building ventilation system for cells, galleries, and working areas (not shown in Figure 15.5) provides two stages of roughing-plus-HEPA bank filtration for exhausted air. The off gases from the concentrated plutonium process areas are first filtered by a roughing filter and two stages of HEPA filters before joining the building ventilation air where they are again filtered by a roughing-plus-HEPA filter bank.

UF₆ Facility

The airborne effluent treatment system for the UF₆ facility is shown in Figure 15.6. Generally, where uranium particles could be released, the effluent treatment consists of some form of filtration (assumed here to be cloth bags and HEPA filters) and possibly other particle collection systems. For cases where HF, F₂, or UF₆ gases are in potential effluents, chemical scrubbing is used for effluent treatment. In most cases, the chemical used in off-gas scrubbing is potassium hydroxide solution. Combustion gases that are produced at various steps for process heating are released to the atmosphere without treatment.

Plutonium Product Facility

The airborne effluent treatment system for the Plutonium Product Facility is shown in Figure 15.7. All off-gases from the process vessels pass through a condenser, a knockout (de-entrainment) pot, a heater, a roughing filter and two stages of HEPA filters in series. They are then routed to the final building ventilation system for one more stage of roughing-plus-HEPA filtration and exit through the main stack of the main separations facility.

FIGURE 15.5. Process Flow Diagram--Process Off-Gas System for Main Separations Facility

FIGURE 15.6. Process Flow Diagram--Airborne Effluent Treatment for UF_6 Conversion

FIGURE 15.7. Process Flow Diagram--Airborne Effluent Treatment System for Plutonium Product Facility

Exhaust air from the vacuum cleanup system used to clean up major spills in the dry process cells discharges through its sintered metal and HEPA filters to the recirculation dry air system. From there it is routed to the building ventilation exhaust duct where it passes through one more stage of HEPA filters before being discharged to the main separations facility main stack.

Estimated Effluent Releases. Table 15.2 presents approximate values for the amounts of radioactive material entering and leaving the main separations facility with their associated activity levels. The radioactivity levels are conservatively high since they are based on reprocessing 160 days and not 5 years after reactor discharge. Tabular data in Croff (1982) was used to construct this table. The residual fission products in the recovered uranium and plutonium are only $1\text{E-}5$ percent of those that entered the plant in the spent fuel. The effluent treatment system is expected to limit the release of particulate fission products to the atmosphere to less than $2\text{E-}7$ percent of the material entering the off-gas filtration systems.

In addition, a factor of $6\text{E-}4$ is applied to estimate conservatively the release from processes into the off-gas system. The ruthenium emission factor is higher owing to its semivolatile character. Time delays in reprocessing which would allow additional nuclide decay in the fuel, of course, would reduce the activity in plant effluents. The approximate 0.5 percent loss of uranium and plutonium is as cladding deposits (even with leaching), with dissolver solids, and with fission products to HLW.

The effluent control systems for the UF_6 facility are similar to those described for the fuel cycle conversion facility in Section 11 of this report; effluent amounts may be inferred by comparison of scale. The principal effluent for the plutonium products facility is particulate plutonium oxide; applying a process release fraction of $1\text{E-}3$ and an effluent control system emission factor of $2\text{E-}9$, the estimated daily release of plutonium is $9\text{E-}8$ g per day (approximately $1\text{E-}6$ Ci). Release of fission product carryover is very low. The Curie release may vary somewhat depending on the time the $\text{Pu}(\text{NO}_3)_4$ is in storage prior to oxalate conversion.

Other daily gaseous atmospheric discharges of nonradioactive materials are estimated as: $2.4\text{E}6$ kg of water vapor; 200 kg of NO_x (90 percent retention efficiency); $2\text{E}5$ kg of CO_2 , and 400 kg of SO_2 from onsite generation of steam; heat discharge of 67 MW; and lesser amounts of materials such as H_2S - 20 kg, HF - 10 kg, H_2 - 90 kg, and NH_3 - 1 kg. Approximately $1.3\text{E}7$ kg of water containing 1600 kg of nonradioactive contaminants is discharged daily.

15.3.2 Waste Management

The waste materials retained in the reference fuel reprocessing facility are radioactive solid and liquid wastes and nonradioactive solid wastes. All liquid process wastes are converted to solids in the reprocessing facility. All other liquid streams containing radioactive materials are treated and recycled for use.

TABLE 15.2. Estimate of Radioactive Materials Input/Output for Representative LWR
Fuel Main Separations Facility

Material	Total Material(1)		Form		Fractional Release Through Effluent Control Systems(2)
	Elemental kg/day	Ci/day	Chemical	Phase	
<u>Input to Facility</u>					
Spent Fuel	46.3	6E5	PuO ₂ , Actinide Oxides	S	(1)
	4752.5	16.4	UO ₂	S	(1)
	175.5	2E7	Fission Products	S,G	(1)
	1175.0	2E5 ⁽³⁾	Zr Cladding	S	(1)
	283.0		SS End Fit, Inc. Spacers	S	(1)
<u>Output from Facility</u>					
Airborne Effluents:					
¹⁴ C	3E-4	1.2	CO ₂	G	1
Noble Gases	29	6E4	Kr, Xe	G	1
⁸⁵ Kr	0.14	5.4E4	Kr	G	1
Tritium	3.7E-5	3.6E2	HTO	G,L	0.1
Iodine	1.4E-4	1.9E-5	I ₂ ,CH ₃ I	G	1E-4
Ruthenium	3.3E-10	4.1E-5	RuO ₄	G,S	5E-8
All Other Fission Pro- ducts	1.7E-10	2E-5	Oxides, Iodides	S	2E-9
Uranium	5.5E-9	1.9E-11	UO ₂	S	2E-9
Plutonium	5.3E-11	5.3E-7	PuO ₂	S	2E-9
All Other Transuranics	8.5E-13	2.3E-8	Oxides	S	2E-9
<u>Wastes</u>					
High-Level Waste	146	2E7	Acidic, Nitrates	L,S	NA
U, Pu, Actinides	49	4E5	Acidic, Nitrates	L,S	NA
Intermediate- Level	213	< 100	Acidic, Nitrates	L,S	NA

TABLE 15.2. Estimate of Radioactive Materials Input/Output for Representative LWR
Fuel Main Separations Facility (Continued)

Material	Total Material(1)		Form		Fractional Release Through Effluent Control Systems(2)
	Elemental kg/day	Ci/day	Chemical	Phase	
Fuel Cladding	1175	2E5 ⁽³⁾	Zirconium	S	NA
Hardware	283		Inconel, SS	S	
Other		~ 10		S	NA
<u>Products</u>					
Uranium	4715	16.3	UF ₆	S	(0.99)
Plutonium	45.2	5.9E5	PuO ₂	S	(0.992)

Key: G = Gaseous
L = Liquid
S = Solid
NA = Not Applicable

- (1) Values based upon one day operating at 100 percent of capacity.
(2) Numbers in parenthesis are not released to the environment.
(3) Activity for cladding, end fittings, and spacers.

All radioactive solid and liquid wastes from the main separations facility, the UF_6 facility, and the plutonium products facility are placed in interim storage as needed and/or are introduced into operations to reduce their volume and convert them to fixated, solid forms suitable for disposal in a federal geologic repository or regulated burial ground (non-TRU wastes). TRU wastes with surface dose rates in excess of 200 mr/hr are designated as remote handled waste, and are handled and stored using methods, equipment, and shielding to minimize operators exposure. These wastes are sealed in drums and specially designed containers. Wastes with surface dose rates at the container below 200 mr/hr are considered to be contact handled waste are placed in 210 liter (55 gallon) steel drums. The containers are stored temporarily onsite in specifically designated areas of the building. Waste types and their treatment operations are described briefly below. The waste management scheme is illustrated in the process flow diagram shown in Figure 15.8.

High-Level Liquid Waste

This waste contains the fission products and actinides in nitric acid as well as the treated dissolver solids and other high-level evaporator bottoms. The liquid volume is reduced by evaporation to about 400 liters per MTHM; this liquid is mixed with glass-forming elements and is fed continuously and directly to the joule-heated melter where the nitric acid is destroyed (Chapman 1978). A portion of the melt is removed periodically through the melter overflow and is solidified in 30-cm diameter, steel canisters. The canister is sealed with a weld. The melter off-gas is treated in an effluent control system for NO_x recovery and semi-volatile fission product retention. The concentrated intermediate level wastes are concentrated, and processed with the HLW (Slate 1981).

The 146 kg of elemental fission products and 49 kg of elemental actinides and 213 kg of elemental ILW produced daily are incorporated in glass. The glass to fission product (as oxide) ratio is 3 to 1 by weight. The density of the vitrified oxide waste is 3 g/cm^3 . A waste canister 30-cm in diameter with a glass monolithic height of 2.4 m has a volume of 0.178 m^3 or the equivalent of waste from 2-1/2 MTHM. Thus, the facility produces 2.7 canisters of solidified high level waste per day (810 per year).

Incorporation of fission products (160-day old fuel) directly into the waste form would produce a canister thermal loading of 32.5 kw which is unacceptably high. Three-year-old waste (after separation) would generate about 6.5 kw in a canister, and 5-year-old waste, less than 4 kw. The liquid waste would require cooling from the time of separation until solidification, or spent fuel reprocessing would be deferred for that time to allow appropriate radionuclide decay. The maximum canister heat rate value for repository insertion has not been finalized. If it is less than the heat rate for 5-year-old waste, additional storage, decay time must be allowed. The canister heat rate is estimated at 2.3 kw for 10-year-old waste.

Combustible Wastes

Waste solvents, other organic liquids, ion-exchange resins, activated carbon, and combustible trash containing actinides and fission products are

FIGURE 15.8. Process Flow Diagram - Waste Management System for Main Separations Facility

incinerated in a fluidized sodium carbonate incinerator in the presence of air. The off-gases are treated before discharge. The resultant ash is washed with water and the carbonate liquid is sent to salt waste treatment. The ash is transferred for waste immobilization.

Solids Decontamination

All solids are treated to reduce actinide content prior to disposal by leaching in nitric acid followed by a nitric acid rinse. This includes cladding hulls, incinerator ash, dissolver solids, noncombustible trash, filters, and failed equipment. Noncombustible trash and filters are compacted and placed into drums. Failed equipment is cut up and packaged appropriately.

Salt Waste

Aqueous waste streams containing sodium salts from TBP/carbonate scrub, incinerator ash liquor, analytical waste, and cooling water regenerate are scrubbed with kerosene and digested with formic acid. Off-gases are routed through NO_x control systems. The digested solution is subjected to ion exchange to remove contaminants. The solution is concentrated and sent for immobilization. The ion-exchange resin is stripped and washed with nitric acid to recover actinides and fission products.

Waste Immobilization

The hardware from fuel assemblies (Zr cladding, end fittings, spacers, etc.) is mixed with an equal volume of sand in special disposal containers, 61 cm in diameter and 3 m long with a loading depth of 2.4 m. The daily processing rate fills 60 percent of one of these containers (180 containers per year). The heat load for a full container is about 250 watts for fuel processed 5 years after discharge from the reactor. For the three main facilities, about 50 drums of concrete-matrix wastes are produced assuming generation of about 10,000 kg of trash; most of this would be TRU waste. Other typical solid wastes are summarized in Table 15.3.

Incinerator ash, salt waste, solid iodate, silver zeolite, and ion exchange resins are mixed with concrete and immobilized in drums; liquid wastes with low and intermediate activity levels are collected from various concentrators and used in the concrete immobilization process.

Nonradioactive Solid Wastes

Most of the nonradioactive solid wastes are generated by support activities. They originate in offices, lunchroom, shop areas, and utility areas, and are typical of those generated in the chemical and nonnuclear industries. They typically contain nothing peculiar to the nuclear industry. These solid nonradioactive wastes are disposed of in a nearby sanitary landfill operation.

The UF₆ facility also generates some spent potassium bifluoride electrolyte waste, waste calciner solid waste, and spent fluorinator bed materials. The electrolyte contains little or no uranium, and the solid wastes are drummed and buried in the sanitary landfill area.

TABLE 15.3. Summary of Typical Solid Wastes from the Representative Fuel Reprocessing Facility

Waste Material	Form	Package
Noncombustible Trash	Glass, Metals	Drums
Combustible Trash	Gloves, Clothing	Drums
Failed Equipment	Disassembled Metal Apparatus	Hulls Canister
Waste Calciner Discharge - UF ₆ Facility	Solid	Drums
Potassium diuranate Mud	Solid	Drums
Spent beds - UF ₆ Facility	Solid	Drums
Fluorinator fines	Solid	Drums
Spent electrolyte	Solid	Drums
Resins, silica gel, Vermiculite	Solid	Drums
Silver zeolite	Solid	Drums
Filters	Metal, Glass Fibers, Asbestos	Drums
Nonradinactive trash	Solid	Dumpsters

15.4 PLANT LAYOUT

This section summarizes the description of the site and the facility and its hardware.

15.4.1 Site Description

The reference LWR fuel reprocessing facility is assumed to lie on the 4.7-km² generic site D described in Section 8. The facility itself (with the inner security fence) requires an area of 0.32 km² within the larger plant site. The facility layout assumes a well-labeled perimeter fence exists around the total site to exclude the public. Another fence surrounds the smaller rectangular facility area and includes security entrances. The plant is located so that the main stack is 1 km away from 2 sides of the generic rectangular site. The liquid effluents are to be discharged onsite into the river that flows through one corner of the site.

15.4.2 Facility Description

The representative facility is based primarily on the Barnwell Nuclear Fuel Plant (BNFP) owned by Allied General Nuclear Services. The plant is located near Barnwell, South Carolina, and is nearly completed.

Three main process buildings are included in the representative reprocessing facility; these are the main separations facility, the UF₆ facility, and the plutonium product facility (PPF). An overall layout of these facilities is given in Figure 15.9. The reference facility is assumed to operate for 40 years.

Main Separations Facility

The main separations facility consists primarily of the main process building and the fuel receiving and storage station. It is designed and constructed to minimize release of radioactive materials to man's environment during routine operation and under accident conditions. At least two physical barriers (and frequently more than two) are used to contain the radioactive materials within the facility during operation. These barriers are typically the process equipment (vessels, pipes, etc.) and the building around the process equipment. In most cases, the building itself provides two barriers; the hot cell or room where the process equipment is located, and the outer building shell. Process equipment is fabricated from materials resistant to failure from corrosion. Where failure of process equipment, under conditions assumed to be credible, could result in major releases of radionuclides, the equipment design basis is designated as requiring special quality control and quality assurance.

Structural barriers are designed to contain process materials if primary equipment barriers are breached. The principal barriers are constructed of heavily reinforced concrete with partial lining of stainless steel added to areas containing large quantities of radiologically hazardous materials in corrosive acid solutions. Lining of walls also facilitates any required decontamination operations.

FIGURE 15.9. Overall Layout of the Representative Fuel Reprocessing Facility

The process structural barriers are generally termed radioactive process cells and are typically surrounded by maintenance or operating areas. The process cells containing the highest radiation levels are designed for remote maintenance (i.e., maintenance from outside the cell through the use of in-cell cranes or shielding windows and manipulators). Other process cells are designed for direct personnel entry using protective clothing and contact maintenance, but only after appropriate remote decontamination has been completed to allow safe entry.

These facilities are constructed of heavily reinforced concrete with walls up to 2 m thick in some areas, and are designed to withstand natural phenomena such as earthquakes and tornadoes. Noncritical areas are generally constructed of concrete slabs or insulated steel siding. The main separations facility has eight cells for shearing and dissolving, initial separations (3), uranium purification, plutonium purification, waste treatment, and remote maintenance. Areas are also provided for contact operations, solid and liquid waste storage, plutonium nitrate storage, service galleries, control rooms and areas, and cold chemical storage. The following auxiliary areas are also considered as part of the main separations facility: analytical laboratories, emergency utility area, administration buildings, maintenance shop, stores warehouse, cask wash station, cooling tower, retention pond and pump house, meteorological tower, utility area, main stack, liquid waste storage area, and sanitary waste system.

UF₆ Facility

The UF₆ facility consists of the process building, the fluorine manufacturing building, UF₆ cylinder storage area, and the chemical tank farm (KOH, AHF, ANH₃, etc.) The process building is a reinforced steel structure enclosed with corrugated aluminum siding and is made according to standard chemical plant construction guidelines. Building construction is described in more detail in Section 11.

Plutonium Product Facility

The plutonium product facility consists of one process building, contiguous to the main separating facility. This building meets the same structural, shielding, confinement, and ventilation requirements as the main separations facility. Included in the facility are cells for plutonium precipitation, for plutonium calcination to oxide, and for plutonium packaging (blending, sampling, canister loading and sealing, decontamination, weighing, and load out in two parallel cell lines). Plutonium containers are stored in a security area and cooled by forced convection. The two parallel cell lines are serviced by a common maintenance gallery. The facility also has shipping package loading facilities, remote maintenance galleries, an archive sample storage area, plutonium scrap recovery and recycle operations, contact decontamination areas, and waste storage areas as well as the normal support functions.

Tables 15.4 and 15.5 summarize process areas and process equipment for the three major facilities.

TABLE 15.4. Process Area Descriptions for the Representative Fuel Reprocessing Facility

Process Area	Overall Dimensions L x W x H, m	Perimeter Construction Material	Other Features
<u>Main Separations Facility</u>	105 x 87 x 30	Reinforced Concrete	
Remote Process Cell (RPC)	15.8 x 7 x 21.3	Reinforced Concrete and SS Floor Plan	35-T, 5-T cranes power manipulator, periscope, four windows and pairs of master manipulators
Remote Maintenance and Scrap Cell (RMSC)	7.3 x 7 x 21.3	Reinforced Concrete SS Floor Plan	5-T, 10-T cranes periscope, power and master manipulators, four windows hull transfer cart
High-Level Cell (HLC)	15.8 x 7.9 x 8.5	Reinforced Concrete SS Floor Plan	Centrifuge can be remotely removed, contact maintenance
High-Intermediate-Level Cell (HILC)	15.8 x 8.2 x 12.2	Reinforced Concrete SS Floor Plan	Electro pulse column is remotely replaceable contact maintenance
Intermediate-Level Cell (ILC)	15.8 x 8.2 x 11.6	Reinforced Concrete SS Floor Plan	Contact maintenance
Uranium Product Cell (UPC)	15.8 x 8.2 x 9.4	Reinforced Concrete SS Floor Plan	Contact maintenance
Plutonium Product Cell (PPC)	15.8 x 8.2 x 6.1	Reinforced Concrete SS Floor Plan	Contact maintenance
Control Room Area/Plutonium Nitrate Storage and Loadout (CRA/FNSL)	20 x 25 x 18.5	Reinforced Concrete SS Floor Plan	48 titanium slab tanks only
Fuel Receiving and Storage (FRS)	45.7 x 45.7 x 27.7	Reinforced Concrete SS Lined Basins	125-T, 10-T, 2-15T cranes; 2 unloading bays; 2 unloading pools

TABLE 15.4. Process Area Descriptions for the Representative Fuel Reprocessing Facility (Continued)

Process Area	Overall Dimensions L x W x H, m	Perimeter Construction Material	Other Features
Waste Tank Equipment Gallery (WTEG)	31 x 33.5 x 12	Reinforced Concrete	3 SS tanks
Emergency Equip- ment and Ventila- tion	35 x 40 x 18	Reinforced Concrete	NA
<u>UF₆ Facility</u>	61 x 30.5 x 35.1	Steel Structure Aluminum Siding	NA
F ₂ Manufacturing	30.5 x 16.8 x 9.1	Concrete Mat and Footing	NA
Process Building	61 x 31.5 x 35.1	Same as above	NA
Cylinder Storage Area	30.5 x 23	Reinforced Concrete	NA
Chemical Tank Farm	52 x 15 x 4.3	Concrete	NA
<u>Plutonium Product Facility</u>	22 x 49 x 16 19 x 28 x 16	Reinforced Concrete	NA
Conversion Feed Cell	12 x 1.5 x 4.7	Reinforced Concrete	NA
Conversion Contact Cell	12 x 5.5 x 12	Stainless steel interior, car- bon steel exterior	NA
Precipitation Cells	4.6 x 1.7 x 3.7	Reinforced Concrete	NA
Conversion Dry Cells	9.1 x 1.7 x 7.3	Reinforced Concrete	NA
Plutonium Storage Stations	14.3 x 9.1 x 3.7	Reinforced Concrete	NA

NA = Not available at this time
SS = Stainless steel

TABLE 15.5. Major Process Equipment Descriptions for the Main Separations Facility, UF₆ Facility, and Plutonium Product Facility

Equipment	No.	Construction Material	Capacity	Location	Size (LxWxH)(a), m
Fuel Transfer Conveyor	1	304L SS	1 Fuel Assembly	Remote Process Cell (RPC)	Cart: 7.2L Track: 30.5L
Fuel Bundle Shear	1	304L SS	1 Fuel Assembly	RPC	Head: 3 x 1.5 x 1.7
Chopped-Fuel Diverter	1	304L SS	NA	RPC	Cone: 2.4D x 3.8H
Dissolver	3	304L SS	6 MTHM/day	RPC	1D x 4.8H
Dissolver Baskets		304LSS	NA	RPC	NA
High Activity Waste Concentrator	1	Titanium			1.6D x 5.6L
Dissolver Transfer Tank	4	304L SS	380ℓ	High Level Cell (HLC)	0.1D x 4.3L
Feed Adjustment Tank	2	304L SS	8,500ℓ	HLC	1.8D x 3.7L
Dissolver Flush Accumulation Tank	1	304L SS		HLC	1.8D x 5.1L
Feed Surge Tank	1	304L SS	11,600ℓ	HLC	2.3D x 3.5L
Accountability Tank	1	304L SS	8,100ℓ	HLC	1.8D x 3.4H
General Purpose Concentration Feed Tank	1	304L SS	16,500ℓ	HLC	2.4D x 3.7H
Centrifuge (Dissolver Solids)	1	304L SS		HLC	0.122 Bowl

(a) G, diameter (D) x length (L) or height (H).

TABLE 15.5. Major Process Equipment Descriptions for the Main Separations Facility, UF₆ Facility, and Plutonium Product Facility (continued)

Equipment	No.	Construction Material	Capacity	Location	Size (LxWxH)(a), m
First Cycle Solvent Extraction Columns	4	304L SS		High-Intermediate Level Cell (HILC)	
High Activity Waste Condenser	1	304L SS		HILC	
Secondary Product Recovery Column	1	304L SS	3,800ℓ	HILC	
No. 1 Solvent Treatment Columns	3	304L SS		HILC	
General Purpose Concentrator	1	304L SS	1,400 ℓ/hr	HILC	
NO ₂ Absorber (Dissolver Off Gas)	1	304L SS	20,400 ℓ/mm	Intermediate Level Cell (ILC)	1.5D x 8.2H
Acid Fractionator	1			ILC	1.7D x 10.4H
Vessel Off-Gas Knock-Out Pot	1	304L SS	125,000ℓ	ILC	1.8D x 2.4H
Low Activity Waste Concentrator	1		240 ℓ/hr	ILC	1.2D x 2.1H
Waste Solvent Burner	1	Inconel	9 kg/hr	ILC	
No. 1 Iodine Scrubber	1	304L SS		ILC	0.9D x 7.0H

TABLE 15.5. Major Process Equipment Descriptions for the Main Separations Facility, UF₆ Facility, and Plutonium Product Facility (continued)

Equipment	No.	Construction Material	Capacity	Location	Size (LxWxH) ^(a) , m
No. 2 Iodine Scrubber	1	304L SS		ILC	1.7D x 8.5H
Second U Solvent Extraction Columns	2	304L SS		Uranium Product Cell (UPC)	
Silica Gel Beds	2	304L SS	587ℓ/hr	UPC	
No. 2 Solvent Treatment Columns	2	304L SS		UPC	
ICU Concentrator	1	304L SS		UPC	
Second Pu solvent Extraction Columns	2	304L SS		Plutonium Product Cell (PPC)	
Third Pu Solvent Extraction Columns	2	304L SS		PPC	
Pu Product Concentrator	1	Titanium	61ℓ		0.2ID x 4.1H
<u>UF₆ Facility</u>					
Uranyl Nitrate Concentrator	1			Process Building	
Uranyl Nitrate Hexahydrate Calclner	1	304L SS		Process Building	
Reduction Reactor	1	304L SS		Process Building	

TABLE 15.5 Major Process Equipment Descriptions for the Main Separations, Facility,
UF₆ Facility and Plutonium Product Facility (continued)

Equipment	No.	Construction Material	Capacity	Location	Size (LxWxH) ^(a) , m
<u>UF₆ Facility (cont'd)</u>					
Hydrofluorinator	2	Nickel		Process Building	
Fluorinator	1	Nickel		Process Building	
Cold Traps	3	Copper		Process Building	
<u>Plutonium Product Facility</u>					
Precipitator Feed Preparation Tank	1	304L SS	410 ℓ	Conversion Feed Cell	3.4 x 2.1 x 0.057
Filtrate Concentration Feed Tank	1	304L SS	210 ℓ	Conversion Contact Cell (CCC)	2.0 x 1.8 x 0.057
Process Vessel Off-Gas Knock-Out Pot	1	304L SS	210 scfm	CCC	1.8 x 1.5 x 0.05
Filtrate Concentrators	2	304L SS	892 ℓ/hr	CCC	5.3 x 0.86 x 0.05
Precipitator	2	304L SS	9 ℓ	One in Each Precipita- tion Cell	0.1D x 1.2H
Filtrate Surge Tank	2	304L SS	25 ℓ	CCC	0.92 x 2.7 x 0.05
1st, 2nd, 3rd State Digesters	6	304L SS	9 ℓ	One in Each State in Each Participation Cell	0.1D x 1.2H

TABLE 15.5. Major Process Equipment Descriptions for the Main Separations Facility,
 UF₆ Facility, and Plutonium Product Facility (continued)

Equipment	No.	Construction Material	Capacity	Location	Size (LxWxH)(a), m
Vacuum Filter	2	304L SS	5 kg/hr	CCC	0.46D x 0.18H
Drier - Calciner	2	304L SS	2.4 kg/hr	Blender Cell	0.4D x 3.1H

15.5 SAFETY-RELATED INFORMATION

This section presents the pertinent safety-related information that can be used to help identify the risks associated with operating the representative fuel reprocessing facility.

15.5.1 Operating and Maintenance Requirements

The representative facility is assumed to operate 24 hours per day, 7 days per week, 300 full rate days per year and to be manned by operating and security forces at all times. Total staff is estimated at 500, of which 100 are supervisory or professional. Many staff people are trained operators certified by the Nuclear Regulatory Commission.

Some equipment requires replacement and/or major repairs frequently where the service is relatively severe. The remotely replaceable equipment present special problems in removal because of their intensely radioactive contamination and large size. Maintenance of most process equipment is assumed to be comparable to replacing it every 20 years. Extensive remote decontamination is necessary before maintenance activities requiring direct access to the process cells can be done. All HEPA filters and roughing filters are replaced annually.

15.5.2 Input/Output of Major Materials

The inputs and outputs of radioactive material are presented in Table 15.2. Other probable material requirements for the representative facility consist of those materials given as inputs in Table 15.6 and packaging materials for storage and disposal of solid wastes. Miscellaneous materials such as office supplies, office furniture, routine maintenance supplies, etc., will also be required.

Besides the radioactive materials in the main separations facility and plutonium product facility, review of Table 15.6 indicates that several hazardous chemicals are stored in large quantities in one or more outdoor storage tanks of capacities of 80,000 or more liters. These chemicals are anhydrous ammonia, anhydrous hydrogen fluoride, nitric acid, sulfuric acid, kerosene, and possibly chlorine. TBP and KOH solution are also stored in large tanks, but present less hazard. The ammonia and HF present the greatest potential hazard in the event of a large release since they form nonbouyant clouds of vapor which move on the ground before warming and becoming bouyant.

Many of the input and output values in this description are judgmental estimates. Some numbers could be in error by up to two orders of magnitude. Even the better known input/output numbers are probably accurate only to within a factor of two. All values quoted require verification. Similar uncertainties prevail in the areas of materials added to the facility and of solid wastes from the plant.

15.5.3 Unit Operations Involved

The unit operations related to each box of the process flow schemes are indicated by the number appearing above the upper left hand corner of each box

TABLE 15.6. Chemicals and Materials Input to the Representative Fuel Processing Facility

Material	Form		Use Rate, kg/day
	Chemical	Phase(a)	
Activated carbon	C	S	1200
Alumina (b)	Al ₂ O ₃	S	200
Ammonia (b) (35,000ℓ)	NH ₃	L	500
Boron oxide	B ₂ O ₃	S	210
Calcium oxide (lime)	CaO	S	800
Calcium phosphates	Ca ₃ (PO ₄) ₂	S	1
Carbon dioxide	CO ₂	L	NA
Chlorine(b) (550 ℓ)	Cl ₂	L	2
Decontaminating agents	Assorted	S & L	NA
Dioctyl phthalate	C ₈ H ₄ = 1, 2[COO(CH ₂) ₇ CH ₃] ₂	L	1
Formic acid	HCOOH	L	1150
Gadolinium oxide	Gd ₂ O ₃	S	50
Halon 1301	CF ₃ Br	L	NA
Hydrazine(b) (6,400 ℓ)	N ₂ H ₄	L	100
Hydrogen fluoride(b) (72,000ℓ)	HF	L	2500
Hydrogen peroxide	H ₂ O ₂	L	5
Hydroxylamine nitrate	NH ₂ OH-HNO ₃	L	70
Kerosene(b) (15,000 ℓ)	Varies	L	225
Lithium fluoride	LiF	S	
Magnesium phosphates	Mg ₃ (PO ₄) ₂	S	1
Mercuric nitrate	Hg(NO ₃) ₂	S	4
Natural gas	CH ₄	G	84,000
Nitric acid (68%)(b) (15,000 ℓ)	HNO ₃	L	5000
Nitrogen (LN ₂ supply)(b) (53,000 ℓ)	N ₂	L	NA
Nitrogen tetroxide (1000 kg)	N ₂ O ₄	L	100
Oil, light fuel	Varies	L	7300
Oxalic acid	H ₂ C ₂ O ₄	S	50
Potassium fluoride	KF	S	15
Potassium hydroxide(b) (15,000 ℓ)	KOH	S, L	300
Potassium nitrate	KNO ₃	S	
Potassium permanganate	KMnO ₄	S	4
Propane(b) (68,000 ℓ)	C ₃ H ₈	L	NA

TABLE 15.6. Chemicals and Materials Input to the Representative Fuel Processing Facility (continued)

Material	Form		Use Rate, kg/day
	Chemical	Phase (a)	
Resins	AgX, IX	S	1
Silicon dioxide	SiO ₂	S	670
Sodium carbonate	Na ₂ CO ₃	S	50
Sodium fluoride	NaF	S	
Sodium monoxide	Na ₂ O	S	120
Sodium nitrate	NaNO ₂	S	160
Sodium sulfate	Na ₂ SO ₄	S	3
Sodium sulfite	Na ₂ SO ₃	S	<1
Sucrose	C ₁₂ H ₂₂ O ₁₁	S	
Sulfuric acid (93%)	H ₂ SO ₄	L	65
Tributyl phosphate ^(b) (24,000 l)	[CH ₃ (CH ₂) ₃ O] ₃ PO ₄	L	140
Trisodium phosphate	Na ₃ PO ₄	S	<1

a. Key: G = gaseous
L = liquid
S = solid
NA = Not available

b. Supplied from storage tank farm.
Quantity stored indicated.

in Figures 15.2 to 15.8; these numbers relate to the list of unit operations presented in Table 15.7. With many operations using flowing liquids and gases, it is not convenient to denote all the pumping operations. For liquids entering various columns, several pumps are driven by the same motor to maintain the proper ratio of feed materials. This information for UF₆ conversion was presented in Section 11 of this report.

15.5.4 Accident Driving Forces

An assessment of driving forces which would potentially cause accidents in the representative facility is presented in Table 15.8. Tentative qualitative values for and the locations of the forces are given. The information for the UF₆ facility is presented in Table 11.6 of this report.

15.5.5 Containment and Shielding Systems

Shielding for the representative main separation and the plutonium products facilities is provided by concrete walls and by leaded glass windows where observation and/or remote manipulation of process equipment is necessary. This shielding is up to 2 m thick for the reprocessing facility high level cells and is not as thick where dilute liquid streams are processed such as in ion exchangers or general concentrators. Portable shields are used in the plutonium product facility in the transport of the product canisters from the end of the cell line to storage; these shields consist of a lead layer for the residual fission product gamma, and organic material for the ²³⁸Pu neutrons.

Permanent shielding is designed to maintain average dose rates below specified levels depending on the functions conducted in the area. For normally occupied work areas such as control rooms and manipulator stations, the average dose rate is not to exceed 1 mR/hr. Maintenance galleries and remote maintenance areas are designed to limit the average dose rate to 10 mR/hr. Shielding is designed to limit the dose rate to less than 0.5 mR/hr in non-nuclear areas such as offices, change rooms, and cold storage within the building. Operations in the cask unloading and fuel element storage pools are conducted underwater at a depth necessary to keep the operator average dose rate below 1 mR/hr. This requires about 4 m of water.

Shielding thickness for waste storage areas (liquid and solid) is commensurate with the waste being stored. All high-level waste operations are conducted remotely behind thick shielding including storage of waste-filled canisters and hulls containers. Most compaction and drumming is done behind moderate shielding.

Confinement of radioactivity in the facilities is provided by treatment and adsorption of process off-gas, by filtration of ventilation air, ion exchange treatment of liquids, and by the process equipment, cells, fuel cladding, plutonium product canisters and packages, waste canisters and drums, and by UF₆ cylinders.

The heating, ventilation and air conditioning system provides confinement for volatile, semi-volatile, and particulate airborne radionuclides in the

TABLE 15.7. Unit Operations Identified in Process Flow Diagrams (Figures 15.2 to 15.8)

Solids

1. Solids materials handling (packaging, conveyance, screening, mixing, pressing)
2. Solids materials cutting (metallic materials)
3. Solids materials size reduction (crushing, grinding)
4. Solids drilling/blasting
5. Solids materials melting (welding, glass making)
6. Solids materials burial

Liquids

7. Liquid materials handling (pumping, mixing)
8. Distillation (concentration, evaporation, fractionization)
9. Natural evaporation
10. Solvent extraction (organic/aqueous)
11. Electrolysis

Gases

12. Physical gaseous separation (diffusion, centrifugation)
13. Gas/vapor compression

Solid-Liquid

14. Solids dissolution (leaching, dissolving)
15. Solid-liquid separation (filtration, settling)
16. Solid-liquid sorption (adsorption, ion exchange)
17. Precipitation
18. Solidification at nonhigh temperatures (cementing, bitumenization)
19. High-temperature processing (drying, roasting, calcining, melting)

Solid-Gas

20. Solids separation (filtration)
21. Sorption

Other Phase Combinations

22. Sorption (scrubbing [l-g])
23. Incineration
24. Vapor/gas cooling (cooling, condensing, sublimation)
25. Chemical reactions (g-g, g-l, g-s, l-l, l-s)
26. Storage (s, l, g)

TABLE 15.8. Accident Driving Forces for the Representative Fuel Reprocessing Facility(a)

<u>Accident Driving Force</u>	<u>Location</u>	<u>Quantitative Estimate</u>
Temperature	Uranyl Nitrate Concentrator	50 - 150°C
	Scrub Waste Concentrator	50 - 150°C
	High Level Waste Concentrator	50 - 150°C
	General Purpose Concentrator	50 - 150°C
	Service Concentrator	50 - 150°C
	Acid Fractionator	50 - 150°C
	Low Activity Waste Concentrator	50 - 150°C
	Plutonium Product Concentrator	50 - 150°C
	Plutonium Filtrate Concentrator	50 - 150°C
	Plutonium Calciner	> 150°C
	High Level Waste Melter	> 150°C
	HTO Traps	< ambient
	Krypton FCRS	< ambient
Pressure	Trash Compactor	1 MPa
	PuO ₂ Canister Decontamination	0.1 to 1 MPa
Chemical Reactivity	HF Storage	High
	Kerosene Storage	Flammable
	NH ₃ Storage	Low
	HNO ₃ Storage	High
	H ₂ SO ₄ Storage	High
Potential Energy	Cask Handling Pool	Cask drop, fuel drop
	Fuel Storage Pool	Fuel drop
	Fuel Conveyor	Fuel drop
	Waste Storage	Canister or drum drop
	Plutonium Cells	Blender drop
	Plutonium Cells	Canister drop
	Plutonium Facility	Container drop
Thermal Energy	Fuel Storage Pool	25,000 kW
	HLW Liquid Storage	19,500 kW
	HLW Container Storage	9,000 kW
	Hulls Container Storage	675 kW
	PuO ₂ Storage Facility	600 kW
Nuclear Reactivity	Fuel Storage Pool	> 500 g fissile
	Fuel Element Shear	> 500 g fissile
	Dissolver	> 500 g fissile
	First Cycle Extraction	> 500 g fissile
	Second and Third Cycle Plutonium Extraction	> 500 g fissile
	Plutonium Nitrate Storage	> 500 g fissile
	Plutonium Conversion	> 500 g fissile
	HLLW Concentrator	> 500 g fissile
	HLW Melter	> 500 g fissile

(a) For main separation and plutonium product facilities.

main separation and plutonium product facilities. The system employs a once-through design with controlled in-leakage. Air flow is from areas of no contamination to those with the highest contamination, e.g., from office areas to work areas to cells. Cells and contaminated areas are maintained at a pressure several inches of mercury below the associated work area to assure inflow of air to the cell in the event of a loss of confinement. Ventilation air from non-nuclear areas is exhausted directly to the stack; however, it is monitored and is diverted automatically to a treatment system if radioactivity is detected. Process off-gas is treated and monitored continuously; flow is stopped if radioactivity levels exceeds established limits after the final treatment stage.

15.6 ALTERNATE PROCESS SCHEMES

The representative process scheme described is typical of that expected in the near to medium term. This section discusses a host of other possible separations processes, and off-gas clean-up processes.

15.6.1 Alternate Fuel Reprocessing Methods

As stated earlier, nuclear fuel reprocessing technology has been studied since the early 1940s. The original reprocessing of irradiated uranium by the Bismuth Phosphate process at Hanford only recovered plutonium for weapons and left a large amount of fission product waste which contained the valuable uranium; this waste was stored in underground tanks. In the mid-1940s, considerable research in volatility, adsorption, and solvent-extraction methods was conducted to identify methods to recover the uranium and reduce the volume of the waste.

Solvent extraction studies showed that nitrates of U and Pu were soluble in certain organic liquids while nitrates of fission products were not, and that U and Pu could be separated by valence adjustments. The first successful process for recovering separated, decontaminated U and Pu was the REDOX process which used hexone as solvent and aluminum nitrate to improve separations. This process also was continuous. In the early 1950s, this was almost immediately replaced by the PUREX process which used diluted TBP as the solvent and nitric acid as the salting agent in the aqueous phase. The PUREX process offered significant improvements in reduction of waste volume, greater flexibility in process conditions, less solvent fire hazard, and decreased operating cost.

These processes were for aluminum clad, slightly enriched uranium used for defense purposes. They have been applied to enriched fuel, and chop-leach techniques were developed to process stainless steel and zircalloy-clad fuels. While many improvements have been made in the process, the basic operations are the same as they were 30 years ago. Of the six reprocessing facilities operated or planned in the U.S., five (Hanford, Savannah River, Idaho, West Valley, and Barnwell) employed the PUREX process. The now closed Midwest Fuel Recovery Plant employed a combined PUREX/Aquafluor process (see description in this section).

Other reprocessing methods which have been studied are described briefly below. Most of this information is taken from Long (1967) and from Murray (1979), each of which present many useful references.

Butex

At Windscale, U.K., the basic process also uses a single uranium cycle and two plutonium cycles using a different solvent for solvent extraction. The uranium contamination prevents direct handling of the plutonium. The process uses dibutyl ether or diethylene glycol as the solvent, and ammonium nitrate and nitric acid as salting agents. All solvent extraction processes are distinguished by their choice of solvent to accomplish the desired separation.

Chelation

A solvent extraction process effective only to produce decontaminated plutonium. The most common chelating agent is thenoyltrifluoroacetone (TTA) in benzene or other solvent with nitric acid salting agent.

Halex

Similar to PUREX except TBP is diluted in carbon tetrachloride. A uranium and a plutonium product are produced.

Ether Process

Pentaether, diethyl ether, or butyl ether have been studied as solvents to produce uranium and plutonium product.

Hexane-25 Process

A variation of the REDOX process for enriched uranium fuels; mercuric nitrate is used as a catalyst in the aluminum dissolution step, and ammonium hydroxide is sometimes used for acid adjustment. It has three cycles of extraction and stripping for uranium.

TBP-25 Process

A modified PUREX process for enriched uranium fuel with TBP/kerosene in the first cycle and hexone in the second and third cycles. Salting agents are aluminum nitrate and nitric acid.

Flurex

An aqueous process for purifying uranium by electrolysis whereby the uranyl ions are attracted to a cathode after passing through a semipermeable membrane; the precipitated product is ammonium uranium fluoride.

Oxidation-Reduction Processes

The standard PUREX and bismuth phosphate processes are oxidation/reduction processes in that various valence states of plutonium are used to separate it from uranium fission products. Other precipitation processes based on different solubility for differing oxidation states (with products separated) are:

Lanthanum Fluoride (Np, Pu); Bismuth Arsenate (similar to Bismuth Phosphate); Zirconium Phosphate (Np, Pu); Double Sulfate (Np, Pu, Cm, Am); Ferric Hydroxide (Pu); Sodium Uranyl Acetate (Np, Pu, Am); and Thorium Iodate (Pu). These processes are just as disadvantaged relative to solvent extraction as is the Bismuth Phosphate process.

Civex

This is an antiproliferation concept modification of solvent extraction wherein the conditions of partitioning are adjusted to obtain a stream with all the plutonium mixed with the desired amount of uranium and contaminated with fission products ($DF = 2$), and a stream with the rest of the uranium with fission products. The U/Pu stream (80/20 w/o) is coprecipitated and refabricated into breeder fuel. The proposal called for separation of the uranium from fission products by the Fluoride Volatility process.

Battelle ANOX

Battelle conceived an antiproliferation coprocessing scheme in which the higher actinides were returned to reactor in the fuel as a waste management measure (Pobreskin 1977). For LWR fuel the mixed plutonium/uranium stream required addition of nonstrategic enriched uranium to reconstitute its reactivity. The mixture of higher actinides with fission products is extracted with di(2-ethylhexyl) phosphoric acid (HDEHP) in kerosene to remove the actinides which in turn are recovered in an aqueous phase and added to the contaminate U/Pu stream which is coprecipitated and calcined to oxide for fuel refabrication. Both CIVEX and ANOX require remote handling for all operations owing to high fission product content.

Fluoride Volatility

Uranium and plutonium are selectively volatilized by fluorination to separate them from the nonvolatile fluorinated fission products. The fuel prior to fluorination must be finely divided; this can be accomplished by chlorination of aluminum or zirconium clad fuels or oxidation of chopped stainless steel clad fuel. Some other materials are carried out of the process in chlorination and must be converted to oxides for trapping; NaF traps are used to trap impurities carried out with UF_6 and PuF_6 . Fluorination is done at $500^\circ C$ on a fluidized bed of Al_2O_3 powder. The Al_2O_3 and NaF become part of the waste. The uranium and plutonium are separated by thermal decomposition of the PuF_6 to PuF_4 .

The fluorination and the waste products are corrosive to the materials of construction. There are no liquid wastes and the fission product waste volume is small. The process has been used in a 5000-ton per year U_3O_8 manufacturing plant for uranium purification (see Element 12). Recent studies have been directed toward reprocessing of breeder fuels.

Molten Salt

The Zircalloy clad U or U alloy fuel and HF are added to a molten salt (NaF) dissolver; the ZrF_4 and UF_4 then enter fluorination, and UF_6 is

sent to NaF traps for purification. The fission products and plutonium are removed in a waste salt stream; plutonium is very difficult to extract. This process is used in a homogeneous mode for continuous reprocessing of the MSBR fuel.

Aquafluor

This is the process that was to be used in the now-closed Midwest Fuel Recovery Plant at Morris, Illinois. It combines aqueous and fluoride-volatility methods with single cycle solvent extraction for primary fission product removal. The U/Pu stream is concentrated and partitioned in a continuous ion exchange (IX) unit; a second IX is used to separate uranium and neptunium. The uranium is calcined and converted to UF_6 for final purification by distillation and adsorption. Plutonium and neptunium exit as nitrate solutions.

Fractional Crystallization

These are pyrometallurgical processes for metallic fuels. Impurities are removed by carefully controlled temperature conditions which influence the solubility of uranium or fission products. Representative processes are: Pyrozinc (melt alloy fuel in zinc); Zone Refining; and Hermex (mercury as a solvent). Pyrozinc has been proven on a commercial basis at ANL.

Partitioning Methods

These processes depend on distribution of plutonium and fission products between molten uranium and immiscible molten materials such as silver, magnesium, halide salts, aluminum, and tin. None except magnesium ever advanced beyond the laboratory stage, and all have received little attention lately except the Tin-Nitride process at Battelle which was not promising.

Selective Oxidation (Melt Refining)

This process takes advantage of the fact that impurities in a molten metal will enter a nonmetallic, removable slag. Slag properties are improved by additives. However, it is applicable to metallic uranium. A small facility was operated at the EBR-II Fuel Cycle Facility in Idaho in the late 1960s.

It is evident that many methods have been examined to separate uranium, plutonium, and the fission products and other actinides. They each have some perceived advantage but most have never advanced beyond the laboratory or pilot plant scale. They also have many disadvantages such as use of more hazardous chemicals, elevated temperatures, more equipment, etc. This review shows the emphasis placed on solvent extraction for reprocessing LWR fuels. Since these alternate processes are either not projected for commercial use, or not well developed, it is not necessary to present safety related information associated with them.

15.6.2 Alternate Gaseous Effluents Treatment Methods

Presently, fission product gases from reprocessing other than iodine are released to the atmosphere. Recent changes in regulations require the control of krypton discharges (U.S. EPA), and probably will require removal of tritium (T), krypton, carbon, and semi-volatiles such as ruthenium, cesium, and technetium; the release of the latter elements are more significant for the waste solidification process. There have been a number of alternative technologies studied for controlling the release of these elements as well as for iodine. For an integrated system to control all radioactive gaseous effluents, the order of treatment would be for ruthenium, tritium or iodine, carbon, and krypton. This system would probably also need an NO_x destructor to enhance the operation of some of the control systems.

The gaseous effluent control components for an integrated system are described below. Several alternatives for each of these are also described. A total off-gas control system is described later.

Ruthenium

Ruthenium oxide is the most volatile of the semi-volatiles released into the off-gas system. The off-gas from the dissolver (and the high-level waste joule-heated melter) are passed through a silica gel bed to trap the volatile ruthenium. Two sets of traps in parallel are installed to allow continuous trapping while changing out a filter bed cartridge; time for change out is signaled by a detector when break-through occurs. Ruthenium trapping becomes more important if the fuel is oxidized (Story 1978) by roasting (Voloxidation) prior to being dissolved. Prior oxidation of the fuel (480 C for 4 hours) breaks down the fuel by volume expansion (UO_2 to U_3O_8) which facilitates a more efficient dissolving step and significantly reduces NO_x generation in the dissolver. Voloxidation also liberates the tritium (as HTO, tritiated water) and the krypton/xenon still remaining in the fuel after shearing. The silica gel bed also traps other semi-volatiles; other fission product particulates are trapped in various filters (porous metal, HEPA, etc.) throughout the system.

Tritium

In the main separations plant described in Section , approximately 10 percent of the tritium is released when the cladding is sheared, and the remainder is released in the dissolver solution; some of this is oxidized to HTO and remains in the aqueous streams in the plant. Buildup of tritium in aqueous streams could be treated by electrolysis or other methods to concentrate the tritium in aqueous fractions which could be immobilized by cementation (U.S. ERDA 1976). Most of the tritium, as HT(gaseous), from the dissolver would pass through the wet scrubber using $\text{Hg}(\text{NO}_3)/\text{HNO}_3$ as scrub solution. The HT from shearing and the dissolver then could be combined and catalytically oxidized to HTO and the HTO is trapped in a molecular sieve (Yarbrow 1974). The sieves would be regenerated periodically and the tritiated water disposed of as immobilized low-level waste. Total water volume would be limited by exclusion of water from the shearing operation.

However, tritiated water can be removed ahead of any wet iodine recovery process. If voloxidation is used, the tritium would be released as HTO and the dissolver iodine could be recovered by dry methods (adsorption on zeolite) prior to molecular sieve trapping of the off-gas to remove the tritium. Tritium from the shearer would still require oxidation before being combined with the dissolver off-gas.

Iodine

Although relatively little ^{129}I is generated, owing to its very long half-life and its nature to concentrate in biota and human organs, its management is important. The possible Mercurex process described for the main separations plant is effective for removal of elemental iodine and is well developed. The scrub solution recirculates and the iodine is not separated. When the mercury/iodine mole ratio decreases to about four, the scrub solution is discarded in the immobilized waste.

There are other scrubbing methods. Caustic scrubbing is less effective for elemental iodine removal and again the solution is discarded as radioactive waste since the iodine cannot be separated (Evans 1980). In the Iodex process, the off-gas is scrubbed with 20 to 22M nitric acid which oxidizes all of the iodine compounds to higher non-volatile species, and iodine is removed as a solid iodate which can be immobilized (Hannaford 1977); this process is highly efficient but presents pyrophoric and corrosive hazards as well as requiring regeneration of the acid.

Iodine can also be removed from off-gas streams by trapping media (AgZ or AgX resin beds) (Murphy 1977). When the bed is loaded, it is disposed of in a drum of solid waste. Methods have also been investigated to regenerate these resins in place and then trap the iodine in a less expensive material for disposal. This involves stripping the elemental iodine out the bed as HI with hydrogen (4 volume percent in argon, 400 C) and then trapping the HI in PbX beds which would be disposed of as solid waste (Murphy 1977). These beds are sensitive to the presence of fluorides, chlorides, and NO_x in the off-gas.

Carbon

For large scale reprocessing ^{14}C , although very little is present in the off-gas, is a candidate for removal owing to its long half-life and its characteristic of concentrating in biological forms. The state-of-the-art system for removal of carbon (as CO_2) is by direct liquid scrubbing of the off-gases with a $\text{Ca}(\text{OH})_2/\text{NaOH}$ or $\text{Ba}(\text{OH})_2$ solution in which CaCO_3 or BaCO_3 is precipitated (Holladay 1978). For this process NO_x must be removed to prevent accumulation of nitrogen salts in the scrubbing solutions. NO_x removal requires heating the off-gas and mixing it with ammonia; by reaction in a 350 C bed of synthetic mordenite, molecular nitrogen and water vapor are formed (Pence 1974). The heated off-gas is then bubbled through the $\text{Ba}(\text{OH})_2$ solution.

A conceptual alternative control system parallel to the liquid scrubber systems consists of two distinct sets of molecular sieves to remove water vapor and then CO_2 in sequence from the treated off-gas (Brown 1978). The

CO₂ is then flushed from the sieve (during regeneration) in a more concentrated form in an inert, CO₂-free gas which is bubbled through a scrub solution. In this approach the H₂O can be removed with the water vapor after zeolite trapping of iodine. All liquid scrubbing techniques require much space and produce large volumes of secondary wastes.

CO₂ can also be recovered as one of the products from the fluorocarbon recovery system (FCRS) described below for krypton removal (Stephenson 1976). The recovery method and waste form for iodine have not been demonstrated.

Krypton

Krypton must be removed from the off-gas since recent regulatory decisions impose a requirement to reduce releases to approximately 10 percent of total input facility after January 1, 1983 (U.S. EPA). The krypton and xenon can be separated from off-gases by cryogenic absorption in a fluorocarbon solvent (FCRS). The operations of absorption, fractionation, and stripping are achieved in a single column where decontaminated off-gas leaves the top and enters the stack, regenerated fluorocarbon solvent leaves the bottom, and krypton/xenon and CO₂ leave as concentrated side streams (Stephenson 1978). The system has a moderate tolerance for water vapor and NO_x. Approximately 100 g of fluorocarbon absorbent are lost per day. Cryogenic separation of krypton from the xenon is necessary to reduce the volume of the gas by about a factor of 10; the DF for krypton is reduced by the xenon separation.

The use of a CO₂ removal process prior to krypton recovery would allow the use of a liquid nitrogen (LN₂) absorption system which cannot tolerate CO₂ (Brown 1978). However, this system also cannot tolerate water vapor, NO_x, and trace organics. Oxygen is carried through the process if not removed, and will form potentially explosive ozone as a result of radiolysis in the concentrated krypton. Oxygen is removed by reaction with hydrogen in a 400 to 500 C catalytic recombiner and then the water must be removed by trapping. The recombiner hydrogen also reduces trace NO_x to nitrogen and water vapor.

A solid adsorbent process has been tried with some success experimentally. Mordenite beds selectively remove xenon and then oxygen leaving a nitrogen/krypton mixture (Pence 1978). Krypton and nitrogen are then separated by cryogenic distillation or in another special adsorbent bed. Columns are regenerated by reverse flow of heated nitrogen. The basic technique has been used extensively for air reduction plants. Krypton decontamination factors are good (e.g., approximately 2 to 3), xenon separation is excellent, and oxygen separation to ppm levels is sufficient to eliminate ozone problem.

Once the krypton is separated, it must be contained for storage until it decays. This can be either as a gas in a pressurized cylinder (Foster 1975), or by fixation in solids by high pressure, high temperature sorption in zeolite (Knecht 1977), by ion implantation (sputtering) in metals (Platt 1978), or by sintered glass encapsulation (Platt 1978). Studies have continued to overcome the potential rubidium-caused corrosion problem in pressurized storage containers to assure 50 to 100 year life (Pinchback 1979). Brown (1978) presents an evaluation of zeolite encapsulation and the concept of a

facility with existing technology safety systems to contain any krypton releases. Platt (1978) presents laboratory results indicating high loadings of krypton in metals and glasses with good retention.

It is necessary to point out that only pressurized cylinder storage had advanced beyond the laboratory scale. Evans (1980) presents advantages and disadvantages for various separation and fixation methods for krypton, as well as for iodine and carbon, and also numerically evaluates them on the basis of simplicity, reliability, safety, cost, state of the art, material handling, and decontamination factor.

Comprehensive Off-gas Control System

A total off-gas control system should consist of the following elements:

- silica gel beds for semi-volatile trapping
- catalytic burner to convert HT to HTO
- molecular sieves for HTO trapping
- scrubbing to reduce NO_x content by at least 90 percent
- Iodex scrubbing (hyperazeotropic nitric acid) and silver zedite trapping to remove iodine (elemental and organic)
- molecular sieves for liquid vapor trapping
- molecular sieves for CO₂ trapping
- scrubbing of trapped CO₂ with hydroxide to precipitate carbonate
- cryogenic separation of krypton from remaining non-radioactive off-gases using a fluorocarbon absorption medium
- storage of separated krypton in storage cylinders for at least 40 years when its radioactivity will be less than 10 percent of the original amount.

Except for the wet processes (Iodex and CO₂ scrubbing in which a great deal of waste is generated) the other trapping and removal processes generate minimum amounts of waste and are reasonably easy to operate. Silica gel beds and silver zeolite beds are changed about once a year. Molecular sieves are regenerated by heating (and purging) to remove trapped material. Duplicate beds and sieves are installed so that one unit is always on line when the other requires service or maintenance.

The iodine and krypton removal processes present the greatest hazard. The Iodex process is pyrophoric and corrosive, and requires regeneration of hyperazeotropic nitric acid; however, it removes all iodine species with a very high DF of 6. The Mercurex process in the representative facility adds the disadvantages of Mercury toxicity and consumption of a valuable resource while only removing 99 percent of mercury (DF = 2). Both produce a

solid iodate in slurry form which requires disposal, the techniques of which are not well developed yet for fixation in concrete. Evans (1980) numerical rating indicate that fixed bed trapping silver minerals (DF = 3) are far superior for the six categories rated (see discussion above) including safety.

Of the three krypton recovery processes, the fluorocarbon system offers the highest DF (3 to 4) and the most advanced technology. Its disadvantages are high energy requirements (common to cryogenic systems), fluorocarbon releases, and a still underdeveloped xenon removal step (also common to cryogenic systems) which degrades the krypton DF.

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16.0 SPENT FUEL AND HIGH-LEVEL AND TRANSURANIC WASTE STORAGE

At this time, no commercial reprocessing of spent nuclear fuel to reclaim valuable uranium and plutonium is occurring. However, no decision has been made as to whether the irradiated fuel should be disposed of as radioactive waste. Even if reprocessing is started, producing high-level or transuranic wastes, or if spent fuel is declared a waste, it will be a number of years before a disposal repository will become operational. In the meantime, the spent fuel assemblies or high-level and transuranic wastes must be stored safely until they can be transported to the repository for final isolation from man's environment. Methods for interim storage of these materials are considered here.

This section describes conceptual facilities for interim storage of spent fuel, solidified high-level wastes, and transuranic isotope contaminated wastes. Section 16.A describes an independent spent fuel storage facility that utilizes the conventional water basin storage technique. Section 16.B contains a description of a solidified high-level waste storage system utilizing the dry-well storage technique. An interim transuranic waste storage facility is described in Section 16.C. The storage techniques are based on available technology and are representative of what could be achieved with current capabilities. Where possible, alternatives to the representative storage techniques chosen for these materials are discussed briefly.

16.A SPENT FUEL STORAGE FACILITY

A hypothetical spent fuel storage facility representative of current technology is described in this section. Currently, spent fuel from commercial nuclear reactors is almost exclusively stored in water basins at the reactor site. However, the capacities of several reactor storage basins are being increasingly utilized and there may be a future need to expand spent fuel storage capabilities by building Independent Spent Fuel Storage Facilities (ISFSF), also referred to as Away-From-Reactor (AFR) storage. These facilities are partly conceptual but utilize established technology in use at the following facilities today:

- at existing and planned commercial light water power reactors (LWRs)
- at the major DOE nuclear-related sites at Hanford, Savannah River, and Idaho National Engineering Laboratory
- at the General Electric and Nuclear Fuel Services reprocessing plants.(a)

The ISFSF described in this section is primarily modeled after the facility described in Technology For Commercial Radioactive Waste Management, DOE/ET-0028 (DOE 1979). This hypothetical facility is capable of storing up

(a) Reprocessing facilities are not operating at this time. However, spent fuel storage facilities are currently being used.

to 3000 MgHM in six water-filled pools, each with a capacity of 500 MgHM. Spent fuel is assumed to be received from either pressurized water reactors (PWRs) or boiling water reactors (BWRs) a minimum of 6 months after discharge. The overall characteristics of the representative ISFSF in this study include the following:

- There is one facility, capable of storing fuel assemblies containing 3000 MgHM in six water-filled pools, and one feed, spent fuel.
- The facility has the capability to receive and/or ship spent fuel at the rate of 1000 MgHM/yr or ~ 3.3 MgHM/day.
- The storage basin water is continuously cooled to remove decay heat and purified to maintain radioactivity in the water at or below $2\text{E-}4$ Ci/m³ so the radiation dose to operating personnel does not exceed 1 mR/hr.
- The ISFSF operates 24 hrs/day, 365 days/yr. About 210 days/year are required for cask and fuel handling operations.

16.A.1 SUMMARY

The representative independent spent fuel storage facility (ISFSF) is a "stand-alone" plant capable of storing fuel assemblies containing 3000 MgHM. Modular construction is utilized with storage pool capacity limited to 500 MgHM in each of six storage basins. Spent fuel is received from the reactor six months after discharge, stored for 6 years, then shipped from the facility. The age of incoming spent fuel will most likely be older than 6 months, but this value was chosen to be conservative. The average age of spent fuel in storage is conservatively assumed to be 3.5 years. The ISFSF is capable of receiving and/or shipping fuel at the rate of 1000 MgHM/yr or ~ 3.3 MgHM/day. Spent fuel is assumed to be shipped to the ISFSF at an average rate of 500 MgHM/yr for 6 years (3000 MgHM total). Thereafter, fuel that has been at the ISFSF for six years is shipped out at the rate of 500 MgHM/yr and replaced with the same amount of 6-month-old fuel from reactor storage pools. The design receipt rate corresponds to the annual spent fuel storage requirements for approximately 16 GW-yr of installed nuclear electrical capacity.

The representative ISFSF operations consist primarily of receiving spent fuel shipping casks (truck or rail versions), off-loading them from their transport vehicles, preparing for fuel unloading, unloading fuel assemblies and transferring them to the storage pool, and preparing the cask for shipping out. Provisions are made for radiation surveys of the cask, exterior washdown and decontamination, cooling and flushing the internal cask cavity, and unloading fuel assemblies under water. Fuel assemblies are placed in stainless steel storage racks (four PWR or nine BWR assemblies per rack) and transferred to the storage pool, also under water.

The majority of the shipping cask receiving and handling procedures are contact operations where unshielded and direct physical contact with the casks is required. These operations include radiation monitoring, washdown, removal

of tie-downs and ancillary equipment, attaching lifting mechanisms, untorquing cask head bolts, and attaching/detaching cooling and sample lines. Once the shipping cask has been placed in the unloading pool, the operations become remote. Remote operations include removing/replacing the cask inner and outer closures, removing fuel and transferring it to the storage pool, and placing the fuel in storage locations. Maintenance operations are generally performed in a contact manner, although flushing of some equipment (e.g., ion exchange columns) is required prior to contact maintenance.

Pool water is continuously cooled and purified. The maximum amount of heat generated by the stored fuel is about 12 MW, based on an average mix of BWR:PWR assemblies of 3:2 and an average exposure of 29,000 MWd/MgHM. The pool water cooling system is a closed system that transfers decay heat to the environment and maintains pool water temperature at or below 40°C. Pool water cleanup is provided by a filter-deionizer system that removes radionuclides, dissolved salts, and particles and maintains radioactivity in the pool water below $2\text{E-}4\text{ Ci/m}^3$.

Effluents from the representative ISFSF include cooling tower water, ventilation air and cask off-gas, and uncontaminated process water. Building ventilation air is normally untreated before it is released to the environment, but provisions are made for charcoal adsorption and filtration of particulates in the event that ventilation air contains quantities that exceed release guidelines. Cask ventilation off-gas and exhaust from other potentially contaminated areas (e.g., waste solidification and storage areas, etc.) is continuously filtered to remove radioactive material. A total of about 300 Ci is released annually from the ISFSF, most of which is ^{85}Kr .

The various liquid waste streams are treated to remove radioactive materials (in filter-deionizer system), the radionuclides are concentrated (in wet waste evaporators), solidified (in cement) and packaged in 210L (55-gal) drums. These drums are stored onsite to await shipment offsite for disposal. Liquid waste streams include water from cask-carrier washdowns, cask flush and cooling, cask decontamination, cask unloading pools (CUP) and storage pools, and filter backwash and deionizer regeneration solutions. The representative ISFSF produces about 26 m^3 (920 ft^3) of wet waste annually, which contain 2200 Ci of radioactivity. The activity in wet wastes is dominated by cesium and several activation products.

The solid radioactive wastes produced by the representative facility constitute 95% of the total waste volume but only 1% of the total activity. These wastes are shredded and compacted into drums and shipped offsite for disposal.

Several potential accident driving forces are identified for the representative ISFSF. Significant quantities of decay heat are generated by the spent fuel in storage or in shipping casks. High temperatures are also present where radionuclides are concentrated, e.g., ion exchange columns and the radwaste evaporators. Potential energy could drive potential cask and fuel handling accidents where some object is dropped from a crane. Chemical reactivity of HNO_3 and NaOH solutions, diesel fuel oil, and chlorine could

also result in an accident. Nuclear reactivity is a primary consideration for spent fuel storage facilities due to the large quantities of fissile material in storage and in shipping casks. Relatively high pressures are utilized in several of the ISFSF systems and are potential driving forces for accidents.

The ISFSF is assumed to be located on the generic site described in Section 8.0. The facility operates 24 hr/day, 365 days/yr and requires 36 operators, 7 maintenance personnel and 5 radiation monitors. The number of supervisors and professional staff are unavailable. Spent fuel receiving and handling operations can be completed in about 210 days.

Several alternative concepts to the representative water basin storage concept are identified and described. Some use forced cooling concepts and others are passively cooled systems. Those that use forced cooling are the water basin storage of packaged or encapsulated spent fuel and air-cooled vault storage concepts. Those that are passively cooled include open field dry caisson, metal and concrete cask, tunnel-rack, and tunnel dry well storage concepts. In all of the concepts, spent fuel may be stored as intact fuel elements or as "consolidated" fuel. Spent fuel consolidation consists of cutting or otherwise removing the assembly hardware (spacer grids, support plates, etc.) and placing the fuel rods in canisters. The canisters of consolidated fuel will be stored in a similar manner to the packaged or unpackaged intact fuel elements.

16.A.2 MAINLINE PROCESS DESCRIPTION

The primary functions of the ISFSF are to receive and store spent LWR fuel and to cool and cleanup storage basin water. An overall flow scheme of normal operations is shown in Figure 16.A.1. Spent fuel shipping casks are received on truck or rail transport vehicles, off-loaded from their transporters, and prepared to have fuel unloaded. The casks are then placed into a water-filled fuel unloading pool, the cask closures and head are removed, and the fuel assemblies placed in storage racks. The storage racks are transferred under water to the storage pool and the shipping casks are decontaminated, returned to their transport vehicles and prepared for shipping out. Retrieval of spent fuel from the storage basin and shipment from the facility are essentially the reverse of the operations above. These operations are discussed in further detail in Section 16.A.2.1.

Spent fuel is stored unpackaged in water-filled storage pools. Water provides a transparent medium in which the fuel can be handled and observed, a heat transfer medium, and radiation shielding for operating personnel. Basin water is continuously cooled by external heat exchangers to remove decay heat and purified by external ion exchange columns to remove dissolved and suspended radionuclides. A more detailed discussion of the ISFSF water treatment systems is given in Section 16.A.2.2.

16.A.2.1 Description of Cask and Fuel Handling Operations

This section describes the operations and facilities at the ISFSF that handle and store spent LWR fuel. See Figure 16.A.1 for an overall flow scheme of these operations.

FIGURE 16.A.1. Overall Flow Scheme for the Representative Spent Fuel Storage Facility (DOE 1979)

Shipping and Receiving

Irradiated fuel assemblies are shipped in massive, shielded, water-filled or gas-filled shipping casks transported by rail or truck. Incoming spent fuel shipping casks are received on their over-the-road vehicles (railcars or truck trailers) in the ISFSF shipping and receiving area. This area is also used to ship outgoing casks offsite or to another facility. Upon arrival at the facility, but outside the fence, the cask-vehicle is connected to a site-operated rail engine or truck tractor for subsequent movement to the storage building. Before this movement, each cask and transport vehicle is identified to verify shipping documents, inspected for damage that could occur during transport, and surveyed for radiation levels. If no irregularities are found by the inspection, the cask-vehicle is moved to the preparation area. If the cask-carrier is damaged, misidentified, or other irregularities are found, they are sent to an inside-the-fence holding area to await special unloading procedures. The holding area provides for temporary storage of the cask-vehicle system and periodic surveillance to ensure safe conditions. Cooling of loaded casks in this area is provided by the system used during transport. Cask-vehicle systems that are damaged are repaired in the maintenance area if it is essential to continuing operations on the cask. The ISFSF shipping and receiving area is also used to give cask-vehicles a final inspection before load-out to ensure that the shipment meets Department of Transportation (DOT) regulations.

Cask and Vehicle Maintenance

The maintenance area is provided to perform essential repairs and maintenance of casks, vehicles, and peripheral equipment. Normally, minor repairs and routine maintenance are performed on out-going shipping systems after fuel has been removed. However, if necessary, extensive work may be performed on loaded cask-vehicles to allow cask processing to continue.

Cask and Vehicle Preparation

The preparation area at the ISFSF is basically an airlock where peripheral equipment, such as personnel shields and crash shields, is removed by a crane. This equipment is checked for contamination and stored. The cask-vehicle is then moved to the cask unloading and loading area. After a cask has been unloaded, the peripheral equipment is replaced and the cask vehicle is transferred to the shipping and receiving area and readied for offsite shipment.

Cask Unloading and Loading

In the cask unloading and loading area, casks containing spent fuel are off-loaded from their vehicles and empty casks are reloaded onto vehicles. This is accomplished by disconnecting the cooling equipment and detaching tie-downs, attaching a crane yoke to the cask, lifting the cask from a horizontal position to a vertical position, and removing the cask from the carrier. Normally, the cask is moved to the cask washdown and cooling bay. However, casks that exceed limits for external contamination levels are

transferred to the cask decontamination pit for washdown. The vehicle is then washed and/or sandblasted, if necessary. A sump system collects the wash water in holding tanks for radiation checks to determine if treatment of the water is required for release.

Cask Decontamination

The cask decontamination is done in a pit fitted with a movable spray ring to facilitate decontamination of the cask. Casks are placed in the pit by the cask handling crane. Water, steam, or a decontamination solution is sprayed onto the cask. Effluents are drained into a sump system and collected in holding tanks for treatment.

Decontamination may be required to clean the surface of some incoming casks to reduce their surface contamination to levels required for immersion in the cask unloading pool. However, this area will be used for all casks returning from the unloading pools. Empty, out-going casks are flushed with deionized water to reduce cavity contamination levels. Next, any pool water left in the cask is drained. The exterior of the cask is washed, as before, to reduce surface contamination levels. Decontaminants include water, steam, and detergent solutions. This area is also where workers tighten the cask head bolts to the desired torque and check gaskets for leak-tightness before the cask is reloaded on its carrier.

Cask Washdown and Cooling

Cask washdown and cooling is done to prepare the cask for immersion in the CUP. Casks that are transferred directly from the unloading area (without the decontamination step) are washed to remove road dirt from the exterior surfaces. Wash water is collected by a sump system into holding tanks for treatment and/or release.

After the exterior washdown, workers check the cask primary coolant temperature and pressure to determine the condition of the fuel and coolant. Water can be circulated in the cask, if needed, to reduce the temperature before the cask is vented to the facility process off-gas system, and the coolant is sampled and analyzed for radioactivity. Normally, however, the casks will be vented before they are cooled. Cooling is accomplished by recirculating water (or steam) through the cask cooling coils and a heat exchanger to gradually reduce cask and fuel temperatures to avoid thermal shock when the cask is lowered into the cask unloading pool. When temperatures are acceptable, the cask coolant is sampled for radioactivity. Casks that contain leaking fuel elements or are contaminated internally are flushed with water until radioactivity levels in the flush water have stabilized, then transferred to an unloading pool. Casks that do not contain leaking fuel elements are transferred to a CUP by the cask handling crane when cask and fuel temperatures are acceptable.

Fuel Unloading

Spent fuel assemblies are removed from casks under water by a cask unloading gantry crane and transferred to storage baskets. The storage baskets are subsequently transferred out of the CUP by a basket handling monorail crane and the cask is removed from the CUP and placed in the decontamination bay. The CUP is physically separated from the storage pool so that a cask cannot tip over and damage stored fuel baskets.

The cask-handling crane is used to lower the cask into the water and place it on the bottom of the CUP. Once placed in the CUP, the cask lid is removed and placed on an underwater shelf. Individual fuel assemblies are then removed from the cask and placed in the fuel storage baskets. Fuel assemblies are lifted by a special grapple attached to the auxiliary hoist of the cask unloading crane. Assemblies with identified leaks are placed in special containers for leaking fuel. An underwater vacuum system is utilized to remove material that flakes or falls off fuel assemblies as they are removed from the cask to protect against contamination of the cask and to keep CUP water as clean as possible.

Fuel storage baskets are multiple-assembly configurations fabricated from stainless steel with boron added as a neutron poison and are designed to store four PWR assemblies or nine BWR assemblies. When a storage basket is full, the basket is moved out of the CUP and transferred to the storage basin. A special fuel storage area basket-handling crane is used for this operation. Storage baskets are described in more detail later.

After a cask is unloaded, it is visually inspected to ensure that all fuel assemblies have been removed. This may require the removal of the fuel baskets used to position the assemblies in the cask. If the visual inspection is satisfactory, the cask lid is replaced and the cask is removed from the pool. A high velocity, demineralized water spray ring at the surface of the CUP washes pool water from the exterior of the cask. Head bolts are replaced manually in the cask lid as it reaches the parapet level and then the cask is removed from the CUP and placed into the decontamination bay.

Special procedures are followed when unloading a cask that contains known or suspected leaking fuel assemblies. If these leaking fuel assemblies were not encapsulated at the nuclear power plant before shipment, special precautions must be made to prevent the spread of radioactive materials at the ISFSF. The CUP is isolated so none of its water can circulate to the storage basins, the flow of cooling water to the CUP heat exchangers is stopped, and a filter-deionizer unit is temporarily dedicated to the water of the CUP that will contain the leaking fuel. Known or suspected leaking fuel assemblies are placed in a special failed fuel container that permits sampling of the water in the container to identify the assemblies that are actually leaking. The cask can then be removed from the CUP by crane, with special precautions taken to prevent the spread of radioactive materials. The CUP remains isolated until radioactivity in the water is reduced to levels near that of the storage basin.

Fuel Storage

Spent fuel assemblies are cooled and shielded by the water in the storage basin. Fuel is contained in special baskets that are placed in racks attached to the floor of the pool. Criticality is prevented by the spacing and neutron absorption characteristics of the storage racks, even during an extreme external event such as an earthquake or tornado. Baskets are arranged to permit first-in-first-out processing of the spent fuel and to avoid excessive handling operations. The maximum flow of materials through the ISFSF is provided when incoming fuel is being placed in one pool and out-going fuel is being removed from another pool. It should be noted here that each storage pool can be sufficiently isolated from the others to permit drainage after all spent fuel has been removed.

16.A.2.2 Description of ISFSF Water Treatment Operations

Storage basin water must be treated to remove decay heat, radionuclides, dissolved salts, and particles. Figure 16.A.2 shows an overall flow scheme for the pool water cleanup and heat removal systems at the ISFSF. Differences in treating the water in the storage basin, transfer aisle, and the CUPs are shown.

Storage Basin Heat Removal System

Spent fuel stored in the water basins generates large quantities of decay heat. This heat must be removed from the water to maintain the basin at or below 40°C. The ISFSF requires a closed system to transfer the decay heat to the environment and minimize the potential for radioactive releases. The heat removal system must be capable of removing at least the maximum heat load on the ISFSF of 12 MW (DOE 1979).

Storage pools are cooled by pumping large flow rates of its water (primary cooling loop) through heat exchangers where heat is transferred to chilled water from a cooling tower (secondary cooling loop). The cask unloading pool is cooled by a different method. Coils are submerged in the CUP through which chilled water is pumped from cooling towers. The coils are protected against impact damage from cask and canister movement. Use of this type of cooling system provides for cooling of isolated pools and eliminates the need to pump large quantities of contaminated water from the pools.

Emergency cooling is provided by water pumped from an emergency cooling pond through a special distribution system. Emergency pumps are powered by diesel generators in the event offsite electric power is lost.

Storage Pool Water Cleanup System

The reference ISFSF utilizes a filter-deionizer system to maintain water activity levels at less than $2\text{E-}4 \text{ Ci/m}^3$ and to rapidly reduce the activity in any pool to normal levels in the event of a release of radioactive materials. Storage basin water activity of $2\text{E-}4 \text{ Ci/m}^3$ is designed to keep the average dose rate to operators below 1 mR/hr (DOE 1979).

FIGURE 16.A.2. Flow Scheme for Pool Water Cleanup and Heat Removal at the Representative ISFSF (DOE 1979)

16.A.3 DESCRIPTION OF EFFLUENT PROCESSING AND WASTE MANAGEMENT SYSTEMS

Effluents are materials that are deliberately discharged to the environment. These materials differ from wastes in that wastes are controlled or disposed of with the intent of isolating them from man's environment. Both types of materials are characterized as either radioactive or nonradioactive. This section describes the systems that treat effluents and wastes at the representative ISFSF.

16.A.3.1 Effluent Processing

This subsection describes treatment of gaseous and liquid effluents. Nonradioactive effluents include water from cooling towers, ventilation air (during normal operations) and uncontaminated water from certain process steps. Radioactive effluents include building ventilation air that has become contaminated, cask ventilation off-gas and ventilation air from the radioactive waste storage area. Building exhaust ventilation air is not normally treated and is routed directly to the atmosphere. However, if ventilating air picks up enough radioactive materials to approach release guidelines, a radioactivity monitor in the exhaust air stream activates a control damper that routes the air through a charcoal adsorber plus high efficiency particulate air (HEPA) filter system. A process off-gas system treats off-gas from cask venting and other areas that may release gaseous fission products, such as the solid waste encapsulation and storage areas. The process off-gas treatment system operates continuously. Only noble gas ^{85}Kr and small amounts of tritium and radioiodine are released to the atmosphere. A flow scheme for the effluent processing system is shown in Figure 16.A.3.

16.A.3.2 Waste Management

The ISFSF primary wastes consist of slurries, solutions, and solids that are treated and packaged for shipment offsite or disposal. Operations that generate radioactive wastes at the ISFSF are shown in Figure 16.A.4. This section describes the liquid and solid waste treatment systems at the facility.

Liquids

The various liquid waste streams at the ISFSF and their treatment processes are shown in Figure 16.A.4. Water that contains no detergent solutions is treated by passing through a filter-deionizer system to remove radionuclides. Detergent contaminated water, spent resins and regeneration solutions from deionizers and filter backwash are flushed to an evaporator to concentrate the radioactive materials. The resulting sludge is solidified by adding concrete to the sludges and allowing the mixture to cure in 210-l (55-gal) drums. The representative cement immobilization system is described in detail in ERDA-76-43 (1976). These drums are placed in a storage area to await shipment offsite for disposal.

Water from the cask washdown, vehicle wash, cask external decontamination, and leakage from the pool is collected in a radioactive drain system and

FIGURE 16.A.3. Flow Scheme for ISFSF Effluent Processing Systems

FIGURE 16.A.4. Flow Scheme of ISFSF Waste Management Systems

gravity fed to holding tanks. The water is sampled and either pumped to the filter-deionizer (high activity liquids) or to the radwaste evaporator (low activity liquids). Cask cooling and flush water is pumped directly to the filter-deionizer systems. Purified water is recycled to the makeup system and reused as pool water or for cask-vehicle washing operations. The CUP and storage pool purification systems are separate from the above filter-deionizer, but they also feed their resin regeneration wastes and filter backwash to the radwaste evaporator, and the deionized water is recycled to the storage pool.

The radwaste evaporators receive liquid wastes from filter-deionizer regeneration, low activity liquids from cask-vehicle wash operations, and decontamination solutions. There are two evaporators; the general purpose evaporator for mildly contaminated solutions and the high-level evaporator for resin-regeneration and filter backwash solutions. Decontamination solutions, estimated at about 760 m^3 (200,000 gal) per 500 MgHM received (DOE 1979), cannot be treated by the filter-deionizer because they contain detergent solutions that reduce the efficiency of the resins. These liquid wastes are concentrated by evaporation. The resulting steam is monitored for radioactivity levels and discharged to the atmosphere, and the concentrated sludge is pumped to the radwaste solidification system.

The water used for filter-backwash and resin regeneration contains radionuclides and small amounts of chemical additives to aid in filter-deionizer cleanup. This water is sent to the evaporators for concentration. The generation rate of this waste water is estimated to be about $24 \text{ m}^3/\text{day}$ (6,400 gal/day), assuming there is no failed fuel and the normal filter-deionizer flow rate (DOE 1979).

Solids

Solid radioactive wastes include failed equipment, maintenance materials, HEPA filters, and general trash. These materials are shredded and compacted to reduce volumes, where possible, and packaged for shipment offsite.

16.A.4 PLANT LAYOUT

This section contains descriptions of the ISFSF site and facility and its hardware.

16.A.4.1 Site Description

The representative ISFSF is assumed to be located on the generic site described in Section 8.0. The facility itself occupies about 0.08 km^2 (20 acres) inside the larger generic site. These facilities are enclosed within a security fence with two guarded penetrations; one for railroad shipments to enter and one for truck shipments and personnel to enter. This fence is posted with warning signs and is continuously and remotely monitored. The facility also includes 0.02 km^2 (5 acres) outside the fence for water disposal ponds, personnel parking, and an administration building. In addition, a well-labeled perimeter fence encompasses the entire site to

exclude the public. The plant is located so that the exhaust stack is about 1.0 km away from three sides of the generic rectangular site.

16.A.4.2 Facility Description

The representative ISFSF consists mainly of a cask and fuel handling area, the fuel storage basins, and the radioactive waste and water treatment area. Support, utility, and service facilities are also located within the secured area, as shown in Figure 16.A.5, the plot plan for the ISFSF. The facility is capable of receiving up to 3.3 MgHM/day of spent fuel and shipping an equivalent amount. A maximum of 3000 MgHM can be stored in this facility.

Cask and Fuel Handling Areas

Spent fuel is received at either the railroad guardhouse or the main security guardhouse. The shipments are unhitched from their over-the-road movers and hitched to site-operated donkey engines or truck tractors for subsequent movement. If unloading stations are occupied, casks are placed in outside-the-building parking areas to await handling. Two parallel bays inside the building are provided for cask and fuel handling operations. The cask and fuel handling area is constructed to Category I specifications with a reinforced concrete foundation, structural steel superstructure and "blow-away" roof deck and siding that is designed to blow away during a tornado and not fall into a basin. Cask unloading pools, fuel transfer canals and wash and decontamination pits are constructed below grade of reinforced concrete and are lined with stainless steel. The CUPs and transfer canals also meet Category I specifications. Figure 16.A.6 shows the ISFSF receiving area plan.

Other cask and fuel handling systems include:

- peripheral equipment handling crane (9 Mg capacity, overhead bridge crane)
- cask handling crane (115 Mg capacity with 9 Mg auxiliary hoist)
- basket transfer crane (9 Mg capacity, monorail crane)
- fuel inspection and failed fuel handling area
- cask and equipment storage area
- energy absorbing impact pads for CUP, decontamination bay, washdown and cooling bay, and areas between them (used to cushion a potentially dropped cask).

Fuel Storage Basin Systems

Spent fuel is stored in six water-filled storage pools, each with a capacity of 500 MgHM. Each pool is constructed below grade of reinforced concrete and is lined with stainless steel. The foundation, superstructure, roof deck and siding are constructed to Class I standards similar to the cask

16.A.16

FIGURE 16.A.5. Plot Plan for the Representative ISFSF (DOE 1979)

16.A.17

FIGURE 16.A.6. Receiving Area for the Representative ISFSF (DOE 1979)

and fuel handling areas. Storage pools are 9 m (30 ft) deep. Gates permit isolation of each storage pool. A plan view of the water basin storage area is shown in Figure 16.A.7.

Spent fuel assemblies are stored in multiple-assembly storage baskets which are placed in storage racks attached to the basin floor. Both the baskets and racks are constructed of stainless steel with boron added as a neutron poison for criticality control. It is assumed that four PWR assemblies are stored in the same space as nine BWR assemblies. The storage baskets are illustrated in Figure 16.A.8.

Other equipment and systems associated with the fuel storage basin area include:

- storage area crane (9 Mg capacity, overhead bridge)
- fuel storage racks
- basin water cooling systems (discussed later in this section)
- building ventilation system
- basin water purification systems (discussed later in this section)
- radiation and criticality monitoring systems.

The radioactive waste and water treatment area is constructed of reinforced concrete to Category I specifications. The superstructure in this area is designed for shielding, including cells. This area contains heat exchangers, filter-deionizer system components, wet waste evaporators, the waste solidification facilities, and the drum storage area. These facilities are placed in shielded hot cells and operated remotely. Solid waste shredding, compaction, and drumming systems are also operated remotely from behind shielded walls.

Additional information for the heat removal and filter-deionizer systems is as follows. Based on the storage facility at Barnwell, a flow rate of $1360 \text{ m}^3/\text{hr}$ (6000 gpm) on the primary side of the heat exchanger is required to remove the design heat load of 12 MW (DOE 1979). Each separate pool is equipped with a separate heat exchanger to permit its isolation from the other pools. Stainless steel, plate-type heat exchangers are utilized. The secondary cooling loop is maintained at a higher pressure than the primary loop so any leakage at this interface will not release radioactive material to the secondary loop.

The flow rate required for the filter-deionizer system was determined to be 340 to $410 \text{ m}^3/\text{hr}$ (1500 to 1800 gpm) which includes excess capacity for rapid clean-up (within 5 days) in the event of contamination of the basin water. This system consists of multiple units comprised of a prefilter, ion-exchange column, and after-filter. Deionizers are combined cation-anion

FIGURE 16.A.7. Plan View of Water Basin Storage Area at the Representative ISFSF (DOE 1979)

A = 71 cm MINIMUM
 B = 35.5 cm
 C = 24 cm
 LENGTH = 457 cm
 MATERIAL - STAINLESS STEEL

PWR Fuel Storage Basket

A = 71 cm MINIMUM
 B = 24 cm
 C = 16.5 cm
 LENGTH = 457 cm
 MATERIAL - STAINLESS STEEL

BWR Fuel Storage Basket

FIGURE 16.A.8. Illustrations and Details of PWR and BWR Fuel Storage Baskets (DOE 1979)

exchange columns that are regenerated by backflushing spent resins to a resin settling tank and recharging the column with fresh ion exchange resins. The prefilter is used to remove particulates that may clog the deionizer and reduce its efficiency. After-filters are used to remove resin particles that break free of the ion exchange columns. Multiple filter-deionizer units permit backflush and refilling of one unit while others are on line and permit dedication of units for rapid cleanup of excessively contaminated pools. Resin regeneration is provided for by flushing deionizers with HNO_3 and NaOH solutions. Regeneration waste solutions are flushed to holding tanks, then transferred to the wet waste treatment facilities.

Support Systems

The major systems and equipment ancillary to the water basin storage operations include:

- heating, ventilation, and air conditioning
- instrumentation and controls
- cooling towers (double cell, induced draft type)
- emergency cooling pond and its water distribution system
- offsite electric power and emergency diesel electricity generator and their distribution systems
- fire protection
- analytical laboratory
- hazardous material storage (NaOH , HNO_3 , diesel fuel oil)
- administration building.

16.A.5 SAFETY-RELATED INFORMATION

This section discusses the pertinent safety-related information that can be used to help identify the risks associated with the representative ISFSF.

16.A.5.1 Operating and Maintenance Requirements

The ISFSF operates 24 hours/day, 7 days/week, 365 days/year. Personnel requirements are as follows:

- operators = 36 man-yr/yr
- maintenance craftsmen = 7 man-yr/yr
- radiation monitors = 5 man-yr/yr
- professional staff - not available.

Based on receiving 50% of the spent fuel shipped to the ISFSF in 4 MgHM capacity rail casks and the remainder in truck casks that carry 0.4 MgHM, the number of shipments the facility must unload annually is 63 by rail and 625 by truck. Outgoing shipments are assumed to be the same. If all shipments are by rail, a total of 125 rail cask shipments are needed. If all one-way shipments are by truck, a total of 1250 shipments are needed. The turnaround time for a rail cask (General Electric Company IF-300 cask) is assumed to average approximately 32 h and that for a truck cask (Nuclear Assurance Corporation NAC-1 cask) is about 9 h (A. L. Franklin 1981). It is also assumed that it requires about 1 hour to move fuel storage baskets from the cask unloading pool to the storage pool and secure them in racks. Therefore, based on handling two casks simultaneously, about 210 days/yr are required to receive and ship spent fuel casks.

16.A.5.2 Input/Output of Major Materials

This section discusses the inputs and outputs of the major materials at the ISFSF. The materials covered in this section includes feeds, products wastes, effluents, and nonradioactive input materials.

Feeds and Products

The reference facility receives 500 MgHM of spent fuel annually, ships the same amount, or a combination of shipping and receiving that totals 1000 MgHM/yr. Incoming spent fuel is conservatively assumed to have decayed for 0.5 yr in the power reactor storage basin prior to shipment to the facility. Spent fuel is stored for 6 years and then shipped offsite to a reprocessing plant or to a packaging facility for disposal. The percentage of PWR fuel assemblies received and stored is assumed to be 40%, with the remainder being BWR assemblies. This corresponds to receiving and shipping on average, approximately 2000 BWR (380 MgHM) and 1350 PWR (620 MgHM) fuel assemblies annually. Table 16.A.1 shows the radioactivity content and heat generation rate in spent fuel at the assumed time of receipt (0.5 yr after reactor discharge), at the assumed average storage age (3.5 yr after discharge), at the assumed time of departure from the ISFSF (6.5 yr after discharge) and at 10 yr after discharge.

Effluents

Effluents are either radioactive or nonradioactive. Radioactive effluents consist of radionuclides released to the ventilation system by handling and storage operations. Routine nonradioactive effluents consist primarily of water released from the cooling towers. The annual release quantities of these effluents are shown in Table 16.A.2. Descriptions of effluent control processes were given in a previous section. The exhaust stack is assumed to be 62 m (200 ft) high and located 1 km away from 3 sides of the generic site. Effluents from sanitary water disposal ponds and the emergency cooling ponds were not available.

TABLE 16.A.1. Radioactivity Content and Heat Generation Rate in Spent Fuel^(a)

	Ci/MgHM			
	0.5 Years After Discharge ^(b)	3.5 Years After Discharge ^(c)	6.5 Years After Discharge ^(d)	10 Years After Discharge
<u>Fission Products</u>				
³ H	4.4E2 ^(e)	3.7E2	3.1E2	2.5E2
⁸⁵ Kr	9.5E3	7.9E3	6.5E3	5.0E3
¹²⁹ I	3.3E-2	3.3E-2	3.3E-2	3.3E2
⁹⁰ Sr + ⁹⁰ Y	1.3E5	1.3E5	1.2E5	1.2E5
⁹⁰ Zr + ⁹⁵ Nb	5.0E5	4.6E0	3.9E-5	NEG ^(f)
¹⁰⁶ Ru + ¹⁰⁶ Rh	6.8E5	8.4E4	1.1E4	6.8E2
¹³⁴ Cs + ¹³⁷ Cs + ^{137m} Ba	3.5E5	2.3E5	1.8E5	1.5E5
¹⁴⁴ Ce + ¹⁴⁴ Pr	1.2E6	8.4E4	5.8E3	1.6E2
All other FPs	5.6E5	5.4E4	2.6E4	1.2E4
<u>Actinide</u>				
²³⁹ Pu	2.9E2	2.9E2	2.9E2	2.9E2
²⁴¹ Pu	1.1E5	9.2E4	8.4E4	6.9E4
Other Pu	2.6E3	2.6E3	2.6E3	2.5E3
²⁴² Cm + ²⁴⁴ Cm	1.8E4	1.4E3	1.0E3	9.0E2
All other actinides	4.5E2	6.1E2	1.1E3	1.7E3
<u>Activation Products</u>				
¹⁴ C	7.5E-1	7.5E-1	7.5E-1	7.5E-1
⁵⁵ Fe	6.1E3	2.0E3	1.0E3	4.1E2
⁶⁰ Co	5.2E3	3.1E3	2.1E3	1.0E3
⁹⁵ Zr + ⁹⁵ Nb	1.1E4	1.0E-1	9.0E-7	NEG
All others	1.7E3	5.6E2	4.0E2	3.9E2
Heat generation rate, W/MgHM	1.6E4	2.7E3	1.4E3	8.9E2

(a) Reference: DOE 1979. Assumes an average mix of BWR:PWR fuel assemblies of 3:2 and an average burnup of 29,000 MWd/MgHM.

(b) Assumed time of receipt of ISFSF; multiply by 500 MgHM/yr for annual quantities received.

(c) Assumed average age during storage at ISFSF; multiply by 3000 MgHM for maximum storage inventory (see Section 15.A.5.3).

(d) Assumed time of departure from ISFSF; multiply by 500 MgHM/yr for annual quantities shipped.

(e) 4.4E2 reads 4.4×10^2 .

(f) NEG. = Negligible

TABLE 16.A.2. Annual Release Quantities of ISFSF
Airborne and Liquid Effluents^(a)

<u>Airborne Effluents</u>	<u>From Fuel Handling (Ci/yr)^(b)</u>	<u>From Fuel Storage (Ci/yr)^(c)</u>
Ventilation Air	Total: 3.7E9 m ³ /yr exhausted ^(d)	
³ H	0.19	2.6
¹⁴ C	1.2E-3	2.3E-5
⁸⁵ Kr	285	16.5
¹²⁹ I	1.7E-6	9E-7
¹³⁴ Cs + ¹³⁷ Cs + ¹³⁷ Ba	1.3E-2	6.3E-3
All other fission products	NEG ^(d)	NEG
Actinides	NEG	NEG
All other activation products	<u>NEG</u>	<u>NEG</u>
Total	285	19.2

Liquid Effluents

	<u>Release (kg/yr)</u>
Cooling Water:	
Evaporated (vapor)	2E8
Drift (aerosol)	1E6
Blowdown (liquid from tower)	3.5E7
Emergency cooling pond	NA
Sanitary sewage	NA

(a) Reference: DOE 1979.

(b) Based on receiving 500 MgHM/yr and shipping the same amount. Fuel has aged 0.5 yr and 6.5 yr out-of-reactor, respectively (assumed to be conservative fuel ages).

(c) Based on 3000 MgHM in storage, average of 3.5 yr after reactor discharge.

(d) NEG = negligible.

Wastes

ISFSF radioactive wastes consist of wet wastes, compactible and combustible trash, failed equipment, and noncombustible items. The volumes and radionuclide contents of these wastes are shown in Tables 16.A.3 and 16.A.4 for wet and solid wastes, respectively. In Table 16.A.3, wet wastes are segregated into resin beads, filter precoat sludges, resin regeneration

TABLE 16.A.3. Quantities and Radionuclide Content of ISFSF Wet Wastes Input to Waste Treatment Systems^(a)

Waste Stream	Chemical	Phase	Volume (m ³ /yr)	Average Activity (Ci/m ³)	Principal Radionuclides (Ci/yr) ^(b)						Total (Ci/yr)
					Handling ^(c)			Storage ^(d)			
					Ru+Cs+Te ^(e)	AOFP ^(f)	AP ^(g)	Ru+Cs+Te	AOFP	AP	
Resin beads	Polystyrene + water	S+L	2	~3.9	4.5	0.1	0.9	2.0	0.02	0.3	
Filter precoat sludges	Cellulose + diatomaceous earth + water	S+L	9	~85	456	5.7	90.1	198	2.4	30.4	783
Resin regeneration solutions	Na ₂ SO ₄ + water	S+L	6	~65	230	3.1	45.1	99	1.2	15.2	394
Detergent solutions	Decon. agents + corrosion product + water	S+L	9	~85	456	5.7	90.1	198	2.4	30.4	783
Total			26			14.6	226	397	6.0	76.	1870

(a) Reference: Black and King 1978.

(b) Numbers contain excess significant figures for ease of calculations. Contributions from actinides are negligible.

(c) Based on receiving 500 MgHM, 0.5 yr after reactor discharge and shipping same amount of 6.5 yr-old spent fuel.

(d) Based on storage of 3000 MgHM of 3.5 yr-old spent fuel.

(e) Ru+Cs+Te: combined contribution of ruthenium, cesium, and tellurium.

(f) AOFP: all other fission products.

(g) AP: activation products.

TABLE 16.A.4. Quantities and Radionuclide Content of ISFSF Solid Wastes Input to the Waste Treatment Systems^{(a)(b)}

Waste Stream	Form		Volume (m ³ /yr)	Average Activity (Ci/m ³)	Ru+Cs+Te ^(c)	AOFP ^(d)	AP ^(e)	Total
	Chemical	Phase						
HEPA filters	Wood, glass, paper	S	18	~0.01	0.15	NEG ^(f)	0.03	0.18
Combustible trash	Organic polymers, paper	S	600	~0.03	13.0	0.2	2.4	15.6
Failed equipment	Metallic	S	18	1E-3	0.01	NEG	0.002	0.012
Noncombustible trash	Metal, Glass	S	<u>60</u>	~0.025	<u>1.2</u>	<u>0.02</u>	<u>0.21</u>	<u>1.43</u>
Total			696		14.4	0.22	2.6	17.2

(a) Reference: Black and King 1978.

(b) Numbers contain excess significant figures for ease of calculation. Actinide content is negligible.

(c) Ru+Cs+Te: combined contribution of ruthenium, cesium, and tellurium.

(d) AOFP: all other fission products

(e) AP: activation products

(f) NEG: negligible contribution of total radionuclide content.

solutions, and detergent solutions. These wastes are input to the waste treatment system (evaporation, solidification, packaging, and short-term storage) and result in about 250 "standard" 210-l (55-gal) drums/yr of solidified wastes (Black and King 1978). Combustible trash is shredded and compacted into about 1890 drums annually. Used HEPA filters are also shredded and compacted and result in approximately 95 210-l drums/yr. Noncombustible trash fills about 270 drums/yr. On average, about 9 drums of waste per day are processed.

Nonradioactive Input Materials

Other materials are utilized at the reference facility for ion-exchange resin regeneration and for decontamination and cask/carrier washdown operations. Annual requirements of these materials are shown in Table 16.A.5. These materials exit the facility as constituents in wastes.

16.A.5.3 Inventories of Major Hazardous Materials at Plant Locations Having Safety Related Importance

The major inventory of radioactive materials at the reference ISFSF is located in the storage basins (spent fuel and basin water). Lesser amounts are in the CUP and receiving areas. Much less is in the filter-deionizer systems (radionuclides), cask unloading pool (spent fuel and CUP water), receiving area (spent fuel), evaporators (wet radioactive wastes), drum storage area (packaged radioactive wastes) and the cask off-gas filters (airborne radionuclides). Other hazardous materials are located in the make-up water treatment facility (chlorine) and the pool water treatment area (5% NaOH and 5% HNO₃). Table 16.A.6 shows the important safety-related information concerning these inventories of hazardous materials.

Several key assumptions were made to facilitate calculations of the radionuclide contents at various locations. The calculations were directed toward finding the maximum amount of radionuclides in each component during normal operations. Spent fuel in the storage basin is conservatively assumed to be an average of 3.5 yr old (measured from reactor discharge) and the

TABLE 16.A.5. Nonradioactive Input Materials

<u>Material</u>	<u>Use</u>	<u>Quantity (m³/yr)</u>
5% NaOH	IX bed regeneration	450
5% HNO ₃	IX bed regeneration	300
Detergent	Decontamination	3.8
Detergent	Cleaning	0.2
IX resin beads	Water treatment	2
Chlorine	Make-up water treatment	Not available

NOTE: NaOH and HNO₃ are most likely bought as dry solid or 50% solution and 60% solution, respectively.

Table 16.A.6. Inventories of Hazardous Materials at Various Plant Locations

Location/Material	Form		Quantity	Radionuclide Content (Ci) (a)(h)								
	Chemical	Phase		N.G.	I	SVFP	³⁴	AOFP	ULA	TRU	AP	Total
<u>Storage Basin</u>												
Spent fuel(c)	UO ₂	S	3,000 MgHM	2.2E7(a)	105	4.7E8	1.1E6	1.3E9	1.2E4	3.2E8	1.7E7	2.1E9
Pool water(d)	H ₂ O	L+S	13,400 m ³	0	1.3E-3	2.7	NEG(f)	NEG	NEG	NEG	1.7	4.4
<u>Cask Unloading Pool</u>												
Spent fuel(c)	UO ₂	S	4.6 MgHM	4.1E4	0.16	3.2E6	2070	1.2E7	20	6.0E5	1.1E5	1.6E7
CUP water(d)	H ₂ O	L+S	760 m ³	0	8E-5	0.15	NEG	NEG	NEG	NEG	0.1	0.25
<u>Receiving Area</u>												
Spent fuel(c)	UO ₂	S	9.2 MgHM	8.2E4	0.32	6.4E6	4140	2.4E7	40	1.2E6	2.2E5	3.2E7
<u>Ion Exchanger</u>												
Resin beads(e)	Organic Polymer	S+L	2 m ³	NEG	NEG	20	NEG	NEG	NEG	NEG	8	28
<u>Evaporator</u>												
Wet wastes(e)	H ₂ O+RM	L+S	1 m ³	NEG	NEG	80	NEG	0.9	NEG	NEG	12.7	94
<u>Evaporator Concentrate Holding Tank</u>												
Wet wastes(e)	Sludge	L+S	5.1 m ³	NEG	NEG	900	NEG	10.3	NEG	NEG	150	1060
<u>Drum Storage Area</u>												
Packaged wastes(e)	Many	S	65.6 m ³	NA(g)	NA	NA	NA	NA	NA	NA	NA	375
<u>Make-Up Water Treatment</u>												
Chlorine	Cl ₂	G	NA									NATURAL AMOUNTS
<u>Basin Water Treatment</u>												
Resin regeneration solutions	5% NaOH(f)	L	450 m ³									NATURAL AMOUNTS
	5% HNO ₃	L	340 m ³									NATURAL AMOUNTS
<u>Diesel Fuel Oil Storage</u>												
Diesel fuel	Hydrocarbons	L	38 m ³									NATURAL AMOUNTS

NOTE: This information was extracted from DOE (1979) and Black and King (1978).

(a) Numbers contain excess significant figures for ease of calculations; 2.2E7 reads 2.2×10^7 .

(h) Abbreviations are as follows: N.G. Noble Gases, I Iodine, SVFP Semi-Volatile Fission Products. Total contribution due to ruthenium, cesium, rubidium, tellurium, and antimony: ²³⁴Th, ²³⁴Pa, ²³⁴U, ²³⁸U, ²³⁵U, ²³⁹U, ²⁴⁰U, ²⁴¹U, ²⁴²U, ²⁴³U, ²⁴⁴U, ²⁴⁵U, ²⁴⁶U, ²⁴⁷U, ²⁴⁸U, ²⁵⁰U, ²⁵²U, ²⁵⁴U, ²⁵⁶U, ²⁵⁸U, ²⁵⁹U, ²⁶⁰U, ²⁶¹U, ²⁶²U, ²⁶⁴U, ²⁶⁶U, ²⁶⁸U, ²⁷⁰U, ²⁷²U, ²⁷⁴U, ²⁷⁶U, ²⁷⁸U, ²⁸⁰U, ²⁸²U, ²⁸⁴U, ²⁸⁶U, ²⁸⁸U, ²⁹⁰U, ²⁹²U, ²⁹⁴U, ²⁹⁶U, ²⁹⁸U, ³⁰⁰U, ³⁰²U, ³⁰⁴U, ³⁰⁶U, ³⁰⁸U, ³¹⁰U, ³¹²U, ³¹⁴U, ³¹⁶U, ³¹⁸U, ³²⁰U, ³²²U, ³²⁴U, ³²⁶U, ³²⁸U, ³³⁰U, ³³²U, ³³⁴U, ³³⁶U, ³³⁸U, ³⁴⁰U, ³⁴²U, ³⁴⁴U, ³⁴⁶U, ³⁴⁸U, ³⁵⁰U, ³⁵²U, ³⁵⁴U, ³⁵⁶U, ³⁵⁸U, ³⁶⁰U, ³⁶²U, ³⁶⁴U, ³⁶⁶U, ³⁶⁸U, ³⁷⁰U, ³⁷²U, ³⁷⁴U, ³⁷⁶U, ³⁷⁸U, ³⁸⁰U, ³⁸²U, ³⁸⁴U, ³⁸⁶U, ³⁸⁸U, ³⁹⁰U, ³⁹²U, ³⁹⁴U, ³⁹⁶U, ³⁹⁸U, ⁴⁰⁰U, ⁴⁰²U, ⁴⁰⁴U, ⁴⁰⁶U, ⁴⁰⁸U, ⁴¹⁰U, ⁴¹²U, ⁴¹⁴U, ⁴¹⁶U, ⁴¹⁸U, ⁴²⁰U, ⁴²²U, ⁴²⁴U, ⁴²⁶U, ⁴²⁸U, ⁴³⁰U, ⁴³²U, ⁴³⁴U, ⁴³⁶U, ⁴³⁸U, ⁴⁴⁰U, ⁴⁴²U, ⁴⁴⁴U, ⁴⁴⁶U, ⁴⁴⁸U, ⁴⁵⁰U, ⁴⁵²U, ⁴⁵⁴U, ⁴⁵⁶U, ⁴⁵⁸U, ⁴⁶⁰U, ⁴⁶²U, ⁴⁶⁴U, ⁴⁶⁶U, ⁴⁶⁸U, ⁴⁷⁰U, ⁴⁷²U, ⁴⁷⁴U, ⁴⁷⁶U, ⁴⁷⁸U, ⁴⁸⁰U, ⁴⁸²U, ⁴⁸⁴U, ⁴⁸⁶U, ⁴⁸⁸U, ⁴⁹⁰U, ⁴⁹²U, ⁴⁹⁴U, ⁴⁹⁶U, ⁴⁹⁸U, ⁵⁰⁰U, ⁵⁰²U, ⁵⁰⁴U, ⁵⁰⁶U, ⁵⁰⁸U, ⁵¹⁰U, ⁵¹²U, ⁵¹⁴U, ⁵¹⁶U, ⁵¹⁸U, ⁵²⁰U, 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maximum amount of spent fuel at the facility is 3000 MgHM.(a) In each cask unloading pool, a maximum of 4.6 MgHM (10 PWR fuel assemblies) can be present and the worst case fuel age would be 0.5 yr from reactor discharge. Data used for radionuclide inventories in spent fuel assemblies was previously shown in Table 16.A.1. Radionuclide concentrations in pool water were extracted from a report by A. B. Johnson (1977). The maximum inventory in the cask receiving area is two full rail casks or up to 20 PWR fuel assemblies. To calculate the radionuclide inventory in an ion-exchange column it was assumed that resin regeneration occurs after treatment of $6.1\text{E}4 \text{ m}^3$ (16E6 gal) of pool water (Black and King 1978) and that 100% of the pool water radioactivity is removed by the ion-exchanger. The maximum radionuclide inventory in the ion exchange columns occur just prior to regeneration. The maximum inventory in the wet waste evaporator was calculated by assuming the evaporator has a maximum volume of 1 m^3 (264 gal) and contains that volume of resin regeneration solution. The evaporator was assumed to reduce the waste volume to one-half the volume of the untreated waste stream. The inventory in the drum storage area is calculated by assuming the area is filled to its maximum capacity of 312 drums, or one month's storage (Black and King 1978). Radioactivity level is dominated by the drums containing solidified wet wastes that contain an average of 15 Ci/m^3 . There are assumed to be 25 of these drums produced every month (Black and King 1978).

16.A.5.4 Unit Operations Involved

This section identifies the unit operations involved for each step in the process flow schemes for the representative ISFSF (see Figures 16.A.1 through 16.A.4). Table 16.A.7 lists these unit operations.

16.A.5.5 Accident Driving Forces

This section identifies the driving forces that could potentially cause accidents at the representative facility. These driving forces and their locations are given in Table 16.A.8.

16.A.5.6 Containment and Shielding Systems

Shielding at the representative ISFSF is provided by lead or concrete walls or by water. Water-shielded facilities include the cask unloading pools and fuel storage basins. Lead-lined concrete shielding is provided for the ion exchange columns, filters, evaporators, radwaste holding tanks, and the waste solidification and storage areas. Confinement of radioactive materials is accomplished by adsorption and filtration of ventilation air and process off-gas, by the water in the storage basins and cask unloading areas, by the integrity of fuel rods, and by filtration and collection of the radionuclides from pool water. The shipping casks also provide one level of containment for spent fuel assemblies. Shipping casks are discussed in detail in Section 19 of this report and will not be discussed here.

(a) Under normal conditions, some space in at least one (and more likely, two) of the six storage pools would not be filled with spent fuel.

TABLE 16.A.7. Unit Operations Involved for Each Step in the
ISFSF Process Flow Schemes

<u>Process Step</u>	<u>Unit Operations</u>
<u>Spent Fuel Handling and Storage</u>	
Cask-vehicle preparation	Solids handling (lifting, conveying)
Cask-vehicle decontamination	Liquids pumping, chemical cleaning
Cask cooling, flush	Liquids pumping, heat exchange, chemical decontamination
Cask/fuel handling	Solids handling (lifting, conveying)
<u>Pool Water Treatment</u>	
Cooling	Liquid pumping, heat exchange, natural evaporation
Purification	Solid-liquid sorption (ion exchange) Solid-liquid separation (filtration)
Resin/Filter Regeneration	Liquid pumping, chemical reaction; slurry handling
<u>Waste and Effluent Management</u>	
Ventilation	Gas pumping Solid-gas separation (filtration) Solid sorption (I ₂ adsorption)
Radionuclide concentration	Solid-liquid separation (filtration) Solid-liquid sorption (ion exchange) Liquids distillation (evaporation) Liquids pumping, vapor condensation Solid-liquid separation (resin settling)
Solid-liquid solidification	Cementing
Drum handling	Packaging, conveyance
Solid materials size reduction	Shredding, compaction.

Shielding at the facility is designed to keep the average dose rate in normally occupied areas below 1 mR/hr. This means that about 4 m (13 ft) of water above the top of the fuel assemblies is required. Therefore, fuel storage basins are 9 m (30 ft) deep and the cask unloading pool is 15 m (50 ft) deep. The additional depth in the CUP is required so that fuel assemblies can be withdrawn from the cask and still maintain 4 m (13 ft) of water above the spent fuel. Fuel and basket handling cranes are equipped with positive electrical and mechanical stops to prevent fuel assemblies from being elevated above the safe shielding depth of water.

TABLE 16.A.8. Accident Driving Forces at the Reference ISFSF

Accident Driving Force	Locations	Characteristics
Temperatures	Radwaste evaporator	50°C-150°C
Pressures	Primary cooling system	1-10 atm.
	Secondary cooling system	1-10 atm.
	Compactor	>10 atm.
	Filter-deionizer system	>1-10 atm.
Chemical reactivity	Resin-regeneration Solutions (NaOH or HNO ₃)	High reactivity
	Chlorine storage	High reactivity
	Diesel fuel oil storage	Flammable
	Impurities in water	Corrosive
	Ion exchange column	Moderate reactivity
	Charcoal adsorber	Moderate reactivity
Potential Energy	Fuel handling area	Cask drop, fuel drop
	Fuel storage basin	Fuel basket drop
Nuclear Reactivity	Fuel storage basin	>500 g fissile
	Cask/cask unloading pool	>500 g fissile
	Filter-deionizer system	<500 g fissile
	Radwaste evaporator	<500 g fissile
	Waste storage area	<500 g fissile
	Ventilation system	<500 g fissile
Thermal Energy	Fuel storage basin	~12 MW ^(a)
	Cask/CUP	~74 kW ^(b)
	Ion exchange column	NA ^(c)
	Resin settling tank	NA
	Steam generation plant	NA

(a) Based on 3000 MgHM of spent fuel 3.5 yr after reactor discharge.

(b) Based on 10 PWR fuel assemblies 0.5 yr after reactor discharge.

(c) NA: Not available.

The ion exchange columns will accumulate radioactivity. Therefore, shielding is required for this component to reduce the dose rate to operators in adjacent normally occupied areas below 1 mR/hr. The design shielding thickness was not available, but at the estimated average concentration expected (about 4E-6 Ci/m³) the shielding requirements are relatively modest. This component is back-flushed to remove the radionuclides so that periodic contact or semi-remote maintenance can be performed with higher dose

rates for these short operating times. The amount of shielding required for liquid waste holding tanks and radwaste evaporators is not immediately available.

Wet wastes are immobilized in cement and dry solid wastes are shredded and compacted into drums. The reference wet waste immobilization facility is located in a hot cell and operated remotely. Solid waste shredding, compaction, and drumming systems are also operated remotely from behind shielding walls. The moderate shielding thicknesses in these areas are not available. Also not available is the shielding thickness of the walls enclosing the drummed waste storage area.

Radioactive material confinement is provided by storage basin and CUP water and by the building ventilation system. Water effectively contains radionuclides that are present in particulate or soluble form. These could potentially be released from the fuel assemblies and the cask. This radioactive material is continuously collected and concentrated for disposal as waste.

The building ventilation system provides confinement for volatile, semi-volatile, and airborne radionuclides. The ventilation system is designed for once-through flow of air and provides controlled in-leakage of air by the use of airlocks at the facility entrances. Flow of air is from areas with lesser potential for contamination to areas with higher potential for contamination. Normally, the storage building ventilation air is exhausted directly to the atmosphere without treatment. However, in the event that air does not meet release guidelines, a radiation monitor in the exhaust duct will trip a control damper that diverts the air flow through a charcoal bed adsorber and HEPA filters before it passes to the exhaust stack. The process off-gas exhaust system continuously filters particulates and adsorbs radioiodine into charcoal beds. Air is exhausted into this system from the cask ventilation operation, drum storage area, the waste immobilization and drumming facilities, and the water treatment area.

16.A.5.7 Degree of Hands-On Operation

The ISFSF is designed for contact operation and maintenance except for fuel handling activities. Operations and maintenance are primarily performed using crane-mounted tools and hand tools. Equipment in the radioactive waste treatment and pool water treatment areas are flushed periodically to allow for either contact or semi-remote maintenance. Failed equipment is flushed (if necessary), drained, wrapped in plastic, and removed offsite for repair or transferred to the waste packaging area. There, it is compacted, placed in drums or boxes, and stored to await offsite disposal.

16.A.5.8 Fire Protection Systems

No information on the fire protection system is contained in the primary reference (DOE 1979). An applicable fire detection and suppression system is described in Section 16.B.5.8 of this study.

16.A.6 ALTERNATIVE PROCESS SCHEMES FOR THE ISFSF

Storage of spent nuclear fuel in water basins is an existing and successful technology. Long-term storage considerations (up to 100 years) have prompted research into additional storage technologies. This section contains brief descriptions of the major alternative spent fuel storage techniques. Also described are some alternative processes that could be incorporated in the representative ISFSF.

Some of the alternative spent fuel storage techniques described in this section require that fuel elements are packaged or encapsulated in canisters prior to storage. Therefore, these facilities are assumed to be co-located with a spent fuel packaging facility although packaging operations could also occur at the reactor. The operations that occur in the packaging facility include the following: placing fuel assemblies in overpack canisters; welding a closure lid on top; testing the weld; decontaminating the canister; and transferring the canister to the storage facility (Rockwell Hanford Operations 1978). The packaging operations are performed in a remotely operated hot cell. A packaging facility is described in detail in Section 16.B of this report as a component of a solidified high-level waste storage system. The reader is referred to this section for more information.

Many of the alternative spent fuel storage techniques utilize natural or passive heat removal systems and are not suitable for fuel that has cooled for 5 years or less. Therefore, water basin storage must be used for spent fuel with relatively large heat generation rates. The spent fuel storage systems described in DOE (1979) use a modified version of an ISFSF for this purpose. The modified ISFSF consists of a cask receiving and unloading area and water-filled storage basins similar to the representative ISFSF. In addition, water-filled canals and associated handling equipment (cranes, transfer carts, etc.) are provided for underwater transfer of spent fuel to a packaging facility. The addition of the modified ISFSF to the spent fuel storage system allows for a receipt and storage of short-cooled fuel until it can be transferred to a packaging facility. Thus, the complete spent fuel storage system that could be utilized in the future would include a modified ISFSF, a spent fuel packaging facility, and a storage facility for fuel that has been cooled for about 6.5 years or longer, all located on the same site. One of the alternative storage concepts described in this section could be selected for the latter storage facility. Figure 16.A.9 shows an overall flow scheme for this conceptual spent fuel management system.

16.A.6.1 Alternative Processes for the Representative Facility

An alternative to storage of unpackaged spent fuel in the representative ISFSF consists of consolidation or disassemblage of fuel assemblies and storage of encapsulated bare fuel rods in a close-packed array. This alternative is being considered to increase the capacity of reactor storage pools. At this time, the details of the conditions and configurations needed for adequate circulation of cooling water or for control of nuclear criticality are not well defined for this storage configuration. This alternative involves essentially the same unit operations and hazards as the

FIGURE 16.A.9. Flow Scheme for a Conceptual Spent Fuel Packaging and Storage System (adapted from DOE 1979)

representative facility. However, the density of spent fuel materials per unit volume and the concentration of radioactivity-related driving forces are higher by about a factor of two in this alternative.

Another alternative storage configuration is being considered to increase the capacity of fuel storage pools at reactors. This alternative consists of racking the spent fuel in a double-tier configuration, or one where a second tier of fuel is placed on top of the first. This alternative requires special considerations for seismic loading, larger pool heat removal and purification systems and a greater depth of the pool. Most likely, a new pool design is required as this cannot be done with existing pools. This alternative involves essentially the same unit operations and accident driving forces as the representative case.

Alternative processes exist for the storage facility's waste management operations. First, instead of immobilization of wet wastes in cement there is a process that utilizes bitumen immobilization. This process uses well developed technology that is commercially available in the United States and elsewhere. The second alternative consists of the addition of an incinerator facility to further reduce the volume of combustible wastes. The latter system would make necessary the addition of substantially more off-gas treatment systems but would reduce the volume of waste shipped. Also, the potential for a fire or other accident caused by the incinerator should be considered. Additional information on these processes can be found in ERDA (1976) and DOE (1978).

16.A.6.2 Surface Storage Cask/Packaged Spent Fuel (DOE/ET-0028)

Massive, heavily shielded, concrete and metal storage casks are being considered for increasing utility spent fuel storage capacity at reactors. Storage casks are also being evaluated for use at potential Federal contingency storage facilities. These storage concepts are particularly attractive due to the fact that storage capacity can be expanded quickly and in small increments. Fuel is assumed to be encapsulated for storage in concrete casks and may or may not be packaged for storage in metal casks. Consolidated fuel may be stored in either type of cask.

For the concrete cask alternative, canisters of spent fuel are stored in a movable, reinforced concrete radiation shield stored above the ground. Heat is dissipated to the atmosphere by convective air flow through openings in the base of the cask. A typical storage cask unit is shown in Figure 16.A.10. Each cask accommodates four packaged PWR assemblies or nine packaged BWR assemblies that generate about 2.5 kW of heat. This facility is constructed in 2000 MgHM modules containing 1120 storage units. Loaded casks are placed vertically on concrete pads in essentially a 1-km² open field, if ten storage modules are constructed. More storage modules could be used if necessary.

This cask storage facility is assumed to be part of the spent fuel storage system shown in Figure 16.A.11. This storage system includes a modified water basin storage facility for storing short-cooled fuel until the heat generation

FIGURE 16.A.10. Surface Cask Storage Concept (DOE 1979)

rate is reduced to acceptable levels. Spent fuel is assumed to be packaged and placed in racks in a conceptual packaging facility connected to the modified ISFSF and then transferred to a remotely operated building where storage racks are placed in the cask. The cask is then moved through the cask transfer tunnel to an outside loading area. Here, a crane will lift the cask and place it on a special tractor-trailer transport vehicle that hauls it to the appropriate storage pad. A gantry crane at the storage area lifts the cask from the transporter and places it on a storage pad.

Inputs to the cask storage facility include up to 2000 MgHM/yr of packaged spent fuel aged at least 6.5 yr, stainless steel storage racks, and the storage casks. The maximum inventory of radioactive material in a particular cask is about $8E5$ Ci (assuming 4 PWR fuel assemblies aged 6.5 yr). Unit operations involved that are not listed in Table 16.A.7 include combustion of gasoline ($58 \text{ m}^3/\text{yr}$ for cask transporter), natural circulation of air through casks, natural convection of heat to air, and remote handling of spent fuel in air. Unit operations related to pool water treatment can be deleted. Combustible liquids (gasoline) is also an additional accident drying force not listed in Table 16.A.8. About 180 m^3 of solid wastes and 5 m^3 of wet wastes are expected to be generated annually, primarily from failed equipment and rare package failures. About 16 curies of radioactive material (mostly cesium) are contained in these wastes annually. Effluents are expected to contain negligible amounts of radioactive material (DOE 1979).

FIGURE 16.A.11. Flow Scheme for a Conceptual Spent Fuel Storage System Utilizing Cask Storage for Packaged Spent Fuel (adapted from DOE 1979)

16.A.6.3. Metal Storage Cask

This concept is essentially the same as the surface (concrete) storage cask concept except the casks are constructed of metal. Spent fuel is assumed to be aged at least 5 years after reactor discharge and is placed in the metal casks in packaged or unpackaged form. Cooling of the fuel is provided by conduction of decay heat through the cask structure where it is dissipated to the atmosphere. No air circulates through the cask cavity to cool the fuel. These casks have the same advantages over other concepts as the concrete casks. In addition, these casks could possibly be used as onsite or offsite shipping containers (may require a special overpack container for licensing for transportation).

Three casks are currently under consideration for use as storage vessels: 1) a cask designed, but not yet built, by Ridahalgh, Eggers and Associates, Inc. (the REA cask), 2) a cask based on modification of existing casks built by Transnuclear, GMBH, of Germany (TN-2100 cask), and 3) a cask based on modifications of casks manufactured by GNS, a German company [the Castor cask, (DOE 1981b)]. The important characteristics of these casks are shown in Table 16.A.9.

These storage casks do not require that intact fuel elements are packaged before being placed in the cask. Consolidated fuel may eventually be stored in these casks. For this study, spent fuel is assumed to be received in a spent fuel receiving building, similar to the ISFSF receiving area. In addition to receiving and unloading shipping casks, this building would contain a lag storage pool to permit continuity of operations. This facility would also contain provisions for loading spent fuel into the storage casks,

TABLE 16.A.9. Characteristics of Storage Casks(a)

<u>Designer/ Manufacturer</u>	<u>REA</u>	<u>Transnuclear</u>	<u>GNS</u>
Model	REA-2023	TN-2100	Castor-V(B and C)
Capacity			
PWR assemblies	24	21	20 - 24
BWR assemblies	52	37	50 - 52
Weight, loaded, Mg	80 - 89	100 - 109	91 - 114
Age of fuel, yrs	5	5 - 8	5
Thermal load, kW	30(b)	15	45 - 55

(a) Reference, DOE 1981b.

(b) Can be increased to 47 kW by addition at the storage site of special fins.

assumed to be performed under water (these operations may eventually be performed in remotely operated hot cells). A storage cask loading pool is assumed to be connected to the lag storage pool by a transfer canal that allows fuel assemblies to be moved to the storage cask loading pool. The fuel assemblies are lowered into a waiting cask. When the cask has received its loading of fuel assemblies, an inner shielding lid is secured and the cask is withdrawn from the pool, drained of pool water, and dried by passage of air through the cask cavity. Then the cask would be placed on a transporter, the outer lid welded in place, transported to the storage area, and placed either vertically or horizontally in its designated position. A general flow scheme for these operations is shown in Figure 16.A.12. Cask storage could also be utilized in the storage system shown in Figure 16.A.9.

The REA-2023 cask is chosen as the representative cask for this study. This cask consists of a double containment design with a welded final closure. The various components include a stainless steel outer skin, a lead gamma shield, and a water/glycol neutron shield. The primary containment vessel is also constructed of stainless steel. The design includes removable internal baskets with neutron adsorbing plates. Basket configurations are different for PWR and BWR assemblies but the storage envelope is the same.

FIGURE 16.A.12. Process Flow Scheme for a Conceptual Cask Storage Facility

The purpose of this is so only one cask body design can be manufactured for both types of fuel assemblies. The cask would be modified at the storage facility to accept PWR or BWR assemblies by placing the appropriate basket into the cask. Some of the key dimensions for the REA-2023 cask are shown in Figure 16.A.13.

The conceptual storage facility shown in Figure 16.A.13 utilizes all unit operations listed in Table 16.A.7, although there is much less water treatment. Additionally, the metal cask storage facility requires vapor or gas compression (cask cavity is filled with inert or nonoxidizing gas), solids melting (cask outer lid is welded in place) and natural conduction (through cask materials) and convection (to the atmosphere) of decay heat. Loaded casks may be stored in a warehouse structure for weather protection (DOE 1981b). It is assumed for this study, however, that these casks will be stored on concrete pads in an open field. Waste volumes generated by the storage operation (excluding packaging and modified ISFSF operations) are estimated to be approximately the same as those generated by concrete cask storage operations.

16.A.6.4 Dry Caisson Storage/Packaged Spent Fuel (DOE/ET-0028)

In this concept, several fuel assemblies in their individual canisters are mounted in a rack and placed in one of many buried caisson or dry wells, located in an open field. Each caisson is constructed of a steel pipe and is placed in a 7.7 m (25 ft) deep drilled hole. The annular space between the pipe and hole is filled with concrete. A concrete collar and shield plug are placed on top of the caisson and a concrete pad is poured to support the

Fuel Type	A(cm)	B(cm)	C(cm)	D(cm)	E(cm)	Neutron Shield	Gamma Shield
PWR	170	203	239	427	465	15.2	11.4
BWR	165	197	235	457	495	15.2	11.4

FIGURE 16.A.13. Dimensions of the REA-2023 Storage Cask

equipment used to transfer spent fuel into the caisson. The dry caisson storage concept utilizes the thermal conductivity of soil to dissipate the decay heat generated by spent fuel to the atmosphere. Dry caisson storage is suitable for packaged intact fuel elements or for consolidated fuel. Spent fuel is received, stored (in a modified ISFSF until at least 6.5 years old) and packaged in hot cells onsite. Fuel canisters are then placed in racks and remotely transferred to the fuel storage area transfer facility via an air-filled tunnel. There, racks are loaded in transfer casks, moved to an empty caisson by a gantry crane, lowered into the caisson, and a shielding plug placed on top. A flow scheme for charging a caisson is shown in Figure 16.A.14. The conceptual storage system (including modified ISFSF and packaging facility) is illustrated in Figure 16.A.15. The modified ISFSF is not required if all incoming spent fuel is at least 6.5 years after reactor discharge.

The conceptual dry caisson storage facility is constructed in 2000 MgHM capacity storage modules. Each module contains space for 1570 caisson units which can contain one storage rack (three PWR or six BWR fuel assemblies) each. Up to about 1.4 MgHM and 5E5 Ci of radioactivity are present in each caisson assuming the minimum age of fuel is 6.5 years. The caissons are located in a fenced, open field that encompasses 1.3 km² (320 acres) for a facility with ten modules. Rain is collected in a runoff diversion system that separates each storage module and routes it to a retention pond where it is allowed to evaporate. The same volumes and radionuclide contents of wastes generated by cask storage operations (see Section 16.A.6.2) are expected to be generated at this facility (DOE 1979). Unit operations performed at the dry caisson storage facility that are not listed in Table 16.A.7 include remote spent fuel handling in air, natural conduction of heat into concrete, steel and soil, and natural convection of heat from soil to air.

FIGURE 16.A.14. Conceptual Flow Scheme for Charging a Dry Caisson with Packaged and Racked Spent Fuel (DOE 1979)

FIGURE 16.A.15. Flow Scheme for a Conceptual Spent Fuel Storage System Utilizing Dry Caisson Storage of Package Spent Fuel (adapted from DOE 1979)

16.A.6.5 Tunnel Dry-Well Storage/Packaged Spent Fuel (DOE/NE-0019)

This concept is essentially the same as the dry caisson storage technique, except the dry wells are placed in a series of tunnels mined into the side of a mountain and not in an open field. Horizontal tunnels are excavated in the mountain and 7.5 m (25 ft) deep dry wells (similar to dry caissons) are drilled and lined into two rows near the walls on each side of the tunnel. Each hole, or dry-well, can accommodate one PWR or two BWR fuel assemblies that are packaged in one canister. Decay heat is transferred to the geologic media (instead of soil in the open field concept) where it is dissipated by natural conduction into the rock formation. No forced circulation of coolant is required for storage, but a forced ventilation unit is available to reduce tunnel temperatures for personnel access. This concept provides double containment by sealing fuel inside canisters (packaging) and then covering the charged dry-wells with shielding plugs. An illustration of a typical tunnel is shown in Figure 16.A.16. A flow scheme for the conceptual facility described in DOE (1981a) is shown in Figure 16.A.17.

A conceptual full-scale tunnel dry-well facility is expected to occupy about 16 km² (4000 acres). Little safety-related information is readily available on this concept because it is relatively new and has not been extensively studied. A conceptual demonstration facility would be capable of receiving, handling, packaging, and storing an inventory of 1333 spent fuel assemblies (50% PWR and 50% BWR for a total of about 430 MgHM) as well as 1000 canisters of high-level waste within a year (DOE 1981). The facility would be expandable to a full-scale facility capable of storing 48,000 MgHM over a period of 55 years (DOE 1981a). Unit operations involved in the tunnel dry-well storage concept are no different than those listed for the open-field dry caisson facility with the addition of the excavation and stabilization of the tunnels. Information on the excavation technique was not available. Estimates of effluents, waste generation rates, and radionuclide contents have not been made, but should be similar to those of the open-field dry caisson concept per unit of spent fuel.

16.A.6.6 Tunnel Rack Storage/Packaged Spent Fuel (DOE/NE-0019)

This concept is also relatively new and also utilizes horizontal tunnels mined in the side of a mountain. This facility could be associated with spent fuel receiving and packaging facilities, although a modified ISFSF is not contained in the documented conceptual system (DOE 1981a). Spent fuel assemblies are encapsulated (one PWR or two BWR fuel assemblies per canister) and tightly grouped in racks that contain 50 canisters of fuel each. A rack is then remotely moved down a central tunnel and placed in an enclosed, prestressed concrete-lined tunnel module that runs perpendicular to the main tunnel. Each tunnel module is 170 m (550 ft) long and provides enough space for 40 storage racks. Decay heat is transferred to air circulating by natural convective air flow through the storage tunnels. Approximately 7 km² (1750 acres) of subsurface storage area would be required for a full-scale facility storing 48,000 MgHM of spent fuel a minimum of 6.5 years after reactor discharge. This concept is illustrated in Figure 16.A.18.

FIGURE 16.A.16. Illustration of a Typical Dry Well That May be Utilized for the Tunnel Dry Well Storage Facility or the Open-Field Dry Well Storage Facility (DOE 1981a)

The tunnel rack concept provides a single level of containment, the sealed canister. No radiation shielding is provided in the storage area so all operations must be done remotely. Tunnel excavation (method is not available) and dry, remote handling of spent fuel are additional unit operations not listed in Table 16.A.7. Little additional information was available on this concept. This facility could consist of a receiving and packaging building and the storage area as shown in Figure 16.A.19 or could include a modified ISFSF for storage of short-cooled fuel until it is suitable for packaging.

16.A.6.7 Air-Cooled Vault Storage/Packaged Spent Fuel (DOE/ET-0028)

This alternative is another of the dry storage techniques that are under consideration. This particular concept involves packaging spent fuel in canisters and storing them in heavily shielded, partially buried, concrete vaults. Cooling is provided by natural circulation of air (using the chimney effect) past spent fuel canisters (once-through airflow). This is a passive cooling system that requires no forced coolant circulation, as do water basin storage concepts. The facility stores the packaged fuel in cells that have a capacity of 2000 MgHM each. This concept can only be used for spent fuel aged at least 6.5 years after reactor discharge because shorter-cooled fuel could develop unacceptable storage temperatures. However, if forced ventilation (fans) is used, shorter-cooled packaged fuel or bare assemblies can be stored

FIGURE 16.A.17. Conceptual Flow Scheme for a Tunnel Dry Well Storage Facility for Packaged Spent Fuel (adapted from DOE 1981a)

FIGURE 16.A.18. Tunnel-Rack Storage Facility (DOE 1981a)

FIGURE 16.A.19. Flow Scheme for Handling and Storage Operations at the Tunnel-Rack Storage Facility (adapted from DOE 1981a)

safely, but this feature would negate one important safety advantage of dry storage, i.e., a passive cooling system. This passive cooling, dry storage concept is illustrated in Figure 16.A.20. The conceptual storage system is shown in Figure 16.A.21.

The conceptual air-cooled vault storage facility can be built in a modular arrangement of storage cells containing 2000 MgHM each. This facility would be used for the packaged fuel storage facility in Figure 16.A.9. The

FIGURE 16.A.20. Flow Scheme for the Air-cooled Vault Storage Concept (DOE 1979)

packaging facility feeds the storage facility at about 2000 MgHM/yr (500 MgHM from modified ISFSF and 1500 MgHM packaged upon arrival from reactors) through an underground tunnel on a remotely operated, track-mounted cart. A shielded, remotely operated canister-charging crane loads fuel into the storage vault. A maximum of about $1\text{E}9$ Ci of radioactivity is present in each cell. Each cell is designed to remove 2.7 MW of heat. Estimates of annual waste volumes are the same for this spent fuel storage facility as for the cask storage facilities. Additional unit operations not listed in Table 16.A.8 include natural circulation of coolant air, remote handling of spent fuel canisters in air, and natural convection of heat to air. A ten-module storage field occupies about 0.05 km^2 (11 acres) of land in addition to the modified ISFSF and spent fuel receiving and packaging facility.

16.A.6.8 Water Basin Storage/Packaged Spent Fuel (DOE/ET-0028)

This concept is primarily the same as storage of unpackaged fuel assemblies. The technology is the same, i.e., storage of spent fuel in water-filled, below-ground basins with provisions made to remove decay heat and impurities from basin water. The main difference is that the facility is co-located with a spent fuel receiving and packaging facility. Spent fuel packaging operations could alternatively occur at the reactor or at a separate

FIGURE 16.A.21. Conceptual Spent Fuel Storage System Utilizing Air-Cooled Vault Storage Concept

facility located on a different site. The water basin storage facility could be part of the spent fuel storage system shown in Figure 16.A.9 or it could be a stand-alone facility.

Packaging the fuel results in a reduced demand on pool water purification systems primarily due to an additional barrier to radionuclide release. As a result, waste volumes produced by the packaged spent fuel storage basin are estimated to be a factor of four less than the amount of wastes generated at the representative ISFSF (DOE 1979). Relatively small amounts of radioactivity are present in these wastes; most likely less than 20 Ci/yr are disposed of as wastes.

Radioactivity in the exhaust air from the building ventilation system during normal operations is assumed to be negligible (DOE 1979). No new unit operations are involved beyond those listed in Table 16.A.7. An additional accident driving force for Table 16.A.8 could be thermal or mechanical degradation of the weld zones on spent fuel canisters due to heat, radiation, fatigue, or any of several corrosion mechanisms. Failure of the weld could release significant quantities of radionuclides to the pool, some of which are

volatile and would be released into the building ventilation system and the remainder would be contained in the pool water where it would have to be cleaned up rapidly.

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16.B SOLIDIFIED HIGH-LEVEL WASTE STORAGE

High-level liquid waste (HLLW) is the raffinate from the first cycle solvent extraction and dissolver sludge from a spent fuel reprocessing plant. This aqueous solution contains essentially all of the non-volatile fission products from the spent fuel. Therefore, HLLW is highly radioactive and generates significant large quantities of heat. Solidified high-level waste (SHLW) is generated by conversion of HLLW to a stable, more durable, solid form, assumed to be borosilicate glass. The SHLW glass is packaged in sealed metal canisters at the HLLW solidification facility (assumed to be co-located with the reprocessing plant) and shipped to the SHLW storage facility.

Open-field dry caisson or dry well storage is the representative concept for the SHLW storage facility. This concept uses shallow holes (7.4 m deep) drilled into the earth and lined with concrete and a sealed carbon steel encasement vessel to provide storage. Heat from the SHLW canisters is transferred to the geologic media and released to the atmosphere from the surface of the field. Although this concept has not been used to store SHLW canisters, it is similar to that being used at the Idaho National Engineering Laboratory (DOE 1979) to store spent fuel from the Peach Bottom high-temperature gas-cooled reactor (HTGR) and is a direct application of existing technology (NRC 1978). The representative facility described in this section is primarily a modification and expansion of the conceptual demonstration spent fuel and SHLW storage facility described in Monitored Retrievable Storage Demonstration Facility and Dry Well Storage Field Conceptual Design Study, RHO-BWI-C-118 (RHO 1981). Where possible, conservative assumptions are made where information must be converted from spent fuel storage to SHLW storage. These assumptions are noted where they occur.

Design bases for the representative facility include the following:

- The dry well storage facility (DWSF) is a stand-alone plant. All needed requirements and services for facility operation are provided as part of the facility.
- The facility has the capability to receive, wet unload, and store canistered SHLW resulting from reprocessing 1500 MgHM of spent fuel per year (600 canisters/yr).
- Fifty percent of the SHLW is received in rail casks and the remaining 50% is received in truck casks (RHO 1981 assumes 100% truck shipments)
- SHLW is received a minimum of ten years after the original spent fuel was discharged from the reactor.
- A 50 canister lag storage pool is provided to permit continuity of receiving, packaging, and dry well emplacement operations.

- SHLW canisters measure 30 cm (12 in.)₃ in diameter and are 3.1 m (10 ft) long and contain approximately 0.18 m³ (6.3 ft³) of SHLW that generates about 2.3 kW of heat.
- Canisters are encapsulated in an overpack container inside remotely operated hot cells in the Receiving and Packaging building prior to storage.
- SHLW canisters are stored in 7.4 m (24 ft) deep dry-wells that are lined with carbon steel. Concrete is poured around the carbon steel liner. A closure plate is seal-welded onto the encasement.
- Dry wells are constructed in 1000-canister-capacity storage modules located in an open field.

16.B.1 SUMMARY

The representative DWSF is a stand-alone plant capable of encapsulating and storing canisters of SHLW. The facility is sized to receive, unload, encapsulate, and emplace 600 canisters/yr. Each canister contains the HLW from reprocessing 2.5 MgHM of spent fuel so the DWSF has the capability of handling the annual volume of SHLW generated by a 1500 MgHM/yr capacity fuel reprocessing plant. Assuming the average ratio of PWR:BWR commercial power plants is 2:1, the representative DWSF is sized to handle the SHLW generated by about 45 GW-yr of annual nuclear electrical capacity. The incoming SHLW contains the amount of fission products that would be present in ten-year-old spent fuel. Each canister of SHLW contains approximately 72 kg (160 lbs) of metallic fission products incorporated in a borosilicate glass matrix. Canisters are 30.5 cm (1.0 ft) in diameter and 305 cm (10 ft) long.

Operations at the representative dry well storage facility occur primarily in the Receiving and Packaging building and the dry well storage field. The Receiving and Packaging building contains equipment and facilities needed to receive and off-load shipping casks from their transporter vehicles, wash and cool the casks, unload canisters underwater, temporarily store canisters underwater, and encapsulate canisters within an overpack container. Shipping cask handling and preparation for unloading are contact operations where personnel have direct contact with the cask. Canister unloading, lag storage and transfer operations are performed remotely. Encapsulation (or packaging) is performed in remotely-operated hot cells. Packaging operations include placing SHLW canisters in overpack canisters, welding caps on the overpack, heat treating the weld zone, ultrasonic non-destructive testing of the weld for discontinuities, leak rate testing, and external decontamination. The Receiving and Packaging building also contains provisions for loading the overpacked (or packaged) canisters into a shielded cask for remote transfer to a storage location.

Dry wells for storage of packaged SHLW canisters are located in an open field adjacent to the Receiving and Packaging building. Dry wells consist of pre-drilled holes in the ground that are lined with a pre-fabricated, carbon steel emplacement vessel. The annular region between the encasement vessel and

the hole is filled with concrete. An overpacked SHLW canister is lowered into the encasement vessel and a sand plug is poured on top of the package for shielding purposes. A steel closure lid is welded on top of the encasement vessel to seal the package inside. Provisions are made for routine monitoring of the encasement vessel temperature, internal pressure, and internal atmosphere.

Effluents from the representative dry well storage facility include cooling tower water, Receiving and Packaging building ventilation air, and uncontaminated process water. Building ventilation air is continuously filtered to remove airborne particulate radionuclides with provisions made for diversion of exhaust air through charcoal bed adsorbers in the event that radioactive gases are present. A process off-gas charcoal adsorber is provided to treat cask off-gas and exhaust air from the special packaging cell hoods prior to release to the building ventilation exhaust system for particulate filtration. Less than 1 Ci/yr is estimated to be released from the Receiving and Packaging building exhaust stack.

The various liquid waste streams generated in the Receiving and Packaging building are treated to remove radioactive materials (filter-deionizer systems), the radionuclides are concentrated (wet waste evaporators), solidified in cement and packaged in 210-l (55-gal) drums. Wet waste streams result from cask/vehicle washdowns, cask flushes and cooling, unloading and lag storage pool purification, and various decontamination operations performed at the facility. Approximately 38 m³ (1350 ft³) of these wastes are produced annually that contain 2800 Ci of radioactive material.

The solid radioactive wastes generated by the dry well storage facility's operations comprise about 97% of the facility's annual waste volume production. However, solid wastes contain only about 2% of the radioactivity in all of the wastes. Solid wastes are shredded, compacted, and packaged in 210-l (55-gal) drums.

Accident driving forces of the DWSF are identified. These include the thermal output and radiation hazards associated with handling and storage of SHLW canisters, removal and concentration of fission products from water (cask unloading pool, lag storage pool, and transfer canals) and building ventilation air, and packaging radioactive wastes for offsite disposal. Other accident driving forces of particular importance include the high temperatures associated with closure welding and heat treating the canister lids on the overpack assemblies, high pressures required for compaction of solid wastes generated at the facility, and the chemical reactivities of certain operational chemicals (diesel fuel oil, chlorine gas, and NaOH and HNO₃ solutions). In addition, there is a significant amount of mechanical handling of shipping casks and SHLW canisters that could result in accidents due to dropping.

The representative dry well storage facility is located on the generic site described in Section 7.0. Cask receiving and canister encapsulation operations occur 8 hours/day, 5 days/week, 250 days/yr. Monitoring and surveillance of dry wells occurs 24 hours/day, 365 days/yr. About 75 operators, 15 maintenance personnel, and 10 radiation monitors are estimated to

be needed to operate the facility. The number of professional staff required was unavailable.

16.B.2 MAINLINE PROCESS DESCRIPTION

The primary functions of the DWSF are receipt, packaging, transfer, and storage of SHLW canisters. An overall flow scheme of normal operations is shown in Figure 16.B.1. Canisters of SHLW are shipped to the DWSF in large truck or rail shipping casks. The shipping cask and carrier (truck trailer or railcar) are inspected, washed, and moved into the cask receiving area inside the Receiving and Packaging building. Activities inside the Receiving and Packaging building include off-loading casks from carriers, underwater unloading of canisters from the cask, lag storage, packaging, and transfer of overpacked SHLW canisters to the storage field. Operations in the dry well storage field include dry-well preparation, package placement, and placement completion activities.

This section is divided into two subsections. The first describes receiving and packaging operations. The second subsection describes the transfer and storage (emplacement) of SHLW canisters in dry wells. Note that in this section, the term "canister" refers to the primary SHLW container and "overpack" refers to the secondary container. An "overpack assembly" or "package" refers to an overpack container that has been charged with an SHLW canister.

16.B.2.1 Receiving and Packaging

Receiving and packaging operations are provided for in the Receiving and Packaging building of the conceptual facility. This section describes the operations that occur in this building to prepare SHLW canisters for transfer to the storage area. Refer to Figure 16.B.1 for an operational flow scheme of these operations.

Receiving.

Cask receiving consists of preparation, loading/off-loading of casks from their transport vehicle (carrier) and unloading under water of SHLW canisters from casks. These operations are essentially the same as the cask and fuel handling operations that were previously described for the representative spent fuel storage facility in Section 16.A. Provisions are made for radiation monitoring of the casks, inspection, washing, removing and storing cask peripheral equipment, off-loading the cask from the carrier vehicle, decontamination of cask and carrier, wash and cooldown of the casks, and underwater unloading of canisters. One additional operation not described in Section 16. that occurs at the SHLW storage facility helps to isolate a leaking canister. A suspected leaker, identified by cask coolant sampling is placed in a sipping station in the unloading pool after they are withdrawn from the cask. This component tests the water volume around the suspected leaking canisters for a rate of change in the concentration of fission products. A failed canister is placed in special failed-canister containers that have a bolted, gasketed closure.

FIGURE 16.B.1. Process Flow Scheme of Normal Operations at the Dry Well Storage Facility (adapted from RHO 1981)

The means of transferring canisters out of the unloading pools is not the same as that described for the representative ISFSF. Transfer at the SHLW Receiving and Packaging facility is accomplished by placing SHLW canisters into a basket that is then placed in a transfer buggy that travels on rails located below the water surface of the pool/canal. The transfer buggy capacity is three baskets containing one canister each. The rigid, telescoping mast on each of the fuel handling gantry cranes in the receiving area are used to push the buggy under a wall separating the receiving area/lag storage area. Here they are within reach of the lag storage area basket handling crane for subsequent movement.

Lag Storage.

The lag storage pool provides for temporary storage of incoming canistered SHLW to permit continuity of operations and to allow for possible interruptions in packaging operations. Approximately one month of surge storage (space for 50 canisters) is provided by this pool (RHO 1981). The storage area basket handling crane is used to move the basket buggy down a water-filled canal to the storage pool and places the baskets in storage racks attached to the floor of the pool. This crane is also used to move SHLW canister/basket configurations to the weld and test cells in the packaging area. Storage capacity for ten additional SHLW canisters is provided in alcoves of the transfer canals.

Provisions are made for cooling and purification of water in the lag storage pool and transfer canals in much the same way as was described in Section 16.A. However, the flow rates through heat exchangers and pool water filter-deionizer units are relatively low due to a decreased demand on these systems. A closed primary cooling system pumps pool and canal water to the external heat exchangers where decay heat is transferred to chilled water in the secondary cooling system. Decay heat is then transferred to the environment by cooling towers located outside the building. A closed primary cooling system is utilized to prevent the spread of radioactive materials outside the building. Purification is accomplished by routing the outlet water from the primary side of the heat exchanger through a filter-deionizer system before it is returned to the pool.

Packaging.

Canisters of SHLW are encapsulated in an overpack container filled with a stabilizing inert gas. Equipment is provided to perform these operations in a remotely operated weld and test cell. Leaking or failed canisters are either encapsulated in a special function cell or moved as quickly as possible to the weld and test cell for packaging. A flow scheme for the packaging operations is shown in Figure 16.B.2.

SHLW canisters are removed from lag storage by cranes and placed in a transfer canal buggy for transfer to the packaging cells. The buggy stops beneath either an air-filled transfer chamber (next to the weld and test cell) or a port beneath the special function cell. Baskets are unloaded from the buggy by cranes in the transfer chamber or in the special function cell.

FIGURE 16.B.2. Flow Scheme for SHLW Canister Encapsulation Operations in the Receiving and Packaging Building (adapted from RHO 1981)

Additional surge storage for loaded baskets is provided in alcoves near the chamber entrance.

Operations performed on normal, unfailed SHLW canisters are as follows. Since the SHLW canisters must be dry before packaging (RHO 1981), the transfer chambers have a canister drying station. There, baskets with canisters are dried by blower-circulated, heated air. The transfer chamber cranes are used to place the baskets in the drying stations. About four hours drying time per basket load is required (RHO 1981). Once dry, the canister is removed from the basket and lowered through a floor port by the transfer chamber crane into an empty overpack container in the weld and test cell below. The empty overpack container is held securely during filling operations by a canister transfer carrousel in the weld and test cell. These pieces of equipment are provided to receive incoming empty overpacks and swing them into an upright position under the canister filling port. Once the SHLW canister has been lowered into the overpack container, the carrousel swings the overpack to the cap placement position where operators place the cap using master-slave manipulators. The carrousel then swings and rotates the overpack assembly into a horizontal position for closure welding. All carrousel motions are preprogrammed and automatically controlled.

Welding operations begin with the welding positioner in a vertical position. The positioner then rotates the overpack assembly into a horizontal position where it is supported on turning rolls and positioned by adjustable pressure rolls. A swing arm device is provided to hold the cap in position until welding is completed. A submerged-arc welder is then lowered into position and the cap-joint welded while the overpack assembly is rotated at welding speed. The overhead crane in the cell is used to lift and transfer the welded overpack assembly to a post-weld heat treat station.

Provisions are made in the weld and test cell for two methods of handling the overpack assemblies. The first and most rapid method consists of using transfer jib hoists to provide semi-automatic lifting and transfer (manual activation of preprogrammed lifting and transfer operations) of overpack assemblies between stations. The second method is via an overhead bridge crane that is capable of making all material moves, although it is slower. The crane is available at all times to assist or override the transfer jib hoist system.

After the caps are welded, overpack assemblies are transferred to a post-weld heat treatment station. An electric induction heat treat system is used to heat only the cap joint weld area to a maximum temperature of about 650°C (1200°F). Heating and cooling rates are automatically controlled to prevent formation of residual stresses in the heated zone. Total cycle time is approximately 2.5 hours. Control is again semi-automatic with manual activation and supervision and automatic heating and cooling.

The next step in the process flow is non-destructive testing (NDT) of the weld area. The overpack assembly is transferred from the heat treatment station to the NDT station and positioned in a workhead that contains ultrasonic transducer equipment. The ultrasonic equipment is rotated in an orbital path around the cap joint to search for discontinuities in the weld.

Overpack assemblies with an unacceptable weld are removed by the overhead crane to an overhead transfer corridor that leads to the special function cell for repackaging. Acceptably welded overpacks are transferred to the inert gas pressurizing station.

Each overpack cap is fitted with a self-sealing charging valve for inert gas filling. This station is capable of positioning the overpack assembly so that a self-sealing coupling probe can descend, couple to, and open the charging valve. The gas-handling system (located in the service gallery and controlled from the operating gallery) then evacuates the overpack to a pressure of about 2.0 kPa (0.02 atm) and backfills the overpack to about 200 kPa (2.0 atm). The coupling probe is retracted and the self-sealing charging valve closes to retain the gas charge. A mechanical-sealing plug is placed in the charging valve opening and automatically seal-welded using the metal-arc inert gas (MIG) welding process. The inert gas charge is useful for conducting decay heat away from the SHLW canister (it has better heat transfer properties than air) and is necessary for leak rate testing.

The leak test stations are equipped with tank-type enclosures that will receive the overpack assembly. A mass spectrometer gas leak detector is provided in the enclosures. This system evacuates the atmosphere in the enclosure and measures any inert gas leakage for comparison with acceptance standards. An overpack assembly that is leaking at an unacceptable rate is removed to the special function cell for repackaging. Acceptable overpack assemblies are transferred to the radiation survey and swipe sampling station.

The radiation survey and swipe sampling station is equipped with an overpack holder and height adjuster. Here, master-slave manipulators are used to obtain surface swipe samples and dispatch them to the health physics laboratory for counting to determine the canister's external contamination levels. A radiation survey of the overpack assembly is also performed to detect excessively radioactive packages. Overpack assemblies that pass the radiation and swipe tests are transferred by the overhead crane through a port in the cell floor into a transfer buggy in the loadout tunnel. Packages that are contaminated to unacceptable levels are transferred to the decontamination and drying station.

In the decontamination and drying station, contaminated packages are decontaminated, dried, and returned to the swipe station for retesting. This station is provided with a tank with a remotely closable cover and internal rack. The tank is equipped with transducers to provide ultrasonic agitation of decontamination solutions to aid cleaning. Provisions are made for recirculating cleaning and rinse solutions that are held in tanks below the floor of the weld and test cells. The drying tank is equipped with an air drying circulation system similar to those in the transfer chamber.

Operations on incoming failed SHLW canisters are occasionally performed in the special function cell. Additionally, overpacked canisters that fail the ultrasonic inspection and leak rate test are removed from the failed overpacks in this cell. The special function cell is a two-compartment cell; one compartment is used to remove canisters from defective overpack assemblies and

the other is for encapsulation of failed canisters or re-encapsulation of canisters removed from defective overpacks. An overhead transfer corridor is provided to allow the dry, remote transfer of canisters and/or overpack assemblies between the special function cell and the weld and test cell.

Incoming failed or leaking SHLW canisters are moved underwater in the transfer canal buggy from lag storage or unloading pools to a port beneath the special function cell. An in-cell overhead crane is used to lift the failed canister into the cell and place it in a dryer, similar to driers in the transfer chamber. This cell has the same provisions and components as the weld and test cells for placing SHLW canisters in overpacks and welding the caps. There are, however, no provisions for heat treatment of the weld joints, ultrasonic NDT, inert gas charging, or leak testing. The overhead transfer corridor and handling system is used for the transfer of packages to the weld and test cells so these operations can be performed.

A second objective of the special function cell is the removal of SHLW canisters from failed or defective overpacks (determined by NDT or leak testing in the weld and test cell). Packages are received by the transfer corridor handling system and placed in the canister cutoff station. There, remotely operated clamps hold the package upright in a portable, pipe-cutoff machine. Normally, the SHLW canister in an opened overpack will be removed by the in-cell crane and then removed to the weld and test cell for repackaging. However, they may also be transferred to the encapsulation compartment of the special function cell for this operation. This cell is also provided with shielded storage sleeves for lag storage of twelve bare SHLW canisters or overpacked canisters.

Transfer and handling operations in the special function cell are performed by the overhead bridge crane and master-slave manipulators. There is no transfer jib hoist system in this cell as there is in the weld and test cell. This is so the special function cell will have more operational flexibility to handle non-standard operations.

16.B.2.2 Transfer and Storage of SHLW Canisters

This section describes the process flow of overpacked SHLW canisters from the weld and test cell to emplacement and completion of emplacement in dry-wells. A process flow scheme for these operations is shown in Figure 16.B.3.

Transfer Operations.

Overpacked SHLW canisters that have passed the ultrasonic NDT, leak rate test, and swipe test are transferred to the dry well storage field. After these tests, the weld and test cell overhead crane is used to lower the overpack assembly through a port in the cell floor into a shielded cask on a transfer buggy beneath the port. The buggy is then remotely moved down an air-filled tunnel to the package loadout room. The tunnel also serves as an airlock that isolates the weld and test cell from the loadout room. The buggy arrives and is positioned under a port in the floor of the loadout area. The buggy is equipped with an elevator mechanism to raise the cask tightly against

Note: RETRIEVAL IS ESSENTIALLY
THE REVERSE OF
EMPLACEMENT OPERATIONS

FIGURE 16.B.3. Flow Scheme for Transfer and Storage Operations at the Dry Well Storage Facility (adapted from RHO 1981)

the overhead port. The port cover is power-operated for opening and closing. Positioned over the port is a transporter cask equipped with a lifting mechanism that lowers a grapple through the port, attaches remotely to the overpack assembly, and lifts it into the cask. Shielding is provided for the transfer operations by a radiation shielding sleeve located inside the transporter cask that is lowered to surround the port. After the package has been raised into the cask, the shield sleeve is retracted, the cask bottom gate is closed, and the transporter vehicle moves to the storage location.

Placement Operations.

Upon arrival at the pre-drilled and lined dry well, the transporter vehicle aligns the cask over the center of the dry well. Special positioning mechanisms are provided for the cask alignment operations. The transporter cask is then lowered and pneumatically sealed to the top of the dry well encasement. Before the package is lowered into the dry well, identification markings on the package are visually confirmed by closed circuit television cameras or periscope. Upon confirmation, the cask bottom gate is opened and the radiation shield is lowered into the dry well. The package hoist mechanism then lowers the overpack assembly onto a support ring welded to the pipe reducer section of the encasement. The grapple is retracted into the cask and sand discharged into the space above the package to fill the upper compartment of the dry well. As the sand fills the compartment, the shield is retracted into the cask. Upon completion of these tasks, the cask is lifted into transport position and the cask bottom gate closed. The transporter vehicle then returns the cask to the Receiving and Packaging building loadout area for another package.

Completion of Placement.

These operations consist primarily of preparation of the dry well for welding, welding the closure plate to the top of the dry well, and inspecting the closure. First, the cask positioning fixture that aids the alignment of the cask for dry well charging operations must be removed. The top of the encasement is then cleaned and the closure plate set into position for welding.

The closure plate is welded to the encasement by the submerged-arc welding method. Two welding units mounted on self-powered vehicles are used to complete the closure weld. The first unit is used to complete an initial circumferential pass and is then moved to another location. The weld is next cleaned, brushed, and tested for cracks using liquid penetrant. The second welding unit then makes a slower second pass (and third if necessary) to complete the weld. Again, the completed weld is brushed, cleaned, and tested with liquid penetrant.

The final placement operations consist of inspecting and testing the dry well sample valve and installation of the weather-protective cover. The thermocouple assembly is inspected and tested for proper function and finally, the aboveground section of the dry well is painted for added corrosion resistance.

16.B.3 DESCRIPTION OF EFFLUENT PROCESSING AND WASTE MANAGEMENT SYSTEMS

This section contains descriptions of the effluent processing and waste management systems at the representative SHLW storage facility. Effluents and wastes differ in that wastes are disposed of with the intent of isolating them from man's environment. Conversely, effluents are released deliberately to the environment. Separate subsections are provided for each of these materials.

16.B.3.1 Effluent Processing

The representative facility releases both gaseous and liquid effluents. A flow scheme for the Receiving and Packaging building effluent control system is shown in Figure 16.B.4. Nonradioactive effluents include water from cooling towers, seepage and evaporation from water disposal ponds, and uncontaminated water from certain process steps. These materials receive little processing, if any. Water in the Receiving and Packaging building from cask and vehicle washdowns is collected by a sump system and routed to holding tanks where it is monitored for potential contamination. If not contaminated, this water is released to onsite water disposal ponds. If found to be contaminated to a level unacceptable for release, this water is routed through the waste treatment system to remove and concentrate the radionuclides. Effluents from the waste treatment system include the steam from radioactive waste evaporators (condensed, deionized, and re-used) and water from the radioactive filter-deionizer system (cleaned water is either recirculated to a deionized water storage tank for re-use or sent to the water disposal pond). Another nonradioactive effluent is rainwater that falls on the dry well storage field. A concrete or asphalt rainwater runoff diversion system is most likely needed, although it was not discussed in the documented facility description (RHO 1981).

Radioactive effluents include Receiving and Packaging building exhaust air and process off-gas exhaust air. The building ventilation system is divided into three subsystems: primary, secondary, and unrestricted. The primary and secondary subsystems continuously filter particulate material in a series of 30% and 90% ASHRAE^(a) pre-filters and 99.97% high-efficiency particulate air (HEPA) filters. The primary subsystem also contains an activated charcoal bed in series with and behind the particulate filters to adsorb radioactive gases. Normally, a diversion damper bypasses the activated charcoal bed. However, in the event that radioactive gases are detected in the exhaust air, the damper diverts the air flow through the charcoal bed. The secondary subsystem contains no charcoal bed.

The primary subsystem receives exhaust air from the waste treatment area, the special function cell, the weld and test cell, the transfer chamber, and the warm maintenance area (for decontamination or repair of equipment that has been removed from hot cells). These areas are the most likely to contain airborne radioactive materials. Two independent exhaust filter banks are provided for the primary subsystem to facilitate filter replacement in one bank while the other bank is on line.

(a) ASHRAE = American Society of Heating, Refrigeration, and Air Conditioning Engineers

FIGURE 16.B.4. Flow Scheme for the Receiving and Packaging Building's Ventilation Exhaust Treatment System (adapted from RHO 1978 and RHO 1981)

Process off-gas exhaust air is routed through a pre-filter and activated charcoal adsorption bed before it is discharged to the secondary subsystem for filtration and release. This subsystem receives exhaust air from cask ventilation operations, and from special exhaust hoods in the lag storage area, special function cell, and weld and test cells. The pre-filter/charcoal adsorber is located as close as possible to the source of radioactive material to minimize contamination of the secondary subsystem.

Exhaust air from the shipping cask receiving area, the cask/vehicle preparation area, the lag storage area, hot cell transfer corridors and operating galleries, the pump room, and health physics and counting laboratories is treated by the secondary subsystem's exhaust filtration system. These areas have a lower potential for content of airborne contamination. Two independent filter banks are also provided for this subsystem.

The unrestricted subsystem exhausts air directly to the atmosphere without treatment. This subsystem serves areas where no contamination is expected. Unrestricted areas include mechanical and electric equipment rooms, personnel support facilities (washrooms, change rooms, and offices) and the new overpack canister receiving and storage area.

16.B.3.2 Waste Management

Wastes from the dry well storage facility are primarily generated in the Receiving and Packaging building. These wastes consist of slurries, solutions and solids that are treated and packaged for shipment offsite. Figure 16.B.5 shows the operations that generate radioactive wastes in the Receiving and Packaging building. The systems that treat and package these wastes are also shown in Figure 16.B.5 and will be described in this section.

Liquids.

The various liquid waste streams at the representative facility are treated to remove and concentrate the radionuclides. The resulting slurries and sludges are then immobilized in cement and placed in 210-l (55-gal) drums and stored onsite to await shipment offsite for disposal.

The Receiving and Packaging building sump system collects water from the cask and vehicle wash and cask external decontamination operations. This water is routed to a storage tank where it is monitored for radiation. Contaminated water is transferred for treatment to a primary filter-deionizer system for radionuclide removal. The recovered demineralized water is sent to a storage tank for re-use.

Water in the cask unloading pools is continuously filtered and deionized to maintain the required quality of water. Each unloading pool has its own filter-deionizer system. Recovered, deionized water is recirculated to the unloading pools. Provisions are also made to process the water trapped between the canal gates.

FIGURE 16.B.5. Flow Scheme for Waste Management Operations in the Receiving and Packaging Building (adapted from RHO.1978)

Lag storage pool and transfer canal water is processed through a separate filter-deionizer system. The recovered, purified water is recirculated to the pool and canal. Also included is a leak collection system beneath the lag storage pool and transfer canal that collects leakage and pumps it back to the pool. A similar system is located beneath the cask unloading pools.

A central resin regeneration system is provided to receive spent resins from ion exchangers throughout the Receiving and Packaging building. Exhausted resins are pumped from their various locations to a spent resin holding tank. Spent resins are either regenerated in a separate tank and pumped back to the ion exchange column or, if they are not regenerable, they are pumped to a resin filtrate tank for dewatering. This tank separates the spent resins from the water and routes them to the waste solidification facility for immobilization in concrete. The separated water is pumped to a filter-deionizer holding tank for subsequent radionuclide removal.

Prefilters in the filter-deionizer system are periodically backflushed to remove the separated radionuclides. This solution is pumped to a waste holding tank. Other solutions that are pumped to this tank include hot cell and package decontamination sump solutions, cask decontamination solutions, and floor and equipment drains. This tank feeds the radioactive waste evaporator system that is used to concentrate these solutions by evaporating and removing the water. The resulting waste sludge is pumped to the waste immobilization area. Evaporated water is cooled and condensed in the evaporator off-gas system, deionized, and pumped to the recovered demineralized water holding tank.

Liquid waste solutions and spent resins are pumped to a waste immobilization system. There, they are placed in a concrete mixer, combined with concrete and water, poured into 210-L (55-gal) drums and cured. The drummed wastes are then moved to a shielded drum storage area to await shipment offsite as low-level wastes for burial.

Solids.

Radioactive solid wastes consist primarily of spent ventilation exhaust pre-filters and HEPA filters, failed equipment, maintenance materials, welding and cutting wastes, and general trash. These wastes are shredded and compacted to reduce volumes, where possible, and packaged in 210-L (55-gal) drums. Failed equipment is generally wrapped in plastic or sprayed to fix the residual surface contamination before removal from the Receiving and Packaging building for repair or shipment offsite for disposal.

16.B.4 PLANT LAYOUT

The SHLW storage facility consists primarily of the Receiving and Packaging building and the dry well storage field. This section contains descriptions of the site, facility, and major components of the representative dry well storage system.

16.B.4.1 Site Description

The representative SHLW storage system is assumed to be located on the generic site for a low-level waste burial ground described in Section 7.0. The facility occupies approximately 0.28 km² (70 acres) of the larger generic site if it is assumed to contain only one 1000-position dry well storage module, as is the case for the documented demonstration facility (RHO 1981). However, it appears that expansion of the storage field is feasible and would add about 0.2 km² (50 acres) for each storage module of 1000-position dry wells. The storage facility is surrounded by a double security fence that is posted with warning signs and is continuously and remotely monitored. In addition, a well-labeled perimeter fence encompasses the entire site to exclude the public. The plant is located so the exhaust stack is located about 0.3 km (0.19 miles) away from the nearest site boundary.

16.B.4.2 Facility Description

The representative facility consists primarily of a Receiving and Packaging building and the dry well storage field. A site arrangement drawing is shown in Figure 16.B.6 and shows these facilities and support, utility, and service areas located inside the secured area. It should be noted here that the documented facility (RHO 1981) that was used as a basis for this facility is limited to a demonstration capability and is assumed to receive both SHLW canisters and spent fuel assemblies in truck casks only. For this study, however, only SHLW canisters are received and a rail spur is assumed to be constructed to deliver rail casks to the facility. This necessitates a railroad guardhouse, an outdoor railcar inspection area (part of truck inspection area in RHO 1981) and rail sidings for empty and loaded railcar storage. These facilities and components are not shown on Figure 16.B.6 due to a lack of information.

Receiving and Packaging Building.

The Receiving and Packaging building is a multi-story, reinforced concrete structure. Those areas that contain SHLW canisters that are either directly exposed or in single containment are designed to Quality Level I. Level I areas are those that contain equipment or systems essential to maintaining safety. Level I areas include lag storage pools, transfer canals, weld and test cells, the special function cell, operating galleries, transfer chambers, the warm maintenance area, the loadout transfer tunnel, the control room, radioactive waste treatment areas, the exhaust filter fan room, and process equipment and pump rooms. These areas are designed to withstand the effects of design basis natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, and floods without impairing their capability to bring the plant to a safe shutdown.

Quality Level II areas are those which could either stop operators or are necessary for safe, normal operation. These areas include the preparation area, cask load and off-load area, the loadout area, laboratories, in-cell piping, and fire detection systems. As a minimum, these areas are designed to withstand the operating basis earthquake (defined as having a vibratory ground

FIGURE 16.B.6. Site Arrangement for the Representative Dry Well Storage Facility

motion at least one-half that of the design basis earthquake) and operating basis wind (fastest wind for the site with a 100-yr. mean recurrence velocity).

Quality Level III areas include heating and air conditioning systems, sewage systems, lighting systems, new canister receiving and storage areas, and personnel support facilities. These areas are related to the process or system operations that are not essential to a safe shutdown but they do provide a service to the facility. These areas are designed to withstand the operating basis wind and a seismic event equivalent to that defined in the Uniform Building Code seismic risk map for the site location.

The Receiving and Packaging building is divided into essentially six process areas; receiving, lag storage, packaging, loadout, waste treatment and support. The following paragraphs discuss these areas and the major components they contain. The floor plan for the Receiving and Packaging building is shown in Figure 16.B.7.

Incoming shipping casks on their transport vehicles are received initially at a perimeter guardhouse. There, they are unhitched from their over-the-road movers and hitched to a battery-powered car mover that moves casks/vehicles throughout the plant. The car mover then moves the railcar or truck to an outdoor surge storage location to await handling or directly to the Receiving and Packaging building. The cask/vehicle first stops in the cask preparation area of the Receiving and Packaging building where peripheral equipment is removed and stored. There are three cask preparation areas in the documented demonstration facility (RHO 1981). Each station serves as an airlock with controlled in-leakage of air to prevent the spread of contamination outside the building. The preparation areas are equipped with 9 Mg (10 ton) overhead bridge cranes and enough adjacent space to store the cask peripheral equipment, such as personnel shields and crash shields.

A 114-Mg (125-ton) overhead bridge crane with an 18-Mg (20-ton) auxiliary hoist is provided to unload and load shipping casks and to transfer casks between the cask washdown and coal pits, decontamination pits, and unloading pools. The crane is provided with lifting devices, such as bails, yokes, and slings, to upend and remove a cask from its carrier vehicle.

Cask cooldown and wash pits contain a cooling system complete with cooling hose connections and a cask washing system with fixed and movable platforms. A cask impact absorber pad and seismic tie-downs are included in each of the two pits. Cask washdown is accomplished by spraying high pressure deionized water on its surfaces and collecting the washwater in a sump. Drainage is directed to either a "clean" holding tank or "dirty" holding tank, depending upon the results of an earlier radiation survey performed on the cask. Cask cooling is accomplished by circulating steam and/or deionized water in the internal cask cavity to gradually reduce the temperature of the SHLW glass and the cask to avoid thermal shock damage upon immersion in the unloading pool. Cooling water and steam is monitored for temperature and radiation, condensed, cooled, and recirculated to the cask. Upon completion of cooldown, the waste water is pumped to a storage tank to await processing through a filter-deionizer system.

FIGURE 16.B.7. Floor Plan of the Receiving and Packaging Building (RH0 1981)

Two cask decontamination pits are provided in the Receiving and Packaging building receiving areas. These stations include provisions for a cask exterior decontamination, internal flushing, vacuum drying (for gas-cooled shipping casks only) and seal testing. These stations consist of a covered pit with an open-center turntable at the top for cask rotation. A watertight, vertically traversable operator's cab is located on one side of the pit. Decontamination solutions, steam, and deionized water nozzles are mounted on this cab. By raising and lowering the operator's cab and rotating the turntable, the entire cask surface is exposed to the spray nozzles. The cab is provided with openings that allow operators to reach the cask with long-handled brushes for local cleaning and for taking swipe samples. An access platform is also included that facilitates connection of hoses to the cask for internal flushing (with deionized water), vacuum drying (vacuum pump), and seal testing (compressed air). A sump system is provided to collect and recirculate decontamination solutions (steam, deionized water, alkaline, and acid solutions) or, if they are no longer effective, pump them to a holding tank to await treatment in a filter-deionizer. Cask interior flush water is recirculated, filtered, and monitored until cask cavity contamination levels are stable and then is pumped to a waste water holding tank.

Two water-filled cask unloading pools are provided in the receiving areas. These pools are constructed of reinforced concrete and lined with stainless steel. Each pool is equipped with a 4.5-Mg (5-ton) capacity gantry crane for removing SHLW canisters from the submerged casks and placing them into one-canister-capacity storage baskets. The crane is also equipped with a telescoping mast capable of loading storage baskets on a basket transfer buggy and pushing or pulling the buggy through the unloading pool transfer canal. The transfer buggy is a cart-type, unpowered vehicle that travels on rails beneath the transfer canal surface. Each unloading pool is equipped with a buggy capable of holding three baskets of SHLW canisters. Additional equipment located in the unloading pools is as follows:

- sipping stations (see Section 16.B.2.1)
- impact absorber: crushable material that protects pool structure and liner in the event a cask is dropped
- racks and accessory items: cask cover lifting fixtures, failed fuel containers, canister grapples and other remotely operated tools are stored in underwater racks in the unloading pools
- baskets for storage of canisters in lag storage pool: stainless steel tubes with capacity for one canister; inverted hooks on opposite sides for handling purposes; K_{eff} is less than or equal to 0.95.
- unloading pool cleanup system (filter-deionizer, no additional information is available).

The second process area in the Receiving and Packaging building is the lag storage area. Lag storage pools and transfer canals are constructed of stainless steel lined reinforced concrete. This area is equipped with two

cranes. One is a 4.5-Mg (5-ton) capacity gantry crane with a telescoping mast that is provided to move transfer buggies, remove storage baskets from buggies, transport the baskets through canals to the lag storage pool and place them in storage racks. The second crane is an 18-Mg (20-ton) capacity overhead crane with a 4.5-Mg (5-ton) capacity auxiliary hook that is used for general maintenance and service. This crane is also used as a backup unit in case of failure of the lag storage area basket handling gantry crane. Storage racks, attached to the pool floor, are provided to store baskets in a vertical position. Racks are constructed of stainless steel and designed to withstand seismic and other credible events without permitting the baskets to tip over. A total of fifty storage racks is provided in the lag storage pool, each with a capacity of one SHLW canister storage basket. Other equipment and facilities associated with the lag storage area include the following:

- canal gates: 1.2 m (4 ft) wide; they permit isolation of each canal and pool; they utilize double inflatable seals
- transfer canal buggy: remotely-operated, self-propelled, stainless steel buggy; used to transfer one loaded storage basket from the lag storage pool to hot cell areas
- changeable load grapples and special tools
- pool cooling and purification system: system includes 3.0 m³/min (800 gpm) pump, heat exchanger, and filter-deionizer; heat removal capacity is 140 kW; precoat filter cake is backwashed and replaced after treatment of every 3800 m³ (1,000,000 gal) of water; ion exchange resins regenerated every 61,000 m³ (16,000,000 gal) of water treated
- emergency cooling system: raw make-up water pumped from wells to replace pool evaporative losses; equipped with dual electric motor and diesel engine driven pump; capacity of 0.4 m³/min (100 gpm).

The packaging cells are the third process area in the Receiving and Packaging building. The documented demonstration facility (RHO 1981) contains one weld and test cell and one special function cell. Additional space is provided in the conceptual design for a second weld and test cell if the facility is expanded to a full-scale storage facility. These hot cells are constructed of reinforced concrete to Quality Level I standards. Access to the weld and test cell for incoming SHLW canisters is through an air-filled, reinforced concrete transfer chamber equipped with a 6.8-Mg (7.5-ton) capacity overhead crane that will be used to lift canister baskets from the transfer canal basket buggy into a canister dryer. A second, smaller 2.7-Mg (3-ton) capacity overhead crane is used to lift the SHLW canisters from the dryer and lower them through a port into an empty canister overpack container in the weld and test cell. The following process stations and equipment are located in the weld and test cell.

- two 2.7-Mg (3-ton) capacity overhead cranes; for transfer of canister packages between in-cell stations.

- empty canister transfer buggy system: transfer buggy remotely travels in a tunnel beneath hot cells; delivers empty overpack canister and cap to ports in floor of cells where they are lifted by jib hoist systems into the cells.
- SHLW canister overpack: constructed of carbon steel; overall dimensions are 35.5 cm (14 in.) outside diameter by 4.9 m (15 ft-10 in.) long, including lifting pintle; wall thickness is 0.95 cm (0.375 in.); includes an internal rack for support of the SHLW canister; it appears that a shorter overpack may be adequate for the representative SHLW canister used in this study.
- canister transfer carrousel: remotely operated; electrically driven; provisions for pneumatic clamping of overpack assemblies, raising and lowering and horizontal and vertical rotation of overpacks; preprogrammed and automatically controlled motions.
- welding equipment: submerged arc welder; equipped with turning rolls and adjustable pressure rolls for overpack assembly positioning and placement; shielding flux is gravity fed through a tube in the cell wall to a welder hopper; master-slave manipulator used to replace welder nozzle tips and to trim welding wire; cell wall pass-through provided to deliver replacement tips into cell.
- transfer jib hoist system: for semi-automatic lifting and transfer of overpack assemblies between in-cell process stations; six transfer jib hoists are provided.
- post-weld heat treat stations: electric induction heat treat systems; maximum temperature is 650°C (1200°F); insulated and water-cooled retractable workhead, power rating of induction coil is 20 kW maximum; two stations are provided in the cell with space reserved for two additional stations.
- NDT weld inspection system: one station; equipped with retractable workhead, overpack holder, height adjuster, and immersion tank; workhead contains ultrasonic transducer equipment that rotates in an orbital path around weld zone; sends signals to readout and strip chart recorder in operating gallery.
- inert gas pressurizing stations: two stations are provided (one on standby); equipped with workhead, canister holder, and height adjuster; workhead has coupling probe that indexes and opens overpack charging valve and provisions for inserting and seal welding a plug in the charging valve; gas-handling system in service gallery capable of evacuating overpack assembly and backfilling with an inert gas.
- leak test stations: two stations, each equipped with tank-type enclosures; mass spectrometer gas leak detection systems at each station.
- radiation survey and external swipe sampling station; equipped with canister holder, height adjuster, and master-slave manipulators.

- decontamination and drying station: cell provided with one decontamination station and one drying station; in-cell tank type enclosure with piping, recirculation pumps, and solution and rinse holding tanks in basement below cell; transducers in tank provide ultrasonic cleansing action; drying tank is equipped with air drying circulation system.
- shielded lag storage; maximum of 12 shielded sleeves provided with a capacity of one overpack assembly each.
- loadout port; opening in cell floor through which finished overpack assemblies can be lowered into the loadout transfer buggy.

The special function cell is a two compartment cell used to repair failed overpack assemblies or encapsulate SHLW canisters on an off-normal basis. Overpack assemblies with faulty or defective welds are moved into this cell from the weld and test cell via an overhead transfer corridor. The transfer corridor is equipped with a remotely operated 4.5-Mg (5-ton) overhead bridge crane used to lift or lower overpack assemblies through ports in the ceilings of the cells. The special function cell is provided with two 6.8-Mg (7.5-ton) capacity overhead cranes and a 2.7-Mg (3-ton) capacity overhead crane for transfer operations between in-cell process stations. Process stations and equipment located in the special function cell include the following:

- canister cutoff station: provided to cut through failed overpack lid welds; two stations provided; equipped with remotely operated clamps, a pipe cutoff machine, master-slave manipulators.
- failed-canister container open-and-unload system: power-operated clamps to hold incoming failed-canister containers from unloading pools so failed canister can be removed; also contains pneumatic wrench for removing bolts from top closure of failed-canister container.
- work-bench station; 1.8 m (6 ft) wide by 6.8 m (22 ft) long stainless steel bench within reach of manipulators for placing cell debris in waste containers for packaging and disposal.
- waste containers; stainless steel, open-top cylindrical containers 0.6 m (2 ft) in diameter by 5.5 m (18 ft) long with a lifting bail used for cell-generated wastes.
- decontamination station: same as weld and test cell decontamination station; also includes a small parts screen basket.
- canister transfer carrousel; sell weld and test cell.
- welding equipment; see weld and test cell.

The fourth process area in the Receiving and Packaging building is the loadout system. The loadout system for overpack assemblies that have passed the NDT and leak rate tests in the weld and test cells is as follows. Overpack

assemblies are lowered through the weld and test cell loadout port into a transfer buggy. The buggy is electrically powered and has a shielded cask designed to hold one packaged SHLW canister in a vertical position. The buggy is equipped with an elevator mechanism capable of raising the shielded cask tightly against the overhead ports during loading and unloading. The cask includes a remotely operated shielding gate which is closed except during loading and unloading. The buggy with a loaded shielding cask travels down a reinforced concrete, air-filled transfer tunnel to another port located beneath the loadout area.

The loadout area is a Quality Level II construction area (i.e., important to the process but not essential for safe shutdown or isolation). At this station, the overpacked SHLW canister is lifted through a port into a surface transporter/shield cask. The shielded cask is a vertical, cylindrical, bottom-loading cask equipped with a hoisting mechanism and automatic grapple device for vertical loading and unloading of overpacked canisters. The cask bottom is provided with a cask closure gate and a retractable radiation shield sleeve that interfaces with the dry well to permit shielded transfer of the overpack assemblies. Closed-circuit television cameras and telescopes are placed inside the cask for visual monitoring and control of package placement operations. The transporters that haul the casks are also specially designed to include both front- and rear-facing operator cabs. The transporter is equipped with vertical, horizontal, and rotational adjustment positioning mechanisms to align the cask with the loadout port and dry well centerlines. For package placement operations, the transporter is furnished with a sand placement system including a watertight, 0.85 m^3 (30 ft³) sand hopper with gate and charging spout. For package retrieval from dry wells, the transporter would be equipped with a vacuum system to remove the sand from the top of the packaged SHLW canister.

The waste treatment area is the fifth process area inside the Receiving and Packaging building. This area contains the ion exchange resin regeneration system, radioactive waste filter-deionizers, waste holding tanks, radioactive waste evaporators, and the waste solidification and drum storage stations. These facilities were described previously in Section 16.B.3. Additional information is as follows:

- The radioactive waste evaporator system is a package unit sized for a feed rate of $1.1 \text{ m}^3/\text{hr}$ to $4.5 \text{ m}^3/\text{hr}$ (5-20 gpm) and will concentrate the wastes to up to 50% solids.
- Off-gas water vapors from the evaporator will be condensed, cooled, deionized, and then collected in the deionized water storage tank (capacity of the tank is 75.7 m^3 (20,000 gal)).
- The waste collection tank that feeds the evaporator system has a capacity 59.4 m^3 (20,000 gal).
- Spent ion exchange resins are held in a 19 m^3 (5000 gal) spent resin holding tank prior to regeneration.

The final process areas in the Receiving and Packaging building are the support facilities. Support facilities are the major systems and equipment ancillary to the receiving, lag storage, packaging, loadout, and waste treatment operations. The support systems in the Receiving and Packaging building include:

- heating and air conditioning
- instrumentation and controls
- radiation monitoring
- cooling towers
- offsite electric power and its distribution system
- emergency diesel electricity generator and its distribution system
- warm maintenance area
- cold maintenance area
- administration building
- health physics and counting laboratories
- chemical makeup system and chlorine storage tanks
- hazardous material storage (caustic, acid, and fuel oil)
- deionized water makeup system
- plant and instrument air system
- gatehouses and security
- sanitary sewage system including water disposal ponds
- fire protection system.

Dry Well Storage Field.

This subsection describes the dry well storage field and associated equipment. Packaged SHLW canisters are moved from the loadout area of the Receiving and Packaging building inside a shielded cask and lowered into a dry well storage location. Each dry well consists of a pre-drilled hole lined with a cylindrical, carbon steel, package encasement vessel. The annular space between the encasement vessel and pre-drilled hole is filled with concrete to help in conduction of heat to the atmosphere. The vessel projects about 15.2 cm (6 in.) above and 7.4 m (24 ft) below grade level. The encasement vessel is fabricated of 45.7-cm (18-in.) and 76.2-cm (30-in.) outside diameter pipe joined by a standard pipe reducer, as shown in Figure 16.B.8. A cap is welded onto the bottom of the 45.7-cm (18-in.) diameter lower section of the encasement. A closure cap is welded on the top of the encasement after it has been charged with an overpacked SHLW canister. The encasement vessel designed, fabricated, and tested in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Division 2.

The encasement vessel has a dish-shaped steel support ring welded to the inside of the pipe reducer section. A similar dish-shaped ring is welded to the upper section of the overpack container when it is being fabricated. The overpack ring rests on top of the encasement ring to suspend the package in the dry well. After the package is emplaced, sand is poured into the dry well. The rings act as a seal to retain the sand in the upper compartment of the dry well. A minimum sand fill of 1.8 m (6 ft) above the package (requires approximately 0.85 m³ (30 ft³) of sand), and reduces the ground surface-level radiation dose in the storage field to 0.25 mR/hr. The sand material is

FIGURE 16.B.8. Cross Section and Details of an Overpacked SHLW Canister in a Dry Well (adapted from RHO 1981)

dry silica, commercial-grade sand that has been washed to remove any organic material and dried to a low moisture content. A stainless steel tube through the sand plug provides a connection to the lower compartment of the encasement vessel to permit sampling of the atmosphere in the dry well interior.

Each dry well encasement is equipped with provisions for direct measurement of the dry well temperature and pressure and for sampling of the interior dry well atmosphere to detect inert gas or krypton-85 leakage from the overpacked SHLW canister. Dry well temperature is measured periodically by a sheathed thermocouple within a protective tube on the exterior surface of the encasement vessel. The thermocouple sensing junction is positioned at the mid-height of the encasement vessel. The temperature indicator is mounted on a monitoring service truck, equipped with a recorder, digital temperature readout, and local alarm capability to indicate unusual temperatures.

Periodic pressure detection and interior sampling are facilitated by a sampling tube in the dry well cover plate. This tube is terminated by a needle valve closure followed by a quick connect, two-way, shutoff coupling. The coupling permits connection of a pressure transducer and sampling bottle but retains encasement integrity. A gasketed, bolted closure is provided to protect the valve and coupling against weather and mechanical damage from transporter or service vehicles. The monitoring service vehicle is equipped with a pressure indicator-recorder-alarm system.³ This truck also delivers the sampling bottles (approximately 150 cm³ (9.2 in.³) capacity to the dry wells. After the sampling bottles have been charged, they are delivered to a laboratory in the Receiving and Packaging building for analysis.

It was mentioned previously that the encasement vessel has a welded closure on top. Two special welding vehicles are provided for seal-welding the closure plate. Each vehicle is equipped with a welding fixture positioner, a semi-automatic submerged-arc welder, power supply, welding wire feeder, flux hopper, a weld testing kit, an air compressor, and controls. For retrieval operations, the vehicle is equipped with a mechanized arc-gouging fixture for weld removal. This fixture is mounted on the same positioning fixture as the welder. Weld testing kits include the necessary equipment and materials for testing the welds by the liquid dye-penetrant method.

The location of the dry well storage field was shown in Figure 16.B.6. The size of the storage field shown on this figure is based on storage of spent fuel in a 32 x 32 square array of dry wells and assumes a uniform 5.2 m (17 ft) center-to-center spacing. However, a similar array of the more thermally hot SHLW canisters will require a 13.5-m (44-ft) spacing. Therefore, a 1000-position dry well storage module occupies about 0.2 km² (50 acres). The storage areas are surrounded by a 1.2-m (4-ft) high chain link fence (to keep wildlife from activating the intruder systems), and a double exclusion fence equipped with intruder alarms and closed-circuit television surveillance systems.

The following facilities and services are provided in support of package storage or retrieval operations:

- forklift vehicles: 0.9-Mg (1-ton) capacity forklift for closure plate installation and removal; equipped with crane adapter for handling closure plates.
- temporary weather covers for uncharged dry wells.
- vehicle maintenance.
- sand storage station: Quality Level III, weatherproof shelter; 113.3-m³ (4000-ft³) sand storage bin; 0.8-m³ (30-ft³) batching system and transporter loading system; a dust collection system is also provided.
- transporter cask decontamination; performed in Receiving and Packaging building cask decontamination pit; one preparation area airlock is enlarged to accommodate the transporter vehicle.
- storage area gatehouse: full-time guard; controls access to storage field; equipped with in-plant telephone communication system.
- security building: same building as security headquarters for Receiving and Packaging building; contains security console, environmental console and closed circuit television monitors for control and operation of cameras monitoring fence lines. Additional security console located in control room of Receiving and Packaging Building.
- area illumination system.
- normal and standby electric power and their distribution systems.
- area and perimeter radiation monitors.

16.B.5 SAFETY-RELATED INFORMATION

This section contains important safety-related information that can be used to help identify the risks associated with the representative dry well storage facility. This information is based on the documented demonstration facility (RHO 1981) except where otherwise noted.

16.B.5.1 Operating and Maintenance Requirements

The representative dry well storage field is monitored and under surveillance 24 hours/day, 7 days/week, 365 days/year. Receiving, surge storage, encapsulation, and emplacement facilities are assumed to operate 8 hours/day, 5 days/week, 250 days/year (RHO 1981). Personnel requirements were not available in the reference documents (RHO 1978 and RHO 1981) but it is assumed that personnel requirements for this facility are about double those for the representative ISFSF (see Section 16.A). Therefore, about 75 operators, 15 maintenance craftsmen and 10 radiation monitors are required for the entire dry well storage facility. The number of professional staff required was not available.

16.B.5.2 Input/Output of Major Materials

This section contains the types and amounts of the major materials input to and output from the representative facility. The materials covered in this section include feeds, products, wastes, effluents, and nonradioactive input materials.

Feeds and Products.

The representative facility receives 600 canisters of SHLW per year or the amount of SHLW that is produced from reprocessing 1500 MgHM of spent fuel. Each in-coming canister is assumed to contain the quantity of fission products present in 2.5 MgHM of spent fuel aged ten years after discharge from a nuclear power reactor. The SHLW has a bulk density of 3000 kg/m^3 (187 lb/ft^3) and has a glass:waste oxide ratio of 3:1. Each canister of SHLW contains 72 kg (160 lbs) of elemental fission products that generate approximately 2.3 kW (7670 BTU/hr) of heat. The radionuclide content of each canister is shown in Table 16.B.1. This table was computed assuming the canister contains the fission product wastes from reprocessing fuel that is enriched to 3.2% ^{235}U and an average fuel burnup of 29,000 MWd/MgHM. In addition, fuel is assumed to be reprocessed 160 days after reactor discharge and the wastes are shipped to the dry well storage facility ten years after the spent fuel was originally discharged from the power reactor. For conservatism, 100% of the fission products that would have been present in the spent fuel is assumed to be present in the SHLW, except for noble gases and radioiodine. For these, 100% and 99% of the radionuclides, respectively, are assumed to be released to the fuel shear and dissolver off-gas systems at the fuel reprocessing plant (DOE 1979). The final assumption made was that chemical processing removes 99.5% of the U and Pu but none of the other actinides or transuranic radionuclides. At the receipt rate of 600 canisters/yr, a total of $4.1 \times 10^8 \text{ Ci/yr}$ of radioactive material is input to the representative facility.

Effluents.

Little information is available in the primary references (RHO 1978 and RHO 1981) on the amounts of the effluents that are expected to be released from this facility. However, the document DOE/ET-0028 (DOE 1979) contains this type of information for a similar facility. The facility in DOE (1979) is a modified packaging facility that receives spent fuel assemblies from a water basin storage building via underwater transfer canals. As a result, no shipping cask receiving and unloading or lag storage operations are performed within the packaging building, i.e., the spent fuel storage facility provides these services. For this report, the data contained in DOE (1979) per Mg of original fuel equivalent provides a rough, conservative estimate of the effluents released from the representative dry well storage facility. The following additional assumptions were made to facilitate calculations of the release quantities:

TABLE 16.B.1. Radionuclide Content of a Single SHLW Canister Used for This Study.^(a)

Radionuclide Category	Curies/Canister ^(b)
Noble Gases ^(c)	0
Iodine ^(d)	8.8E-4 ^(e)
SVFP ^(f)	2.05E+5
Tritium (³ H)	6.5E+5
AOFP ^(g)	4.6E+5
ULA ^(h)	2.8
TRU ⁽ⁱ⁾	2.2E+4
Total	6.9 x 10 ⁵

- (a) Radionuclide data tables used are from DOE/ET-0028, Vol 1 (DOE 1979)
- (b) Assumes SHLW contains 100% of the radionuclides from 10 yr old fuel, unless otherwise noted.
- (c) 100% of noble gases released to fuel reprocessing plant (FRP) off-gas systems
- (d) 99% of radioiodine released to FRP off-gas systems
- (e) 8.8E-4 reads 8.8×10^{-4}
- (f) SVFP: Semi-volatile fission products; combined contributions from radioisotopes of rubidium, ruthenium, antimony, tellurium and cesium.
- (g) AOFP: All other fission products.
- (h) ULA: Uranium plus lower actinides; assumed 99.5% of U was extracted at the FRP.
- (i) TRU: Transuranic radionuclides; assumes 99.5% of Pu was extracted at the FRP.

- Handling and packaging of SHLW canisters is assumed to release the same quantities of radioactive materials as packaging and handling spent fuel assemblies. This is conservative because SHLW is already inside one containment barrier when it arrives.
- The amount of ⁸⁵Kr and radioiodine contained in the SHLW is assumed to be negligible so none is available for release.
- The amount of water released from cooling towers is assumed to be scaled down from the release quantities at the representative ISFSF (Section 16.A.5.2) assuming they are linearly dependent upon their respective flow rates.
- The baseline for calculating release quantities from the Receiving and Packaging building is shown in DOE (1979, Table 5.7.35).

- Gaseous emissions from the dry well storage field are assumed to be negligible due to the number of sealed containment barriers (canister, overpack, and dry well encasement).
- Negligible amounts of radioactive materials are released from the dry well storage field.

Table 16.B.2 shows the results of these calculations.

TABLE 16.B.2. Effluents from the Dry Well Storage Facility for SHLW

Effluent	Description	Annual Quantity	Radionuclide Content ^(a)
Ventilation Air	Receiving and Packaging building exhaust air (excluding receiving area; this data was not available)	NA ^(b)	³ H - 0.8 Ci/yr. ^(c) ⁸⁵ Kr, I ₂ - negligible SVFP - 5E-3 Ci/yr. AOF - 3E-4 Ci/yr. ULA - negligible TRU - negligible
Cooling Tower Water	Evaporation (vapor)	2.6E+7 kg/yr ^(d)	Natural Amounts
	Drift (Aerosol)	1.3E+5 kg/yr ^(d)	Natural Amounts
	Blowdown (liquid from tower)	4.6E+6 kg/yr ^(d)	Natural Amounts
Water Disposal Ponds	Seepage (liquid) and evaporation (vapor)	NA	Natural Amounts

(a) Calculated by multiplying radioactive release factors in DOE (1978), Table 5.7.35) by the input activity per canister (Table 16.B.1) and 600 canisters/yr. Assumptions are outlined in the text.

(b) NA: Not Available

(c) Abbreviations are defined in Table 16.B.1.

(d) Calculations of these figures are facilitated by assumption outlined in the text.

Wastes.

As with effluents, waste quantities and radionuclide contents of the wastes generated at the representative facility were not available in the primary references. A similar procedure was used to calculate rough estimates for these wastes as was used for effluents. It was assumed that the waste volumes and radioactivity factors per Mg of original fuel equivalent for the spent fuel packaging facility described in DOE (1979) are conservative estimates of the waste volumes produced in the representative SHLW storage facility's packaging and loadout areas. These figures were added to

conservative estimates of the wastes generated by receiving and lag storage operations from Section 16.A of this report. Calculations of receiving and lag storage waste volumes were facilitated by the following assumptions:

- Calculations are based on the conservative assumption that receiving, storage, and packaging of SHLW canisters generates similar wastes as those operations for spent fuel.
- The volume for the wet wastes produced by lag storage pool purification is conservatively assumed to be 1/2 of the volume produced by ISFSF pool purification systems, although the flow rate is less than 1/2 of the ISFSF flow rate. Detergent solutions are assumed to be equal to those listed in Figure 16.A.3.
- The values listed in Table 16.A.4 were used to conservatively estimate the volumes of solid wastes generated by cask receiving and lag storage operations. The values listed in DOE (1979, Table 5.7.33) conservatively estimate the solid waste volumes from the packaging and loadout operations. Radionuclide concentrations are assumed to increase linearly with the increased waste volumes generated by the dry well storage facility.

Table 16.B.3 shows the estimated waste volumes and radionuclide content of wet and solid wastes generated at the representative dry well storage facility.

The wastes discussed here are solidified (if wet wastes), shredded and compacted (if the proper type of solid waste), packaged in 210-l (55-gal) drums and stored onsite to await disposal offsite. Assuming wet wastes are mixed with equal volumes of concrete for solidification purposes (RHO 1981), approximately 360 drums of solidified wet wastes are produced each year. The number of drums of solid waste produced annually is assumed to be scaled linearly up from the number of drums generated at the representative spent fuel storage facility (see Section 16.A.5). The multiplying factor is the ratio of the waste volumes reported in Table 16.B.3 to those reported in Table 16.A.4. This results in annual production rates for the dry well storage facility of about 3750 drums/yr for HEPA filters and combustible wastes and 460 drums/yr for failed equipment and non-combustible trash.

Non-Radioactive Input Materials

Other input requirements to the conceptual representative facility are listed below.

- NaOH for ion exchange resin regeneration and decontamination (most likely bought as 50% solution or 100% dry solid; annual requirement is not available)
- HNO₃ for ion exchange resin regeneration and decontamination (most likely bought as 60% solution, annual requirement is not available)

TABLE 16.B.3. Quantities and Radionuclide Content of Wastes Generated at the Dry Well Storage Facility as Input to the Waste Treatment Systems^(a)

Waste Stream	Form		Volume (m ³ /yr)	Principal Radionuclides (Ci/yr)		
	Chemical	Phase		SVFP ^(b)	AOFP ^(c)	Total
Wet Wastes	Water, polystyrene, NO ₂ SO ₄ , etc., (See Table 16.A.3)	S+L	38	2750	57	2807
Compactable and Combustible Wastes	Wood, glass, paper HEPA filters) etc. (See Table 16.A.4)	S	1168	25	0.4	25
Failed Equipment and Non-combustible trash	Metal, glass	S	133	2.0	0.03	2.0

(a) Rough estimates only. Calculations facilitated by assumptions outlined in text. Numbers contain excess significant figures.

(b) SVFP: Semi-volatile fission products. Combined contributions from rubidium, ruthenium, antimony, tellurium, and cesium.

(c) AOFP: All other fission products.

- Detergent for cask/vehicle cleaning (annual requirement is not available)
- Ion exchange resin beads (type and annual requirement is not available)
- SHLW canister overpacks and lids (600 per year are required)
- Chlorine for make-up water treatment (annual quantity is not available)
- Inert gas for back-filling overpack assemblies (annual requirement is not available)
- Sand for filling charged dry wells (850 m³/yr (30,000 ft³/yr) required).

16.B.5.3 Inventories of Major Hazardous Materials at Plant Locations Having Safety-Related Importance

This section discusses the inventories of hazardous materials at various locations on the dry well storage facility site. Table 16.B.4 lists these materials as well as their locations, chemical form, phase (solid, liquid, or gas), quantity, and radionuclide content. As can be seen from Table 16.B.4, the locations having the largest inventory of radioactive materials are the dry well storage field (1000 canisters of SHLW), the lag storage pool (50 canisters) and the shipping cask receiving area (24 canisters).

The information contained in Table 16.B.4 was calculated with the intent of showing the maximum inventory of radionuclides in a particular component during normal operations. Several key assumptions were made to facilitate these calculations. Many of the calculations involved multiplying the radionuclide content of a single canister of SHLW (see Table 16.B.1) by the maximum number of canisters a particular component can handle. For example, for the receiving area, a maximum of 27 canisters was assumed (three rail SHLW shipping casks, each hauling nine canisters). The data shown in Table 16.B.4 for this area was obtained by multiplying the radionuclide contents of a canister, shown in Table 16.B.1 by 27 canisters. This procedure was followed for several locations throughout the plant. The quantities of SHLW, both in terms of the number of canisters and the quantity of spent fuel that was reprocessed, are shown in Table 16.B.4.

Radionuclide concentrations in pool water were extracted from BNWL-2256 (Johnson 1977). These values represent typical radionuclide concentrations in spent fuel storage pools. This is considered to be conservative due to an additional containment barrier provided for SHLW (i.e., the canister) that will most likely release little or no radioactive material to the pool water. It was also assumed that the activation products originally present in the spent fuel assemblies are retained in the cladding hulls and fuel assembly hardware at the fuel reprocessing plant. As a result, their contributions to the radionuclide inventory of SHLW and pool water is not significant.

The maximum inventory of radioactive material in an ion-exchange column occurs just prior to regeneration. Regeneration occurs after the resins have treated approximately $6.1 \times 10^4 \text{ m}^3$ ($16 \times 10^6 \text{ gal}$) of pool water. To calculate the amount of radioactive material in the ion exchange column, it was assumed that 100% of the radionuclides in the $6.1 \times 10^4 \text{ m}^3$ of water that was treated is removed by the resins. Pool water radionuclide concentrations were extracted from BNWL-2256 (Johnson, 1977).

A rough estimate for the maximum total amount of radioactive material in a 1 m^3 (260 gal) waste evaporator was calculated as follows. It was assumed that the maximum inventory of radionuclides in a single 210-2 (55-gal) drum is 15 Ci (Black and King 1978). Each drum is assumed to contain 50% waste sludge and 50% concrete. Therefore, the waste sludge contains 15 Ci for each 100 of wet wastes or 150 Ci/m^3 , so a 1 m^3 evaporator, at the most, holds 150 Ci.

**TABLE 16.B.4. Inventories of Hazardous Materials
at various Plant Locations**

Location/Material	Form			Radionuclide Content (Ci) ^{(b)(c)}						
	Chemical	Phase	Quantity ^(a)	Iodine	SVFP	³ H	AOFP	ULA	TRU	Total
Receiving Area										
SHLW ^(d)	Glass ^(e)	S	60MgHM(24)	2.1E-2	5.0E+6	1.6E+4	1.1E+7	6.7E+1	5.3E+5	1.7E+7
Cask Unloading Pool										
SHLW	Glass	S	20MgHM(8)	7.0E-3	1.6E+6	5.2E+3	3.7E+6	2.2E+1	1.8E+5	5.5E+6
Pool Water	H ₂ O	L+S	760 m ³	8.0E-5	0.15	NEG ^(g)	NEG	NEG	NEG	0.15
Basket Buggy ^(h)	Glass	S	7.5MgHM(3)	2.6E-3	6.2E+5	2.0E+3	1.4E+6	8.4E+0	6.6E+4	2.1E+6
Lag Storage/Transfer										
SHLW ⁽ⁱ⁾	Glass	S	125MgHM(50)	4.4E-2	1.0E+7	3.3E+4	2.3E+7	1.4E+2	1.1E+6	3.4E+7
Pool Water	H ₂ O	L+S	1000 m ³	1.0E-4	2.0E-1	NEG	NEG	NEG	NEG	2.0E-1
Canal Buggy ^(j)	Glass	S	2.5MgHM(1)	8.8E-4	2.1E+5	6.5E+2	4.6E+5	2.8E+0	2.2E+4	6.9E+5
Alcove ^(k)	Glass	S	25MgHM(10)	8.8E-3	2.1E+6	6.5E+3	4.6E+6	2.8E+1	2.2E+5	6.9E+6
Transfer Chamber										
SHLW	Glass	S	5MgHM(2)	1.8E-3	4.1E+5	1.3E+3	9.2E+5	5.6E+0	4.4E+4	1.4E+6
Packaging Cells										
Component ^(l)	Glass	S	2.5MgHM(1)	8.8E-4	2.1E+5	6.5E+2	4.6E+5	2.8E+0	2.2E+4	6.9E+5
Storage	Glass	S	30MgHM(12)	1.1E-2	2.5E+6	7.8E+3	5.5E+6	3.4E+1	2.6E+5	3.3E+6
Loadout Tunnel/Cask										
SHLW	Glass	S	2.5MgHM(1)	8.8E-4	2.1E+5	6.5E+2	4.6E+5	2.8E+0	2.2E+4	6.9E+5
Emplacement Cask										
SHLW	Glass	S	2.5MgHM(1)	8.8E-4	2.1E+5	6.5E+2	4.6E+5	2.8E+0	2.2E+4	6.9E+5
Dry Well (Each)										
SHLW	Glass	S	2.5MgHM(1)	8.8E-4	2.1E+5	6.5E+2	4.6E+5	2.8E+0	2.2E+4	6.9E+5
Storage Field										
SHLW	Glass	S	2500MgHM(1000)	8.8E-1	2.1E+8	6.5E+5	4.6E+8	2.8E+3	2.2E+7	6.9E+8
Ion Exchangers										
Resin Beaks	Organic Polymer	S+L	2 m ³	6.1E-3	1.2E+1	NEG	NEG	NEG	NEG	1.2E+1
Evaporator										
Wet Wastes	H ₂ O+RM	S+L	1 m ³	NA -					- NA	1.5E+2
Drum Storage Area										
Packaged Wastes	Many	S	79.8 m ³	NA -					- NA	4.5E+2
Make-Up Water Treatment										
Chlorine	Cl ₂	G	NA		Natural Amounts					
K and P Building										
Resin - Regen-eration Solns. & Decon. Solns. ^(m)	5% NaOH	L	NA		Natural Amounts					
	5% HNO ₃	L	NA		Natural Amounts					
Diesel Fuel Oil Storage										
Diesel Fuel	Hydro-carbons	L	115 m ³		Natural Amounts					

(a) Certain locations have a capacity for a particular number of SHLW canisters. These numbers are shown in parentheses.

(b) Numbers contain excess significant figures for ease of calculations. 1.4E-2 reads 1.4x10⁻².

(c) Abbreviations are as follows: SVFP: semi-volatile fission products. Total contribution due to rubidium, ruthenium, antimony, tellurium, and cesium. AOFP: all other fission products. ³H - Tritium. ULA: uranium plus lower actinides. TRU: transuranic radionuclides.

(d) SHLW: solidified high-level waste.

(e) Glass refers to the high-level waste nuclides incorporated in a borosilicate glass matrix.

(f) Represents the amount of spent fuel reprocessed to yield a particular quantity of SHLW.

(g) NEG: negligible

(h) Refers to the basket transfer buggy; capacity - three canisters of SHLW.

(i) Capacity of the lag storage pool is 50 canisters of SHLW.

(j) Refers to the transfer canal buggy; capacity - one canister of SHLW.

(k) Capacity of the alcove storage location is ten canisters of SHLW.

(l) Capacity of each piece of operating equipment is one canister of SHLW.

(m) NaOH is most likely bought as a dry solid or as a 50% solution. HNO₃ is probably bought as 50% solution.

The drum storage area is sized to hold one month's production of waste drums. The only drums that contain a significant quantity of radioactive materials are the drums containing wet wastes. These drums are assumed to have a maximum inventory of 15 Ci each (Black and King 1978). The production rate of these drums is 360/yr or 30/month. Therefore, 30 drums, each with an inventory of 15 Ci, results in a maximum inventory at this location of 450 Ci.

16.B.5.4 Major Unit Operations Involved

The major unit operations involved in Receiving and Packaging building and dry well storage and process flow schemes are shown in Table 16.B.5.

16.B.5.5 Accident Driving Forces

Table 16.B.6 identifies the driving forces that could potentially cause accidents at the dry well storage facility.

16.B.5.6 Containment and Shielding Systems

Water, sand, and lead or concrete walls are provided at the representative dry well storage facility for shielding purposes. Water shielding is provided for removing SHLW canisters from casks, lag storage operations, and for any movements of SHLW prior to encapsulation (except when SHLW is contained in the casks). Water provides, in addition to shielding, confinement of radioactive material released from shipping casks and SHLW canisters. Radioactive materials released to pool water are further contained by their continuous removal from pool water by filters and ion exchangers. Concrete shielding is provided for the packaging cells, pool water and waste treatment areas, transfer chambers, the warm maintenance area, the loadout tunnel, and the loadout area. Radioactive materials in these areas are further confined by filtration and charcoal adsorption of airborne radionuclides in the primary exhaust treatment subsystem (see Section 16.B.3.1). Sand shielding is provided to cover overpacked SHLW canisters in dry wells to reduce the ground surface-level radiation dose in the storage field to 0.25 mR/hr. About 1.8 m (6 ft) of sand above the package (SHLW canister and overpack) is required.

Shielding in the Receiving and Packaging building is designed to keep the average dose rate in normally occupied areas below 1 mR/hr. This means that about 4 m (13 ft) of water above SHLW canisters is required. To prevent operators from raising canisters above the safe shielding depth, canister-handling cranes are equipped with positive electrical and mechanical stops.

Shielding requirements for the special function cell and weld and test cell are about 1.2 m (4 ft) of concrete treated with a nonporous, easily decontaminable finish (RHO 1978). The transfer chamber and overhead transfer corridor have similar requirements. More modest shielding is required for the loadout area (and tunnel), waste treatment room, and warm maintenance room. The lag storage process equipment room is essentially unshielded with the exception of ion exchange columns and filters. These components collect significant amounts of radioactive material and require modest shielding to

**TABLE 16.B.5. Unit Operations Involved for Each Step
in the Dry Well Storage Facility Flow Schemes**

<u>Process Step</u>	<u>Unit Operations</u>
<u>Cask Receiving and Handling</u>	
Cask/vehicle preparation	Solids handling (lifting, conveying)
Vehicle washdown	Liquids pumping
Cask cooling	Vapor (steam) compression, Liquids pumping, solid-liquid separation (filtration), Heat exchange, Vapor cooling (condensing)
Cask washdown	Liquids pumping, Solids handling (movable spray platforms)
Cask decontamination	Liquids pumping, Chemical decontamination, solids surface removal (brushing), solids handling (cask rotation mechanism and vertically traversable operator's cab)
Cask flushing	Liquids pumping, solid-liquid separation (filtration)
Cask handling	Solids handling (lifting, conveying)
<u>Cask Unloading Pool Operations</u>	
Cask/canister handling	Solids handling (lifting, conveying)
Water purification	Liquids pumping, Solid-liquid sorption (ion exchange) Solid-liquid separation (filtration)
<u>Lag Storage and Transfer</u>	
Lag Storage Pool	Solids handling (lifting, conveying), Solids storage
Pool cooling and purification	Liquids pumping, Heat exchange, Solid-liquid sorption (ion exchange), solid-liquid separation (filtration)
Resin/filter regeneration	Liquids pumping, Chemical cleaning, Slurry handling, Solid-liquid separation (settling), Solids handling (mixing)
<u>Weld and Test Cells</u>	
Canister drying	Distillation (evaporation), Gas compression
Overpack closure	Solid materials melting (welding)
Stress relief heat treatment	Solid materials heating (electric induction)
NDT weld inspection	Ultrasonics
Inert gas pressurizing	Gas compression, Solids melting (welding)
Leak rate test	Gas compression, Mass spectrometry
Decontamination	Liquids pumping, Chemical decontamination, Vapor (Steam) compression Liquids handling (ultrasonic agitation), (See above)
Drying	
Manipulators, cranes, jib hoist system	Solids handling (lifting, conveying)
<u>Special Function Cell</u>	
Canister drying, overpack closure, stress relief heat treatment, decontamination	(same as above under Weld and Test Cells)
Canister cut-off	Solid materials cutting
Manipulators, cranes	Solids handling (lifting, conveying)
<u>Loadout and Emplacement</u>	
Loadout	Solids handling (lifting, conveying)
Emplacement	Solids handling (lifting, conveying sand handling), Solids melting (welding)
<u>Waste and Effluent Management</u>	
Ventilation	Gas pumping, Solid-gas separation (filtration), Solid-gas sorption (I ₂ adsorber)
Radionuclide Concentration	Liquids pumping, Solid-liquid separation (filtration), solid-liquid sorption (ion exchange), Liquids distillation (evaporation), Vapor cooling (condensation), Liquids storage
Solid-liquid solidification	Solidification of non-high temperatures (cementing)
Drum handling	Solids handling (lifting, conveying)
Waste shredder and compactor	Solid materials size reduction

reduce the dose rate to operators in adjacent normally occupied areas below 1 mR/hr. The shielding thicknesses required are not available. Also not available are the shielding requirements for the waste holding tanks, the waste evaporators, waste compactor, waste solidification area, and the drummed waste storage area.

The Receiving and Packaging building ventilation system provides confinement for airborne, volatile, and semi-volatile radionuclides. The ventilation system is designed for once-through flow of air and provides high-efficiency filtration and charcoal adsorption of exhaust air before it is released to the atmosphere. Controlled in-leakage of air is provided by airlocks at the building entrances. Flow of air is directed from areas of lesser potential for contamination to areas of higher potential for contamination by maintaining progressively greater negative static air pressures in areas with higher potential for contamination. This minimizes back-diffusion of potentially contaminated air. A more detailed discussion of the ventilation systems was given in Section 16.B.3.1.

16.B.5.7 Degree of Hands-On Operation

The representative facility is designed for contact operation and maintenance except for SHLW canister handling, lag storage, encapsulation, and loadout activities. Operations and maintenance are primarily performed using crane-mounted tools, hand tools, master-slave manipulators, and crane-mounted manipulators. Shielded hide-away storage locations for SHLW canisters are provided within the weld and test cell and special function cell to allow personnel access for contact maintenance of in-cell equipment. In addition, the Receiving and Packaging building will be provided with four designated maintenance rooms: cold manipulator maintenance room; warm manipulator maintenance room; warm maintenance room; and a general maintenance shop. Nonradioactive and warm manipulator maintenance rooms will be located close to the operating gallery to minimize the handling systems required to transport the manipulators between the operating galley and the maintenance rooms. The warm maintenance room will be equipped for maintenance of contaminated equipment removed from packaging cells. The room is also equipped with provisions for decontamination and special packaging of equipment for shipment offsite. Maintenance and repair operations will include light welding and cutting, pipe fitting, sheet metal working, light machinery, and painting.

Equipment in the radioactive waste treatment area and the lag storage process equipment room is flushed periodically to allow for either contact or semi-remote maintenance. Failed equipment from these areas is flushed, drained, wrapped in plastic, and removed offsite for repair or packaged for offsite disposal.

16.B.5.8 Fire Protection System

The fire protection system at the documented facility (RHO 1981) includes a 950 m³ (250,000 gal) water storage tank and two 7.6 m³/min (2000 gal/min) fire pumps (one electric-motor driven and one diesel driven). The fire pumps discharge water into a distribution system that supplies outdoor, 1.9 m³/min

(500 gal/min) five hydrants and a wet pipe sprinkler system in the Receiving and Packaging building. Portable fire extinguishers are also located throughout the Receiving and Packaging building. Fire pumps are tested weekly for proper operation.

Sprinkler heads in confined areas automatically actuate at an air temperature of 140°C. Those subject to mechanical damage are protected with guards. Water drainage from the sprinkler systems is collected and drained to the waste collection tank for subsequent monitoring, processing, and disposal.

The fire alarm system is part of the plant communication system. Fire alarms can be activated by manual operation of fire alarm pull boxes located throughout the plant, by signals from heat-actuated detectors in confined areas, or by a discharge from the sprinkler system. When a fire alarm is activated, a signal is transmitted to a local fire station, a local alarm sounds to evacuate personnel, and lights flash on a control console to show the location of the fire.

16.B.6 ALTERNATIVE PROCESS SCHEMES FOR THE DRY WELL STORAGE FACILITY

The current experience with storing SHLW is as follows. At the DOE's Idaho Chemical Processing Plant, HLLW is calcined to a granular solid then stored in underground bins. SHLW packages generated at Pacific Northwest Laboratory (PNL) during the Waste Solidification Engineering Prototype (WSEP) program have stored both in air and water for waste verification and storage studies. France also has experience with storage of packaged SHLW in air in caissons in the floor of a hot cell. Currently U.S. attention for a SHLW storage method is focused on the sealed storage cask concept described previously in Section 16.A.6 of this study for storage of spent fuel. This concept is being considered for the longer-aged SHLW and for longer-term interim storage in case a geologic repository is not available in the reasonably near future.

The other storage concepts described in Section 16.A.6 of this study could be used for SHLW storage with reasonable extensions of existing technology. These concepts include water basin storage, air-cooled vault storage, tunnel dry-well storage, tunnel rack storage, and concrete shield cask storage of packaged SHLW canisters (SHLW canister within a sealed overpack container). The safety-related features of these alternative concepts were given in Section 16.A.6.

Alternative processes for treatment of wastes generated at the SHLW storage facility could be incorporated as well. These alternatives were discussed previously in Section 16.A.6.1 for the representative spent fuel storage facility. Alternative processes could include incineration of combustible wastes and immobilization of wet wastes in bitumen as opposed to concrete. These processes are described in detail in ERDA-76-43 (ERDA 1976).

An alternative SHLW handling technique that could be considered at the dry well storage facility is "dry" unloading of the SHLW canisters from the shipping casks as opposed to "wet" unloading. Dry unloading would be

accomplished by mating the shipping cask to a sealed port that leads to a remotely operated hot cell. Operations within the hot cell would include removing the inner closure head of the cask, removing SHLW canisters from the cask, and lag storage of canisters awaiting transfer to packaging cells. This system eliminates the cask cooldown operations and the water-filled lag storage pools and transfer canals and their associated water cooling and purification equipment. More information on a conceptual dry unloading system can be found in document ARH-2799 (Atlantic Richfield Hanford Co. 1973).

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16.C TRANSURANIC WASTE STORAGE

A hypothetical transuranic waste (TRUW) storage facility representative of current technology is described in this section. This facility is designed to handle the store TRUW for an interim period. The TRUW, with low-level radiation levels, is considered to be contact-handled (CH) waste because the surface dose rate of containers holding CH-TRUW is less than 200 mR/hr and thus can be handled by contact methods. The TRUW with higher radiation levels is considered to be remotely handled (RH) waste because the surface dose rate of containers holding RH-TRUW is greater than 200 mR/hr and in most cases must be handled by remote methods. The TRUW are produced in fuel reprocessing plants (FRP) and mixed-oxide fuel fabrication plants (MOX FFP).

The representative TRUW storage facility described in this section is primarily modeled after example facilities described in Technology for Commercial Radioactive Waste Management, DOE/ET-0028 (DOE 1979, Sections 5.2 and 5.3). This hypothetical representative facility consists of 20 modular CH waste storage areas and 20 RH modular waste storage buildings. The overall characteristics of the representative TRUW storage facility include the following:

- One facility made up of a 20 modular CH-TRUW storage area with each module capable of storing 10,000 210-l canisters or the equivalent as 210-l canisters (drums) and 3800-l canisters (metal boxes), and 20 modular RH-TRUW storage buildings with each module capable of storing the equivalent volume of 20,000 210-l canisters or the equivalent as 210-l canisters and 820-l canisters.
- The storage of the equivalent of 2000,000 210-l canisters of CH-TRUW and of 400,000 210-l canisters of RH-TRUW is equivalent to 70,000 MgHM or 1840 GW/yr electrical capacity. (DOE 1979, Sections 3.3 and 5.3).
- The facility is of a modular design and each module is assumed to be filled to capacity in one year. Therefore, similar storage modules would be required annually to continue storage operations.
- The facility receives wastes from FRPs and MOX FFPs.
- The facility operates 8 hr/d, 5 d/wk, 50 wk/yr for CH waste storage and 20 hr/d, 6 d/wk, 50 wk/yr for RH waste storage.
- The facility receives wastes containing radionuclides assumed to be processed 1.5 years out of a reactor.

16.C.1 SUMMARY

The representative TRUW storage facility receives, handles, and stores both CH-TRUW and RH-TRUW. The CH-TRUW is packaged in canisters that are received in overpacks. The overpacks are removed from their transporters, and the canisters are unloaded. The canisters are then stacked on a pad area until the pad area is full. The canisters are covered with plastic sheeting and then with a layer of dirt.

The RH-TRUW is packaged in canisters that are received in shielding shipping containers (SSC). The SSCs are removed from their transporters, vented and moved into an unloading hot cell. After the main shield doors on the unloading hot cell are closed, all further unloading and storage operations are performed remotely. The canisters are next unloaded from their respective SSCs, inspected, and stored in shielding storage cells.

Under normal conditions the CH-TRUW storage operations produce only nonradioactive effluents: diesel exhaust and water runoff. Under normal conditions the RH waste storage operations produce some nonradioactive and radioactive effluents. The nonradioactive effluents are diesel exhaust, ventilation air, and personnel wash and sewer water. The only radioactive effluents are off-gas from venting shielding shipping containers.

The wastes produced during normal operations at the CH storage area include small amounts of solid wastes. The nonradioactive wastes are broken plywood, damaged plastic sheeting, and miscellaneous trash. The slightly radioactive wastes result from swipes or gloves used in inspecting overpacks or drums. The wastes produced during normal operations at the RH storage building include nonradioactive solid wastes and radioactive liquid and solid wastes. The nonradioactive solid wastes consists mainly of trash. The radioactive liquids result from shielding shipping container decontamination. The radioactive solids consist of failed equipment, maintenance materials, HEPA filters, and general trash.

The representative TRUW storage facility is located on the generic site described in Section 8.0. The overall facility requires 0.6 km² within the larger generic site.

The representative facility requires 2 professionals, 4 supervisors, 15 operators, 5 radiation monitors and 2 maintenance craftsmen. All the CH-TRUW storage operations (receiving, unloading, storing), and 50 percent of the RH-TRUW operations (receiving and unloading) are performed by contact means. About 50 percent of the RH-TRUW operations (unloading and storage) are performed by remote means.

The major amount of nonradioactive waste comes from the CH storage area. These wastes amount to about 60m³/yr of plywood and plastic sheeting. These wastes are all buried onsite. The major radioactive waste comes from the RH operations and consists of liquids and solids. Amounts of these wastes and their radioactivity content are not shown. The liquids are solidified and

packaged and the solids are packaged. The packaged wastes are stored in either the CH or RH storage area, depending on the container dose rate. The major nonradioactive gaseous effluent comes from the RH-TRUW building ventilation air. The total amount of ventilation air and radioactivity content is not known. The ventilation air is monitored and released. The major radioactive gaseous effluent is the off-gas from venting the shielding shipping containers. The amount of off-gas and radioactivity content is not known. The off-gas is HEPA filtered before release to the atmosphere. The major radioactive liquid effluent consists of the CH-TRUW storage area water runoff and the RH-TRUW storage area personnel wash and sewer water. These amounts are not known. These liquids are monitored and transferred to the on-site disposal system.

The total curie content of a full CH-TRUW area module is $9.5E5$ Ci. Transuranic radionuclides make up 99.9 percent of this $9.5E5$ Ci. The total curie content of a full RH-TRUW building module is $3.9E7$ Ci. Activation products amount to $3.2E7$ Ci of this. The only nonradioactive hazardous material is diesel fuel, which is stored in a $7M^3$ tank.

The unit operations at both the CH-TRUW area and RH-TRUW buildings involve primarily solids handling of the radioactive waste containers. The more important accident driving forces of the CH-TRUW areas are the potential energy of dropping the waste canisters with cranes and forklifts. The accident driving forces at the RH-TRUW buildings are the potential energy of dropping the waste containers with cranes and forklifts, and the small amount of thermal energy related to the stored radioactive waste containers. Chemical reactivity (for fires) is the basic driving force in the diesel fuel oil storage area.

The levels of containment at a CH-TRUW area consist of the storage containers (canisters), plastic cover, and dirt cover and the underlayment. The levels of containment at an RH-TRUW building consist of storage containers (canisters) and building structure and ventilation system.

The alternatives for CH-TRUW outdoor storage involve two above-grade concepts (totally exposed surface storage and surface storage with plastic weather cover), and three below-grade concepts (hole-in-ground storage, trench storage, and cave or mine storage). The alternative for CH-TRUW indoor storage is a traditional warehouse storage concept.

The alternatives for RH outdoor storage are a sealed caisson storage concept or a vault storage concept.

16.C.2 MAINLINE PROCESS DESCRIPTION

The primary functions of the transuranic waste (TRUW) storage facility are to receive and store both contact-handled (CH) waste and remote-handled (RH) waste. The CH waste is stored in a separate area from the RH waste but both are located on the TRUW storage facility site.

An overall flow scheme of normal operations for a CH waste storage area is described in Section 16.C.2.1. The CH waste is assumed to be received on truck or railcar packaged in 210-l drums and 3800-l metal boxes transported in overpacks called TRUPACTs (Schneider, et al. 1981). The TRUPACTs are unloaded from the trucks or railcars and then the drums and boxes are removed from the TRUPACTs. The drums and boxes are then stacked on an asphalt pad, covered with plastic sheeting, and covered with earth.

An overall flow scheme of normal operations for a RH waste storage building is described in Section 16.C.2.2. The RH waste is received on truck or railcar packaged in either 210-l drums or 820-l steel cylinders transported in shielding shipping containers. The shielding shipping containers are unloaded from the trucks or railcars and prepared for remote unloading of the drums or cylinders from the shipping containers. The drums and cylinders are unloaded, inspected, and overpacked for storage if necessary. The drums and cylinders are then stored in storage cells.

Retrieval of CH and RH-TRUW for disposal is essentially the reverse of the storage emplacement operations. These operations are discussed in further detail in Section 16.C.2.1 for CH-TRUW and Section 16.C.2.2 for RH-TRUW.

16.C.2.1 Description of Contact-Handled TRUW Storage Operations

This section describes the operations at the contact-handled TRUW storage area. An overall flow scheme of these operations is shown in Figure 16.C.1.

Receiving and Shipping

Transuranic wastes that can be placed into interim storage by contact handling are packaged in either 210-l (55 gallon) drums or 3800-l steel boxes. The drums and boxes are shipped in TRans Uranic PACKAGE Transporters (TRUPACT) by either truck or railcar. Upon arrival at the CH-TRUW storage receiving and shipping area, the TRUPACTs are checked to verify shipping documents, to inspect for damage that could occur during transport, and to survey for surface contamination and radiation levels. The TRUPACT atmosphere is then sampled for contamination. If the TRUPACT checkout is acceptable, the TRUPACT is vented. Decontamination facilities are not provided at the CH-TRUW storage area. Any waste packages received with surface contamination are assumed in this study to be returned to their point of origin for cleanup.

The receiving and shipping area is also used to ship CH-TRUW to a disposal site.

Unloading and Loading

The unloading operations involve unloading the TRUPACTs from the truck or railcar and unloading the drums and boxes from the TRUPACTs. Following the

FIGURE 16.C.1. Overall Flow Scheme for a Representative Contact-Handled TRUW Storage Area

TRUPACT checkout, acceptable TRUPACTs are lifted from the truck or railcar by a mobile yard crane and lowered to a receiving pad. On the receiving pad the TRUPACT is opened and the drums and boxes are removed from the TRUPACTs. The drums and boxes are conveyed to the opening of the TRUPACT by using built-in rollers and manual efforts. A fork-lift truck removes the drums and boxes from the open end of the TRUPACT.

The loading of waste for shipment to a disposal site is essentially the reverse of the unloading operations.

Storage

The drums and boxes are moved from the receiving area to the storage area by fork-lift truck. The storage area is covered by an air-supported structure to enable storage of drums and boxes to continue during inclement weather. The fork-lift truck must therefore enter the air-supported structure through an air lock. Then the drums and boxes are arranged in horizontal layers on an asphalt pad. Sheets of plywood are manually placed over each layer of drums and boxes before another layer is added. As the storage area is filled, polyethylene sheets are laid over the stacked drums and boxes. Back-fill material is hauled in by trucks over the access ramp and roads leading to either side of the area. The bulldozer spreads the backfill from the outside toward the center until all the drums and boxes are covered with earth; the temporary air-supported structure is removed. The top of the fill is then either seeded or covered with a bitumen layer for stabilization.

Operations are reversed to retrieve the CH-TRUW packages for shipment to a disposal site. Any contaminated backfill material will be placed in drums or boxes and stored at another CH-TRUW storage area.

16.C.2.2 Description of the Remotely Handled TRUW Storage Operations

This section describes the operations at the remotely handled TRUW storage building. An overall flow scheme of these operations is shown in Figure 16.C.2.

Receiving and Shipping

Transuranic wastes that need to be placed into interim storage by remote handling are assumed to be packaged in either 210-l (55-gallon) drums or 820-l steel canisters. The drums are shipped in shielding shipping containers (SSCs) by either truck or railcar. The steel cylinders are shipped in SSCs similar to those used to ship spent fuel by either truck or railcar. Upon arrival at the (RH) TRUW building, the truck or railcar is moved into its respective air lock. In the air lock the SSCs are checked to verify shipping documents, to inspect for damage that could occur during transport, and to survey for surface contamination and radiation levels. The truck or railcar is then moved into the unloading area.

The receiving and shipping area is also used to ship RH-TRUW to a disposal site.

FIGURE 16.C.2. Overall Flow Scheme for a Representative Remotely Handled TRUW Storage Building

Transporter Unloading and Loading

A bridge crane is used to lift the shielding shipping containers (SSC) off the transporter and handle them at the unloading area. The SSCs holding the canisters (drums) are designed to be positioned such that the drums can be moved vertically in an upright cylindrical position. The necessary attachments to handle and position the SSCs are part of the SSC construction. The SSCs are placed on a steel transfer pallet in position for a vertical removal of the drums from the SSC. The SSC is secured to the pallet, and the crane is disengaged from the SSC and is returned to the shipping transporter unloading area. The SSCs holding canisters (steel cylinders) have trunnions and a handling yoke for upending and lifting operations. With the crane hoisting mechanism attached to the handling yoke, the SSC is rotated on trunnions to a vertical position. The crane hoist continues to lift the SSC off the transporter and then moves horizontally to a position over the same steel transfer pallet as is used for the SSC holding drums. The SSC is lowered onto the pallet and secured to it. The hoist-lifting yoke is disengaged from the SSCs and the crane is returned to the shipping transporter unloading area. The SSC on the pallet is then moved from the unloading area with an air mover (air-bearing pallet) to the unloading hot cell.

The loading of the SSC containing RH-TRUW onto trucks or railcars for shipment to a disposal site is essentially the reverse of the unloading operations.

Shielding Shipping Container Preparation

If, during the contamination check, the SSC is found to be externally contaminated, a decision is made to cover or otherwise contain the contamination until after the SSC is unloaded or to decontaminate it at this time. The SSC decontamination is done in a pit fitted with a movable spray ring to facilitate decontamination. The SSC attached to its steel pallet is placed in the pit by the unloading crane. Water, steam, or a decontamination solution is sprayed onto the SSC. The resulting liquids are drained into a sump system and collected in holding tanks for treatment. After decontamination, or if it arrives uncontaminated, or if the contamination has been contained, the SSC is taken to the preparation station. Effluent gas from the SSC is vented to the process ventilation system. The roughing and HEPA filters in the vent collection line are monitored at short intervals to detect any unexpected rapid accumulation of radioactive particulates on the filters. Before the SSC (holding drums) is moved to the unloading cell, the SSC head fasteners are partially prepared for removal. Similarly, some of the SSC (holding steel cylinders) head fasteners are removed and an SSC head adapter is installed on the SSC before moving the SSC into the unloading cell. This and subsequent operations where plant personnel are in direct contact with an SSC are closely monitored to ensure that allowable personnel dose limits are not exceeded.

Shielding Shipping Container Unloading

The SSC is next transferred to the unloading cell. To do this, the unloading cell shield doors are opened and the air mover transfers the SSC on

the pallet directly into the cell. The air mover is used to position the SSC under the second floor unloading port doors. Next, the SSC head fasteners are completely removed, the operator leaves the unloading cell, and the shield doors are closed. All operations that require remote handling of RH-TRUW drums or cylinders are performed either in the unloading cell with all shield doors closed, on the second floor storage area, or in the storage cells. These areas are designated as exclusion zones to which personnel entry is forbidden while drums or cylinders are being handled or stored without necessary shielding. Closed-circuit television (CCTV) cameras in the exclusion zone aid viewing and monitoring all handling operations.

The operations for unloading the drums are performed in the following sequence.

1. The second floor unloading port shield doors are opened.
2. The SSC head removal fixture is lowered by the storage area bridge crane and remotely engages the SSC head removal adapter, the SSC head is then lifted through the second floor unloading port and set on the second floor laydown area.
3. A vacuum-operated lifting device is then remotely attached to the storage crane hook.
4. The storage crane and vacuum-operated lifting device are used to remove the drums from the SSC, through the unloading port, and temporarily stored on the second floor laydown area.
5. When all drums have been removed from the SSC, the head is remotely placed back on the SSC, and the second floor unloading port shield doors are closed.

The operations for unloading the cylinders are performed in the following sequence.

1. The second floor unloading port shield doors are opened.
2. The storage area bridge crane and grapple are used to lower the head removal fixture and to remotely engage the SSC head removal adapter. The SSC head is then lifted through the second floor unloading port and set on the second floor laydown area.
3. The storage crane and grapple are used to remove spacers and canisters from the SSC. The cylinders are lifted through the second floor unloading port and temporarily stored on the second floor laydown area.
4. When all the cylinders have been removed from the SSC, the head is remotely placed back on the SSC, and the second floor unloading port shield doors are closed.

The loading of the SSC with drums or cylinders containing RH-TRUW for shipment to a disposal site is essentially the reverse of the unloading operations.

Storage

The operations for storing RH-TRUW drums are performed in the following sequence.

1. Each drum is moved by the storage crane and vacuum-operated lifting device from the second floor laydown area to the inspection station. Each drum is inspected and if not externally contaminated, it is then stored in a storage cell (see item 4, below). The inspection station contains equipment to hold a drum in the vertical position. Visual inspection and testing functions are accomplished through shielding windows located between the control room and the operating gallery. With the aid of a master-slave manipulator, drums are swipe tested. Swipe samples are passed into a shielded drawer. A very sensitive radiation detector is used to monitor the swipe to determine whether the drum is contaminated above allowable levels.
2. If a drum requires overpacking due to surface contamination and/or damage to the drum, the drum is transferred by crane to the overpack station. An empty 310-l (83-gallon) drum overpack in the overpack station enables the contaminated or damaged 210-l (55-gallon) drum to be overpacked by welding with a minimum spread of contamination.
3. After completion of the overpack operation, the overpacked drum is returned by crane to the inspection station for re-inspection (see item 1, above).
4. The storage crane and vacuum-operated lifting device move drums or overpacked drums with clean external surfaces to the appropriate storage cells. All 310-l overpack drums are stored in separate storage cells from the noncontaminated 210-l drums. The 210-l drums are stacked six levels high with an array of 10 x 10 drums for each level for a maximum total of 600 210-l drums per storage cell. The 310-l drums are stacked five levels high with an array of 9 x 9 drums for each level for a maximum total of 405 310-l drums per designated storage cell. Plywood is remotely placed to separate each layer of drums as they are stacked.

The operations for storing RH-TRUW cylinders are performed in the following sequence.

1. Each cylinder is moved by the storage crane from the second floor laydown area to the inspection station. Each cylinder is inspected and if not contaminated, it is then stored in a storage cell (see item 4, below). The inspection station contains equipment to hold a cylinder in the vertical position. Visual inspection and testing

functions are accomplished through shielding windows located between the control room and the operating gallery. With the aid of a master-slave manipulator, cylinders are swipe tested. Swipe samples are passed into a shielded drawer. A very sensitive radiation detector is used to monitor the swipe to determine whether the cylinder is contaminated above allowable levels.

2. If a cylinder requires overpacking due to surface contamination and/or damage to the cylinder, it is transferred by crane to the overpack (weld) station. An empty standby overpack in the weld station enables the contaminated or damaged cylinder to be overpacked by welding with a minimum spread of contamination.
3. After completion of the overpack operation, the overpacked cylinder is returned to the inspection station for re-inspection (see item 1, above).
4. The storage crane moves the cylinders or overpacked cylinders with clean external surfaces to the appropriate cells designated for canisters. The cylinders or overpacked cylinders are lowered into storage racks within the storage cells. The racks allow for one level of cylinders to be stored in each storage cell.

Positioning, viewing and unloading of drums and cylinders is aided by two television cameras (with pan, tilt and zoom lenses) carried by the crane at bridge level, and by spotlights that illuminate the entire work area. When maintenance work is required on the crane, it can be moved to the crane maintenance area, which is separated from the storage area by a guillotine-type shielding door.

The percentage of drums or cylinders needing overpacks is assumed to be only 1 percent of those received.

The retrieval of RH-TRUW drums and cylinders from the storage cells is essentially the reverse of the storage operations.

16.C.3 DESCRIPTION OF EFFLUENT PROCESSING AND WASTE MANAGEMENT SYSTEMS

Effluents in this study are materials that are deliberately discharged to the environment. Wastes are controlled or disposed of with the intent of isolating them from man's environment. Both types of materials are characterized as either radioactive or nonradioactive. This section describes the systems that treat effluents and wastes at the representative TRUW storage facility.

16.C.3.1 Effluent Processing

Under normal conditions, the CH-TRUW storage area operations produce only nonradioactive effluents. The nonradioactive gaseous effluents consist of diesel exhaust from diesel engines on the mobile crane and fork-lift truck.

The nonradioactive liquid effluent consists of noncontaminated water runoff from the dirt covered storage area. The diesel exhaust is not treated before being released. The runoff water drains from the asphalt pad to two plastic-lined retention ponds. The water is monitored for radioactivity and released to the surface drainage system if not contaminated (normal circumstance). If any of the runoff water is contaminated, it would be transferred either to a RH waste storage building radioactive waste water treatment system or to the nearest waste treatment facility based on quantity and scheduling.

Under normal conditions, the RH-TRUW storage building operations produce some nonradioactive and radioactive effluents. The gaseous nonradioactive effluents consist of diesel exhaust and ventilation air. The diesel exhaust is not treated before being released. The ventilation air is normally not treated before release but the air is monitored for radioactivity. If the radioactivity monitor in the exhaust air stream is triggered, it activates a control damper that routes the air through a HEPA filter system. The liquid nonradioactive effluent consists of the personnel wash and sewer water which is monitored for radioactivity and released to a disposal system. If the water should contain radioactivity, it is diverted to a retention pond. The only normal gaseous radioactive effluent is the off-gas from venting the shielding shipping containers. This off-gas passes through a HEPA filter process ventilation system before being released. There are no normal radioactive liquid effluents.

16.C.3.2 Waste Management

Under normal conditions, the CH-TRUW storage area operations result in only very small amounts of solid wastes. These may be nonradioactive or slightly radioactive. The nonradioactive solid wastes would be from miscellaneous trash, broken plywood, or unuseable plastic sheeting. This waste would be buried onsite. The slightly radioactive solid wastes could result from swipes or gloves used in inspecting TRUPACTs or drums for radioactivity. These wastes could be placed in a drum at the storage area and stored with the rest of the CH-TRUW drums.

Under normal conditions the RH-TRUW storage building operations would result in some radioactive liquid wastes, some nonradioactive solid wastes, and some radioactive solid wastes. The radioactive liquid wastes come from the shielding shipping container decontamination operations. The liquids are collected and solidified with concrete in drums and stored either in the RH storage building or the CH storage area depending on the drum surface dose rate. The nonradioactive solids consist of mainly general trash, that are buried onsite. The radioactive solids consist of failed equipment, maintenance materials, HEPA filters, and general trash. These materials are shredded and compacted to reduce volumes, where possible, and packaged in drums. The drums are stored either in the RH storage building or the CH storage area depending on the drum surface dose rate.

16.C.4 PLANT LAYOUT

This section contains descriptions of the TRUW storage site and facility. The TRU waste storage facility consists of the CH-TRUW storage area and the RH-TRUW storage area.

16.C.4.1 Site Description

The representative TRUW storage facility is assumed to be located on the generic site described in Section 8.0. The CH-TRUW storage area consists of modular storage areas that can each hold the equivalent of 10,000 210-L drums. Each CH modular storage area occupies about 0.004 km² (1 acre). The RH-TRUW storage area consists of modular storage buildings that can each hold the equivalent of 20,000 210-L drums. Each RH modular storage building occupies about 0.016 km² (4 acres). The representative TRUW storage facility is assumed to consist of 20 modules each for the CH storage area and RH storage area. Overall, the representative TRUW storage facility itself is assumed to occupy about 0.6 km² (150 acres) inside the larger generic site. The additional area provides for unloading areas, railroad tracks, roads, and service facilities. The facility is enclosed within a security fence with three guarded penetrations--one for truck and railcar shipments to enter, one for truck and railcar shipments to leave, and one for personnel to enter. This fence is posted with warning signs and is continuously and remotely monitored. In addition, a well-labeled perimeter fence encompasses the entire site to exclude the public.

16.C.4.2 Facility Description

The representative TRUW storage facility consists mainly of CH and RH waste handling areas and storage areas. The plot plan for the representative TRU waste storage facility is shown in Figure 16.C.3. The CH waste storage area is based on the storage concept presented in Technology for Commercial Radioactive Waste Management (DOE 1979). The RH waste storage building is primarily based on the storage concept presented in the same reference with some modifications to the receiving, preparation and unloading areas.

CH-TRUW Storage Area

When CH-TRUW is received at the truck and railroad guard house, the trailer and railcar are unhitched from their over-the-road movers and hitched to site-operated truck tractors and railroad donkey engines for subsequent movements. A plan and section view of a CH-TRUW modular storage area is shown in Figure 16.C.4. All structures, systems, and components at the CH-TRUW storage area are Design Class III items. (Design Class III relates to those structures, components, and systems whose failure will not result in release of excessive amounts of radioactivity and whose operations is not essential to safe operation.)

The receiving and shipping area of the CH waste storage area is basically the section of roadway and railroad track next to the unloading area. The

1. OFFICE AND SUPPORT AREA
2. PERSONNEL PARKING AREA
3. MAIN SECURITY GUARD HOUSE
4. TRUCK AND RAILROAD GUARD HOUSE
5. RAIL CAR AIR LOCK
6. TRUCK AIR LOCK
7. RH UNLOADING AND PREPARATION AREA
8. RH STORAGE CELLS
9. CH STORAGE AREA
10. UTILITY BUILDING

FIGURE 16.C.3. Plot Plan for the Representative TRUW Storage Facility
(Showing One Module of Each CH-TRUW and RH-TRUW Areas)

FIGURE 16.C.4. Contact-Handled Transuranic Waste Storage Area Plan and Section Views (DOE 1979)

unloading and loading area consists of the TRUPACT receiving pad. A mobile yard crane is used to unload the TRUPACTs from the truck trailers or railcars to the receiving pad. A fork-lift truck is used to transfer the 210- ℓ drums and 3800- ℓ boxes from the TRUPACT to the storage area.

The storage area basic module is constructed around a curbed asphalt pad 19.5 m (64 ft) wide by 50 m (164 ft) long having a center crown to allow water, if present, to flow to the sides of the pad and drain to the retention ponds. Part of the final fill material is accumulated initially at three sides of the asphalt pad and is used to form a berm. An access road is provided on top of the berm for use during backfilling over the drums. The bulldozer is used to backfill over the drums. A temporary air-supported structure is installed over the basic module, allowing storage operations to continue during inclement weather. The structure is 32-m (105 ft) wide, 69-m (226 ft) long, and 12-m (40 ft) high at its center. It incorporates ventilation and inflation equipment, two truck-size air locks and a personnel access door. A continuous concrete beam at grade level is required all around the area for anchorage of the air-supported structure. Two plastic-lined retention ponds are provided for collection of water that drains from the asphalt pad.

A personnel building is provided for the convenience of the workers in the facility. It is a 5-m (16-ft) by 10-m (33-ft) light structural steel building with internal partitions. It is insulated and weather-proofed and contains an office, change room and wash room. Portable lighting is provided for operations as needed.

RH Waste Storage Area

When RH waste is received at the truck and railroad guard house, the trailer and railcar are unhitched from their over-the-road movers and hitched to site-operated truck tractors and railroad donkey engines for subsequent movements. Plan and section views of an RH waste storage modular building are shown in Figure 16.C.5.

The receiving and shipping areas of the RH waste storage building are the air locks through which the trailers and railcars must pass to enter or leave the building. The air locks are built to Design Class II specifications. [Design Class II relates to those structures, components, and systems important to the operation of the RH-TRUW area but not essential to the long-term safety of the facility following a Maximum Credible Event (MCE).] Each air lock has room enough for one truck or railcar and its on-site mover. The air locks ensure that the interior of the RH waste handling area is maintained at a pressure lower than atmospheric pressure. This prevents air from the waste handling area from passing directly to the exterior environment. The doors at each end of the air lock are interlocked to prevent both doors from becoming opened simultaneously.

The unloading and preparation area is the portion of the RH waste storage building on the first floor between the office and support area and the storage cell area. All the structures, systems, and components in this area are Design

FIGURE 16.C.5. Remote-Handled Transuranic Waste Storage Building
Module Plan and Section Views

Class I items. [Design Class I relates to those structures, components, and systems, including instruments and controls, whose failure might cause the release of excessive amounts of radioactivity or that are essential to safe shutdown of operations without endangering the public health or safety during or following a Maximum Credible Event (MCE).] The bridge crane used to unload/load the RH waste shielding shipping containers has a 127-Mg (140-ton) capacity main hoist and a 23-Mg (25-ton) auxiliary hoist. The transfer pallet used for holding, securing, and moving the RH shielding shipping containers is a steel fabricated 127-Mg (140-ton) capacity structure. The pallet has a means for centering and orienting the shielding shipping containers (SSC) in a vertical position. The pallet rides on a mating, air-bearing-type self-propelling pallet transporter or "air mover" of 136-mg (150-ton) capacity. The transporter draws air from the plant air system. A drawing of the steel transfer pallet and air mover is shown in Figure 16.C.6. The ventilation system for SSC venting consists of an exhaust fan, roughing and HEPA filters. A radioactivity monitor detects unexpected rapid accumulation of radioactive particulates on the filters and shuts down the venting operation upon activation. The decontamination pit for cleaning the externals of the SSC is below the floor level and requires the use of the unloading crane to lower and raise the SSC mounted on the pallet into or out of the pit. The decontamination pit contains a sump and a sump pump. The sump pump transfers liquids to the holding tanks for treatments. The unloading cell internal dimensions of 3.0-m (10-ft) by 3.0-m (10-ft) by 13.7-m (45-ft) in height. The unloading cell has 1.2m (4-ft) thick concrete walls and shield doors on one side of 7.6-m (25-ft) in height. The unloading cell also has two shield doors on the ceiling that separates the unloading cell from the second floor remote handling areas.

The storage area consists of the second floor areas such as the laydown area inspection station, and overpack station and the storage cells. The laydown area is simply the open space on the second floor for temporary storage of drums and canisters until they can be transferred to a storage cell. The inspection station is a second floor location near a control room shielding window where a drum or canister can be visually inspected and swipe tests performed. Swipe tests are made with a manipulator operated from the control room and swipes are transferred to the control room in a shielded pass-through drawer. The overpack station contains equipment to hold and rotate overpacks during loading and welding. An automatic welding machine at the overpack station welds the overpack containers closed and makes repair welds. Lead glass shielding windows are located between the control room and the work stations. Master-slave manipulators are located in pairs at both the inspection station and the overpack station. The storage cells are 6.1-m (20-ft) square and 6.1-m (20-ft) high. An additional 6.1-m (20-ft) above the cells is needed for crane operation. The exterior/cell walls are 1.2-m (4-ft) thick and separation walls between cells are 0.6-m (2-ft) thick. All interior surfaces of the cell walls, floors, and ceilings are smoothly finished with a sealant compound. Air circulation through the storage cells is provided by openings in the walls of each cell. The air is exhausted through ductwork to a stack. The exhaust ventilation is monitored and if necessary the exhaust air can be automatically diverted through a HEPA filter system prior to release to

FIGURE 16.C.6. Steel Transfer Pallet with Shielding Container (for cylinders) and Air Mover (DOE 1980).

the atmosphere. The storage cells used to store RH-TRUW cylinders have racks to separate and hold the cylinders vertically. The racks also allow adequate air ventilation to remove decay heat. The bridge crane used to move drums and cylinders to temporary storage, inspection station, overpack station and storage cells is a 13.6-Mg (15-ton) capacity crane.

The support systems for the RH waste storage building are as follows:

- heating ventilation, and air conditioning
- instrumentation and controls
- compressed air system

Support Systems for the TRU Waste Storage Facility

The following support systems are necessary for both the CH and RH waste storage areas:

- off-site electric power and emergency diesel electricity generator and their distribution systems
- fire protection

16.C.5 SAFETY-RELATED INFORMATION

This section discusses the pertinent safety-related information that can be used to help identify the risks associated with the representative TRUW storage facility.

16.C.5.1 Operating and Maintenance Requirements

The operating and maintenance requirements are given first for the CH-TRUW storage areas and then the RH-TRUW storage buildings.

CH-TRUW

The CH-TRUW storage area is available for storage operations one 8-hour shift/d, 5 d/wk, and 50 wk/yr. The personnel required for the CH-TRUW storage operations are as follows:

- supervisor - 1 man-year/yr
- operators - 3 man-year/yr
- radiation monitors - 2 man-year/yr
- maintenance craftsmen - 0.4 man-year/yr

All of the CH storage operations are as indicated: contact operations (i.e., operators directly contact handle the waste containers). Cranes and fork-lift trucks are used to handle the overpacks, drums, and boxes, not because of the radioactivity, but because of the efficiency of operation. Based on receiving 50 percent of the CH waste drums and boxes by railcar and the other 50 percent by truck, the number of loads per year is 70 by rail and 139 by truck. A

railcar can transport two TRUPACTs (see Section 19) that can each hold 36 drums, and a truck can transport one TRUPACT that can hold 36 drums. The handling rate for CH waste canisters is 40 drums/8-hour day.

RH-TRUW

The RH-TRUW storage building is available for storage operations two 10-hour shifts/day, 6 days/week, and 50 weeks/year. The personnel required for the RH waste storage operations are as follows:

- professionals - 2 man-year/yr
- supervisors - 3 man-year/yr
- operators - 12 man-year/yr
- radiation monitors - 3 man-year/yr
- maintenance craftsmen - 1.6 man-year/yr

The RH storage operations involve both contact and remote operations. The shielding shipping containers are contact-handled until they are located in the unloading cell, then the drums and cylinders are removed, inspected, and stored remotely. Thus, approximately 50 percent of the operations are contact operations and 50 percent are remote operations. It is assumed that about 70 percent is received in drums and about 30 percent of all RH waste is received in cylinders. One cylinder-type of canister contains approximately 3.77 times the volume of waste that a single drum-type of canister contains. Thus, based on a storage area for 20,000 drums, an RH waste storage building receives 14,250 drums and 1,525 cylinders. Based on receiving 50 percent of the RH waste by railcar and the other 50 percent by truck, the numbers of railcar loads per year are 792 for drums and 254 for cylinders. The numbers of truck loads per year are 2,375 for drums and 762 for cylinders. The railcar casks for carrying cylinders are assumed to hold three cylinders. The railcars are assumed to carry three shielding shipping containers for drums, each of which contains three drums per container for a total of nine drums per railcar. The truck shielding shipping containers for carrying cylinders are assumed to hold one cylinder. The trucks are assumed to carry one shielding shipping container holding three drums. The handling rate for RH waste canisters is 5 per 20-hour day and for RH waste drums is 48 per 20-hour day.

16.C.5.2 Input/Output of Major Materials

This section discusses the inputs and outputs of major materials at the TRUW storage facility. The materials covered in this section include RH-TRUW and CH-TRUW in cylinders and drums, non-radioactive input materials, effluents, and wastes. The representative facility receives the volumetric equivalent of wastes to fill a modular CH waste storage area and a modular RH waste storage building in one year.

RH-TRUW in Drums

A representative RH storage building receives 14,250 drums, or about 3087 m³ (109,000 ft³), of RH-TRUW. Table 16.C.1 shows an average

TABLE 16.C.1. Average Radioactivity Content Per Drum-Type of
Canister Containing RH-TRUW
(DOE 1979, Sections 3.3 and 5.3)

<u>Fission Products</u>	<u>Average Ci/Drum</u>
^3H	8.4E-2
^{129}I	2.5E-5
$^{90}\text{Sr} + ^{90}\text{Y}$	9.8E-1
$^{95}\text{Zr} + ^{95}\text{Nb}$	2.2E 0
$^{106}\text{Ru} + ^{106}\text{Rh}$	7.6E 1
$^{134}\text{Cs} + ^{137}\text{Cs} + ^{137\text{m}}\text{Ba}$	2.4E 0
$^{144}\text{Ce} + ^{144}\text{Pr}$	3.8E 0
All other FPs	<u>7.4E-1</u>
Total FPs	8.6E 1
<u>Actinides</u>	
^{239}Pu	3.2E-1
^{241}Pu	1.6E 2
Other Pu	5.6E 0
$^{242}\text{Cm} + ^{244}\text{Cm}$	1.4E-1
All other actinides	<u>1.1E-2</u>
Total Actinides	1.7E 2
<u>Activation Products</u>	
^{14}C	5.5E-6
^{55}Fe	2.0E-1
^{60}Co	2.1E-1
$^{95}\text{Zr} + ^{95}\text{Nb}$	1.4E-2
All others	<u>3.3E-2</u>
Total Activation Products	4.6E-1

radioactivity content for a drum containing RH-TRUW at 1.5 years out of the reactor. Table 16.C.2 gives a description of the representative TRUW drum.

RH-TRUW in Cylinders

A representative RH storage building receives 1,525 cylinders, or about 1,246 m³ (44,000 ft³) of RH-TRUW. Of these 1,525 cylinders, 1,331 contain spent fuel hulls, hardware, and sand, and 194 cylinders contain other Fuel Reprocessing Plant (FRP) RH-TRUW. Table 16.C.3 gives a description of the representative cylinder. Table 16.C.4 shows the average radioactivity content for a cylinder containing spent fuel hulls and hardware at 1.5 years out of the reactor. A single cylinder of spent fuel hulls and hardware contains 698 kg (1,540 lbs) of spent hulls, 147 kg (324 lbs) of hardware, and 849 kg (1870 lbs) of sand. Table 16.C.5 shows the average radioactivity content for a cylinder containing other FRP RH-TRUW at 1.5 years out of the reactor.

CH-TRUW in Drums and Boxes

A representative CH storage area receives the equivalent of 10,000 drums or about 2167 m³ (76,500 ft³) of CH-TRUW. Of the 10,000 drums, 6,055 drums originate at the FRP and 3,945 drums originate at the MOX fuel fabrication plant (FFP). Table 16.C.6 shows an average radioactivity content for a drum and a box containing CH-TRUW from the FRP at 1.5 years out of the reactor. Table 16.C.7 shows an average radioactivity content for a drum and a box containing CH-TRUW from the MOX FFP at 1.5 years out of the reactor plus 1 more year to process, or 2.5 years total out of the reactor.

TABLE 16.C.2. Description of the Representative TRUW
Drum Type of Canister

Drum Size

Inside dimensions	0.57 m dia. x 0.84 m long (22.5 in. dia. x 33.25 in. long)
Outside dimensions	0.61 m dia. x 0.89 m long (24 in. dia. x 35 in. long)
Wall thickness	0.03 m (0.75 in.)
Drum Weight Empty	31 kg (68 lbs)

TABLE 16.C.3. Description of the Representative TRUW Cylinder Type of Canister

Cylinder Size	
Inside dimensions	0.58 m dia. x 3.05 m long (23 in. x 10 ft)
Outside dimensions	0.61 m dia. x 3.07 m long (24 in. x 10 ft. 1 in.)
Wall thickness	0.01 m (0.5 in.)
Cylinder Material	stainless steel (assumed to have a specific gravity of 8.0)
Cylinder Weight Empty	626 kg (1,380 lbs)

TABLE 16.C.4. Average Radioactivity Content per Cylinder-Type of Canister Containing Spent Fuel Hulls and Hardware (DOE 1979, Section 3.3)

<u>Fission Products</u>	<u>Ci/Canister</u>
^3H	1.7E 2
^{129}I	negligible
$^{90}\text{Sr} + ^{90}\text{Y}$	1.6E 2
$^{95}\text{Zr} + ^{95}\text{Nb}$	1.4E 1
$^{106}\text{Ru} + ^{106}\text{Rh}$	5.0E 2
$^{134}\text{Cs} + ^{137}\text{Cs} + ^{137m}\text{Ba}$	3.9E 2
$^{144}\text{Ce} + ^{144}\text{Pr}$	6.3E 2
All other FPs	1.2E 2
Total FPs	2.0E 3
<u>Actinides</u>	
^{239}Pu	4.7E-1
^{241}Pu	2.4E 2
Other Pu	8.2E 0
$^{242}\text{Cm} + ^{244}\text{Cm}$	2.2E 1
All other actinides	1.2E 0
Total Actinides	2.7E 2
<u>Activation Products</u>	
^{14}C	2.9E-1
^{55}Fe	1.1E 4
^{60}Co	1.1E 4
$^{95}\text{Zr} + ^{95}\text{Nb}$	7.3E 2
All others	1.7E 3
	2.4E 4

TABLE 16.C.5. Average Radioactivity Content Per Cylinder-Type
of Canister Containing Other FRP RH-TRUW
(DOE 1979, Sections 3.3 and 5.3)

<u>Fission Products</u>	<u>Average Ci/Canister</u>
^{129}I	1.5E-8
$^{90}\text{Sr} + ^{90}\text{Y}$	5.4E-2
$^{95}\text{Zr} + ^{95}\text{Nb}$	4.8E-3
$^{106}\text{Ru} + ^{106}\text{Rh}$	1.7E-1
$^{134}\text{Cs} + ^{137}\text{Cs} + ^{137\text{m}}\text{Ba}$	1.3E-1
$^{144}\text{Ce} + ^{144}\text{Pr}$	2.1E-1
All other FPs	<u>4.1E-2</u>
Total FPs	6.1E-1
 <u>Actinides</u>	
^{239}Pu	1.6E-4
^{241}Pu	7.9E-2
Other Pu	2.7E-3
$^{242}\text{Cm} + ^{244}\text{Cm}$	7.5E-3
All other actinides	<u>4.0E-4</u>
Total Actinides	9.0E-2
 <u>Activation Products</u>	
None	

TABLE 16.C.6. Average Radioactivity Content Per Drum and Box
Types of Canisters Containing FRP CH-TRUW
(DOE 1979, Sections 3.3 and 5.3)

<u>Fission Products</u>	<u>Average Ci/Drum</u>	<u>Average Ci/Box</u>
^3H	4.2E-8	0
^{129}I	3.5E-6	1.4E-9
$^{90}\text{Sr} + ^{90}\text{Y}$	1.2E-3	4.9E-3
$^{95}\text{Zr} + ^{95}\text{Nb}$	7.7E-4	4.4E-4
$^{106}\text{Ru} + ^{106}\text{Rh}$	2.7E-2	1.5E-2
$^{134}\text{Cs} + ^{137}\text{Cs} + ^{137m}\text{Ba}$	3.0E-3	1.2E-2
$^{144}\text{Ce} + ^{144}\text{Pr}$	4.8E-3	1.9E-2
All other FPs	<u>9.3E-4</u>	<u>3.7E-3</u>
Total FPs	3.8E-2	5.5E-2
<u>Actinides</u>		
^{239}Pu	2.2E-2	7.2E-2
^{241}Pu	1.1E 1	3.6E 1
Other Pu	3.7E-1	1.2E 0
$^{242}\text{Cm} + ^{244}\text{Cm}$	6.8E-1	3.4E 0
All other actinides	<u>3.7E-2</u>	<u>1.8E-1</u>
Total Actinides	1.2E 1	4.1E 1
<u>Activation Products</u>		
^{14}C	7.7E-11	0
^{55}Fe	2.9E-6	0
^{60}Co	2.9E-6	0
$^{95}\text{Zr} + ^{95}\text{Nb}$	2.0E-7	0
All others	<u>4.6E-7</u>	<u>0</u>
Total Activation Products	6.5E-6	0

TABLE 16.C.7. Average Radioactivity Content Per Drum and Box
Types of Canisters Containing MOX FFP CH-TRUW
(DOE 1979, Sections 3.3 and 5.3)

<u>Fission Products</u>	<u>Average Ci/Drum</u>	<u>Average Ci/Box</u>
Any	Negligible	Negligible
<u>Actinides</u>		
^{239}Pu	5.2E-1	1.3E-4
^{241}Pu	2.3E 2	5.8E-2
Other Pu	8.6E 0	2.2E-3
All other actinides	<u>7.7E-1</u>	<u>9.8E-5</u>
Total Actinides	2.4E 2	6.0E-2
<u>Activation Products</u>		
Any	None	None

Nonradioactive Input Materials.

Other materials are utilized at the representative facility for drum layer separation, CH storage area coverage of drums, shielding shipping container decontamination, RH storage overpack operations, and fuel. The nonradioactive innput materials required by the CH storage areas and the RH storage areas on a per-year basis are given in Table 16.C.8.

TABLE 16.C.8. Nonradioactive Input Materials at the TRUW
Storage Facility

<u>Input Material</u>	<u>Amounts Needed in the CH Storage Area</u>	<u>Amounts Needed in the RH Storage Area</u>
Plywood (for drum layer separation, 1.9 cm thick)	3900 m ² (42,000 ft ²)	4420 m ² (47,600 ft ²)
Plastic sheeting (for drum coverage)	1750 m ² (18,800 ft ²)	
Decontamination detergents		Unknown amount
1100ℓ overpack cylinders		15 cylinders
310ℓ overpack drums		143 drums
Diesel fuel	30,000ℓ (7930 gal)	52,800ℓ (13,900 gal)

Effluents

Effluents are either radioactive or nonradioactive as described in Section 16.C.3. Refer to Table 16.C.9 for quantities of effluents.

TABLE 16.C.9. Annual Release Quantities of Airborne and Liquid Effluents at the TRUW Storage Facility

	<u>Amount From CH Storage Area</u>	<u>Amount From RH Storage Area</u>
<u>Airborne Effluents</u>		
Diesel Exhaust ^(a)		
CO	304 kg (670 lbs)	544 kg (1200 lbs)
NO _x	200 kg (440 lbs)	349 kg (770 lbs)
SO _x	113 kg (250 lbs)	200 kg (440 lbs)
Aldehydes	24 kg (54 lbs)	43 kg (95 lbs)
Particulate	91 kg (200 lbs)	159 kg (350 lbs)
Hydrocarbon	109 kg (240 lbs)	191 kg (420 lbs)
Ventilation Air	-	unknown
Off-Gas from Venting the Shielding Shipping Containers	-	unknown
<u>Liquid Effluents</u>		
Water Runoff	unknown	-
Personnel wash and sewer water	-	unknown

(a) (EPA 1973)

For the CH storage area, the only effluents are diesel exhaust and water runoff from the dirt-covered storage area. The amount of water runoff is dependent on the amount of rainfall and snowfall as defined in the representative site description in Section 8. For the RH storage area, the effluents are diesel exhaust, ventilation air, personnel wash and sewer water, and off-gas from venting the shielding shipping containers. The quantity of ventilation air and personnel wash and sewer water is not known. Both of these are normally not contaminated. The quantity of off-gas from venting the shielding shipping containers is not known, but could contain amounts of the radionuclides as indicated in Tables 16.C.1, 16.C.4 and 16.C.5.

Wastes

The wastes generated at the TRUW storage facility are either nonradioactive or radioactive as described in Section 16.C.3. For the CH storage area the nonradioactive wastes are the following:

● damaged plywood (1.9 cm thick)	39 m ² (420 ft ²)
● damaged plastic sheeting	18 m ² (188 ft ²)
● miscellaneous trash	unknown

The only radioactive wastes from the CH storage area are slightly contaminated swipe or gloves used for contamination checks. The amounts of these wastes are unknown. The RH storage building operations produce radioactive liquid waste from shielding shipping container decontamination and the amounts are unknown. The nonradioactive solid wastes consist of general trash of which the amount is unknown. The amounts of radioactive solid wastes from failed equipment, maintenance materials, HEPA filters, and general trash are unknown.

16.C.5.3 Inventories of Major Hazardous Materials at Plant Locations Having Safety-Related Importance

The major inventory of radioactive materials at the representative TRUW storage facility is located either in the CH storage area or the RH storage cells. The quantity, as drums or cylinder types of canisters, of these materials are given in Section 16.C.5.2 and the radionuclide content per container is indicated in Tables 16.C.1, 16.C.4 and 16.C.5 for the RH Storage and in Tables 16.C.6 and 16.C.7 for the CH storage. An overall inventory of tables is given in Table 16.C.10.

16.C.5.4 Unit Operations Involved

This section identifies the unit operations involved for each step in the process flow schemes for the representative TRUW storage facility. (See Figures 16.C.1 and 16.C.2.) Table 16.C.11 lists these unit operations.

16.C.5.5 Accident Driving Forces

This section identifies the driving forces that could potentially cause accidents at the reference facility. These driving forces and their locations are given in Table 16.C.12.

16.C.5.6 Containment and Shielding Systems

The containment and shielding at the CH waste storage area is basically provided by the storage containers. After the storage area is filled and covered with plastic and dirt, the dirt provides about 0.91-m (3-ft) of additional shielding.

At the RH waste storage area, the containment of radioactive material is primarily provided by the drums or canisters and secondarily provided by the building structure and ventilation system. The storage cell exterior walls are

TABLE 16.C.10. Inventories of Hazardous Materials at Various TRUW Storage Facility Locations

Location/Material	Form			Radionuclide Content (Ci) ^{(a)(b)(c)}									
	Chemical	Phase		N.G.	I	SVFP	³ H	AOFP	ULA	TRU	AP	Total	
<u>CH Storage Module (full)</u>													
CH-TRUW in 210-l canisters from FRP	Many	Solid	1810 m ³	NEG.	1.6E-2	7.2E+1	2.0E-4	1.0E+2	2.6E+1	9.5E+5	3.0E-2	9.5E+5	
			1010 m ³	NEG.	1.6E-2	7.2E+1	2.0E-4	1.0E+2	1.2	5.6E-4	3.0E-2	5.6E+4	
			800 m ³	NEG.	NEG.	NEG.	NEG.	NEG.	2.5E+1	8.9E+5	0.0	8.9E+5	
CH-TRUW in 3800-l canisters from FRP	Many	Solid	356 m ³	NEG.	4.8E-8	5.6E-1	NEG.	1.3	4.4E-2	1.4E+3	0.0	1.4E+3	
			88 m ³	NEG.	4.8E-8	5.6E-1	NEG.	1.3	4.4E-2	1.4E+3	0.0	1.4E+3	
			258 m ³	NEG.	NEG.	NEG.	NEG.	NEG.	1.7E-4	6.2	0.0	6.2	
<u>RH Storage Building Module (full)</u>													
RH-TRUW in 210-l canisters	Many	Solid	3087 m ³	NEG.	3.5E-1	5.7E+5	1.2E+3	6.6E+5	4.2E+1	2.4E+6	6.5E+3	3.6E+6	
RH-TRUW in 820-l canisters (contains fuel hulls)	Many	Solid	1087 m ³	NEG.	NEG.	7.2E+5	2.2E+5	1.7E-6	1.1E+1	3.6E+5	3.2E+7	3.5E+7	
RH-TRUW in 820-l canisters (other than fuel hulls)	Many	Solid	1159 m ³	NEG.	3.0E-6	3.5E-1	NEG.	8.3E-1	5.2E-4	1.7E+1	NEG.	1.4E+2	
<u>Diesel Fuel Oil Storage</u>													
Diesel Fuel	Hydro Carbons	Liquid	7 m ³	Natural Amounts									

(a) The basis for these numbers is one full module each of CH-TRUW and RH-TRUW with each module being filled in one year.

(b) Abbreviations are as follows: N.G.: Noble Gases; I: Iodine; SVFP: Semi-Volatile Fission Products. Total contribution due to ruthenium, cesium, rubidium, tellurium, and antimony ³H: Tritium; AOFP: All Other Fission Products; ULA: Uranium plus Lower Actinides; TRU: Transuranic Nuclides; AP: Activation Products.

(c) NEG: Negligible contribution to total activity.

TABLE 16.C.11. Unit Operations Involved in the Representative TRUW Storage Facility

<u>Process Step</u>	<u>Unit Operation</u>
<u>CH storage operations</u>	
<u>Overpack receiving and shipping</u> Cask venting	Gas handling (venting and filtering)
<u>Transport and overpack unloading/loading</u> Transport unloading TRUPACT unloading	Solids handling (mechanical lifting) Solids handling (conveyance)
<u>Storage</u> Placement of drums Placement of plywood Placement of plastic Placement of dirt cover	Solids handling (conveyance) Solids handling (mechanical and manual lifting, conveyance) Solids handling (conveyance) Solids handling (conveyance)
<u>Effluent and waste management</u> Runoff water handling Radioactive waste handling Nonradioactive waste handling	Liquid sampling Solids handling (packaging) Solids storage Solids handling (shredding, compaction) Solids burial
<u>RH storage operation</u>	
<u>Cask or overpack receiving and shipping</u> Transporter moved into airlock	Solids handling (conveyance)
<u>Transporter unloading and loading</u> Transporter unloading	Solids handling (mechanical lifting)
<u>Overpack or cask preparation</u> Decontamination (nonroutine) Venting Transfer to unloading cell	Liquids handling (pumping) Gas handling (pumping) Solids-gas separation (filtration) Solids handling (conveyance)
<u>Overpack or cask unloading</u> Removal of drums or canisters	Solids handling (lifting)
<u>Storage</u> Inspection Overpacking (possible) Placement of drums or canisters Placement of plywood	Solids handling (conveyance) Solids handling (conveyance) Solids melting (welding) Solids handling (lifting, conveyance) Solids handling (lifting, conveyance)
<u>Effluent and waste management</u> Ventilation Personnel wash and sewer water Radioactive liquid wastes Nonradioactive solid wastes Radioactive solid wastes	Gas handling (pumping) Solid-gas separation (filtration) Liquid sampling Liquids distillation (evaporation) Liquids handling (pumping) Solid-liquid solidification (cementation) Solids handling (drum handling) Solids handling (shredding, compaction, packaging) Solids burial Solids handling (size reduction, packaging) Solids storage

TABLE 16.C.12. Accident Driving Forces at the Representative TRUW Storage Facility

<u>Accident Driving Force</u>	<u>Location</u>	<u>Characteristics</u>
Chemical reactivity	Diesel fuel oil storage Plywood, plastic	Flammable
Potential energy	Shielding shipping container handling areas	Shielding shipping container drop
	Storage cells	Canister drop
Thermal energy	Canister storage cells	20 kW/cell

1.2-m (4-ft) thick concrete. Exterior walls and interior separation walls provide shielding for all operations outside the exclusion areas. The building ventilation system provides confinement for airborne radionuclides. The ventilation system is designed for once-through flow of air and provides controlled in-leakage of air by the use of air locks at the facility entrances. Flow of air is from areas with lesser potential for contamination to areas with higher potential for contamination. Normally, the storage cell ventilation air is exhausted directly to the atmosphere without treatment. However, in the event that air does not meet release guidelines, a continuous radiation monitor in the exhaust duct will trip a control damper that diverts the air flow through HEPA filters before it passes to the exhaust stack. The off-gas exhaust system filters particulates. Air is exhausted into this system from the overpack and cask ventilation operation.

16.C.5.7 Degree of Hands-On Operation

Refer to the discussion in Section 16.C.5.1.

16.C.6 ALTERNATIVE SCHEMES FOR THE TRUW STORAGE FACILITY

This section contains brief descriptions of the major alternative TRU waste storage concepts.

16.C.6.1 Alternatives for the Contact-Handled Waste Storage Area

The major alternatives for CH waste storage include variations of outdoor or indoor storage. In the outdoor storage, waste can be stored either above or below the ground. The representative facility uses outdoor above ground storage. The other alternatives for outdoor storage are illustrated in Figure 16.C.7.

FIGURE 16.C.7. Alternative Outdoor Storage Concepts for CH-TRUW (DOE 1979)

These concepts include: 1) totally exposed surface storage, 2) surface storage with plastic weather cover, 3) below-grade, hole-in-ground storage, 4) below-grade trench storage, and 5) below-grade cave or mine storage. Starting from the left of Figure 16.C.7, concept 1, totally exposed surface storage, may be applicable to waste containers that have demonstrated a service life for the duration of storage without additional protection. Concept 2 provides additional weather protection and some additional confinement in the event of canister failure. The three below-grade storage concepts, concepts 3 to 5, provide a certain amount of additional environmental protection for the waste canisters. Concepts 1 and 2 are not as resistant to the spread of fire, contamination spread from leakage, or vandalism as the representative concept. Concept 3 does not provide the versatility in accommodating the variety of waste canisters as the representative concept. Concepts 4 and 5 may meet or exceed the same basic considerations for storage as the representative concept. The same basic unit operations and accident driving forces take place or exist for all of the outdoor storage concepts.

The alternative for CH indoor storage is an above-ground concept as shown in Figure 16.C.8. The concept consists of a traditional warehouse for storage and requires no special handling equipment or shielding for storing the waste canisters. Variations in the concept are mostly in the choice of structural materials. Each module of the building would contain two storage cells plus service and transfer areas. The drums would be stacked in the storage cell by forklift. When the cell was full, a sliding door would be closed and sealed. Drums could be inspected and monitored through storage cell roof hatches. The basic unit operations and accident driving forces are essentially the same for the alternative indoor concept as for the representative outdoor concept.

FIGURE 16.C.8 Alternative Indoor CH-TRUW Storage Above Ground Concept (DOE 1979)

16.C.6.2 Alternatives for the Remotely Handled Waste Storage Area

The major alternatives for RH-TRUW storage also include variations of indoor or outdoor storage. The indoor storage concept is the representative RH waste storage concept. There are two basic major alternatives for outdoor storage: sealed caisson storage or vault storage. The sealed caisson concept can be designed for above grade or below grade. Figure 16.C.9 illustrates an above-grade sealed caisson concept. The facility consists of two parallel above-grade berms constructed from suitable soils materials; waste containers are stored in individual steel caissons within the berms. The individual storage positions have removable shielding plugs 1.2 m (4 ft) thick to allow for storage and retrieval of containers through a special-purpose hulls transfer device. A 54-Mg (60-ton) gantry crane runs over one-half of the berm and handles the hulls cask, the hulls transfer device, and the shielding plugs. Cooling of the berms is achieved by conduction through the soils to the atmosphere. (DOE 1979, Section 5.2)

The vault storage concept is illustrated in Figure 16.C.10. The vault storage concept consists of a modular arrangement of partially buried reinforced concrete structures called cells. The individual cells have roof slabs 1.2-m (4 ft) thick in which removable shielding plugs are built to allow storage and retrieval of waste canisters. A 54-Mg (60-ton) gantry crane is provided that services over one-half of the storage unit at a given time to handle the transfer cask, the hulls transfer device and the shielding plugs. Rails for the crane are provided for both halves of the storage unit. The vault is cooled by natural air circulation. (DOE 1979, Section 5.2)

The two outdoor alternative concepts are very similar to the representative indoor concept regarding safety-related aspects such as operations and maintenance requirements, input/output of major materials, inventories of hazardous materials, unit operations, and accident driving forces. The main difference is that the outdoor alternative concepts provide less containment than the indoor concept in the event of a release while loading or unloading canisters.

FIGURE 16.C.9. Alternative Outdoor RH-TRUW Above-Grade Sealed Caisson Concept (DOE 1979)

16.C.37

FIGURE 16.C.10. Alternative Outdoor RH-TRUW Vault Storage Concept (DOE 1979)

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17.0 SPENT FUEL AND HIGH-LEVEL AND TRANSURANIC WASTE DISPOSAL

The purpose of the geologic disposal facility described in this section is to permanently dispose of high-level waste (HLW), transuranic waste (TRUW) and spent unprocessed fuel (SURF). Because of the uniqueness and novelty of disposal of long-lived, highly-radioactive materials, a retrievability option is included in the event that early assessment indicates the actual performance of the repository is inadequate. The facility is designed to handle 10-year-cooled wastes of the spent fuel or high-level wastes from commercial nuclear power reactors, namely Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs), which include all those in the U.S. except for the Fort St. Vrain High Temperature Gas-Cooled Reactor (HTGR). Only the facility description related to the operational or preclosure period of the repository is given in this section. The preclosure period includes the times for construction of surface and mine facilities, emplacement of HLW and TRUW or SURF, closure of the mine and decommissioning.

At present (mid-1982), there are no disposal facilities for the stated purpose in the U.S. Furthermore, only a few conceptual designs have been prepared and these were prepared without benefit of the final Code of Federal Regulations 10CFR60 and 40CFR191 under preparation by the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA), respectively. The facility described below is generic in nature and is located above a bedded salt formation on a generic site (see Section 6). The facilities are based primarily on the conceptual design of a SURF-Bedded Salt Repository (Ritchie 1978), although the conceptual design of a HLW-Domed Salt facility was also extensively considered (Jones 1979).

Presently, SURF from commercial nuclear power plants is being stored in internal water pools at the nuclear power generation stations and some external water pools at such locations as General Electric's plant at Morris, Illinois. Some of the power companies have found it necessary to re-rack their pools or expand their pool capacity to accommodate the growing backlog of SURF. Since there are no reprocessing plants operating in the U.S., there is little commercial HLW or TRUW. Consideration is given to interim storage facilities of various waste types in Section 16.

17.1 SUMMARY

This section describes a hypothetical repository for disposal of HLW and TRUW and/or SURF from commercial light water reactor (LWR) nuclear power plants in the USA. A generic representative bedded salt repository located at a hypothetical representative site was selected to initially represent the disposal function in the nuclear fuel cycle until designs of other facilities are developed and published. The repository is primarily a Kaiser Engineers' conceptual design consisting of a total area of about 35 km² (8,590 acres) that includes a surface area above the mine of about 14.5 km² (3,600 acres) at an elevation of 1448 m (4748 ft) above sea level. The total site area includes a buffer zone of 1 km around the mining workings. The mine-level facility is located 625 m (2050 ft) directly below the surface facility and consists of 106 rooms (1219 m [4000 ft] long by 5.3 m [17.4 ft] wide by

5.8 m [19 ft]). The mine is designed to hold 204,000 SURF canisters or about 34,400 HLW canisters and about 12,400 remotely handled TRUW canisters or some combination of HLW, TRUW and SURF. This capacity should accommodate the amount of spent fuel (86,000 MgHM) or equivalent waste from generation of 2570 GWe/yr. The design life for construction and operation of the facility is 45 years, extending from site preparation to final closure of the mine and decommissioning of the surface facilities. There are 22 years of emplacement operations and an overlapping of 25 years when retrievability of the waste is maintained (continuing until 8 years after final emplacement).

The average number of canisters received per operating day at the repository is 37 of SURF or 6.25 of HLW, 2.25 of RH-TRUW and 9 of CH-TRUW. The processing rate of the facility is based on 250 days/year and three transporters on a two-shift operation. This rate could be increased by going to 365 days/year, three shifts, and four transporters. Maintenance, health physics and security operation will continue around the clock all year. The repository staff size will vary in number over its life reaching a maximum of 259 during the highest emplacement rate period (17 of these represent supervisory or professional staff).

The canisters arrive at the facility by truck or rail in shipping casks. The shipping casks pass through receiving inspection, where they are checked for external damage or contamination. Then they are moved to the waste handling building, where the canisters are removed, inspected, and cleaned and/or repaired as required. Each canister is loaded into a transfer cask, which is moved to a hoist, lowered from the surface level to the mine level, and loaded onto a diesel-powered transporter. The transporter moves each transfer cask to a predetermined hole in a predetermined room, removes the floor shield plate that covers the hole, lowers the canister from the cask into the hole, replaces the shield plate and covers the hole with salt. These activities are all basically performed semi-remotely or remotely.

Unloading of canisters from shipping casks and loading into transfer casks is accomplished dry in a hot cell facility located at the head of the waste handling shaft to the mine. The radwaste handling building is attached to the hot cell building. A receiving rail yard, truck yard, and service and administrative buildings are located nearby. A rather large area is allocated to the storage pile of excavated salt which will eventually be used in backfilling and closure operations. The excess is removed from the site.

Decontamination water is collected from the receiving area and hot cells and is processed through ion exchange resins to remove the radioactivity. The demineralized water is either directed to the retention basin prior to discharge to surface drainage or is incorporated into concrete in low-level waste (LLW) drums for shipment to a LLW repository. Gaseous effluents from the liquid radwaste, vented from shipping casks and leaking canisters, and off-gases from a solid waste incinerator are processed through the gaseous radwaste system to remove radioactivity and reduce concentrations to acceptable levels before venting to the atmosphere. Solid wastes such as prefilters, HEPA filters, spent ion exchange resin, charcoal absorbers, and contaminated

paper, clothes, etc., are burned in the radioactive waste incinerator. The ash residue is incorporated into concrete in LLW drums and shipped offsite. About 220 drums of LLW are generated for each year of repository operation.

Four shafts connect the mine and the surface facilities. Two shafts are used to ventilate the confinement side of the mine (i.e., the part of the mine already filled or being filled with waste). The intake ventilation shaft also contains a waste handling hoist. The ventilation exhaust shaft header contains radioactivity filtration trains which come on-stream when radioactivity is detected on the air. The second pair of shafts ventilate the development side of the mine. The intake shaft contains a hoist to handle men and materials traffic and the exhaust shaft hoist handles the excavated salt. The boundary between the confinement and mine development sides of the mine is continually altered as the disposal rooms are filled and new rooms are excavated. Leakage between the two sides is controlled to permit flow only from the mine development side to the confinement side.

The most important potential safety hazard in the facility is the presence of the radioactivity of the HLW, the TRUW and the SURF. Under normal conditions, the canisters containing the radioactivity are handled in hermetically sealed and shielded canisters and casks. Transfer of canisters from shipping to transfer casks is accomplished by remote handling devices in hot cells. Emplacement of the canisters in the salt bed is also remotely handled from shielded cabs of the mine-level transporters. Only small quantities of gaseous radioactive materials are expected to be released under controlled conditions from "as-received" leaking or partially damaged canisters. ^{85}Kr is normally vented in small quantities as a result of leakage from a number of defective canisters.

Potentials for accidental release of radioactivity include the possibility of dropping canisters or casks, e.g., 625 m (2050 ft) down a mine shaft, and in the possibilities of fire and flooding of the mine, incinerator explosion, and combustion of spilled fuels, etc. (see Pepping 1981).

Although site selection and grading are preventive factors, there is a remote potential for flooding the mine with surface water overflowing into the mine shafts. Seepage of water into the mine from overbearing aquifers, and in the extreme, the collapse of geologic strata above the mine level could also potentially lead to flooding the mine. Flooding, which is both hazardous to operating personnel and injurious to the salt bed, could potentially cause a nuclear criticality event which is normally eliminated by design. Water also could accelerate corrosion of waste canisters leading to the release of radionuclides earlier than planned.

There are a number of flammable materials of small volume in the mine, including diesel fuel, gasoline, transformer oil, wooden tunnel roofing, and perhaps mining explosives such as lost or strayed charges from earlier mining operations and shaft driving. Ventilation system-driven smoke from fire could be a threat to mine personnel. Fire warning and protection equipment are included and personnel evacuation procedures have been studied.

In the facility described herein, canisters are essentially unshielded. Alternatively, canisters have been designed that are self-shielded (WE/AESD 1981). (If canisters turn out to be thick walled to provide a long life in a corrosive environment, a significant amount of self-shielding would result). Alternative cooling periods to the 10-years used here for SURF and HLW could affect the thermal load for any facility designed and built. The potential of locating the repository in alternative geologic media is also discussed.

17.2 MAINLINE PROCESS DESCRIPTION

This section discusses the different types of waste that could be deposited in the repository; the key processes in the construction of the repository and the handling and emplacement of waste packages; a gross operating schedule in the repository, assuming two work shifts per day; and the waste receiving and handling schedules for two alternative fuel cycles: once-through fuel and reprocessing/recycle.

17.2.1 Waste Description

The representative repository disposes of spent unprocessed fuel assemblies (SURF) from commercial light water reactors, and/or vitrified high-level wastes and remote-handled transuranic waste (RH-TRUW) resulting from the reprocessing of spent nuclear fuel elements, and contact-handled transuranic waste (CH-TRUW) produced in fuel fabrication and reprocessing, power plant operation, and non-fuel cycle applications.

Spent fuel assemblies emit high levels of radiation in the form of beta particles, gamma rays, and neutrons. The latter two types can penetrate even dense materials and, therefore, require proper shielding to protect workers who handle the waste. The contents of spent fuels were given in Table 16.A.1. The spent fuel canisters are of carbon steel, about 36.8 cm (14.5 in.) in diameter and 4.7 m (15.3 ft) long. Each canister contains either one PWR fuel assembly (0.461 kgHM) or 2 BWR assemblies (0.378 kgHM).

High-level waste contains essentially all of what is in the spent fuel oxide except the volatile elements (iodine, tritium, krypton, xenon, carbon) and the recovered uranium and plutonium. The content of high-level waste canisters was given in Table 16.B.1.

Transuranics are elements heavier than uranium characterized by long half-lives and the emission of alpha radiation. Transuranic waste also contains small quantities of gamma- and beta-emitting radionuclides. The transuranic element of principal concern to repository operations is plutonium. RH-TRUW has a surface dose rate of >200 mR/h (DOE/ET-0028 1979). These wastes consist of all of the hardware that constitutes the fuel assembly except the fuel itself, and other wastes from fuel reprocessing. The RH-TRUW is in canisters that are 0.62 m (2 ft) in diameter by 3.1 m (10 ft) long. The radioactivity content of cladding hull hardware was given in Table 16.C.2; that for other RH-TRUW was given in Tables 16.C.3 and 16.C.4.

CH-TRUW has a surface dose rate of less than 200 mR/h. CH-TRUW received at the repository is assumed to be in solid and nondispersible form, and is

shipped in 210-l (55-gal) steel drums that are lined with heavy plastic. The radioactivity content of these drums was given in Tables 16.C.5 and 16.C.6.

17.2.2 Key Process Scheme

The repository described in this section was originally designed to handle SURF. However, the repository facilities are also capable of handling HLW and TRUW. Certain modifications may be required, such as adjusting surge storage cell capacity, waste emplacement rows due to thermal characteristics of emplaced canisters, storage capacity, etc. These modifications were applied conceptually in this report.

The repository is designed to have 14.45 km² (3576 acres) of disposal with areal thermal loading limited to 7500 kW/km² (30 kW/acre). On the average, the repository is expected to receive either SURF or HLW and TRU wastes with an equivalent heat generation rate of 3.7 MW/yr.

The conceptual design of the repository is based on two sources: Stearns-Roger Engineering Company (SRENCO) (Jones 1979) and Kaiser Engineers (Ritchie 1978), who were the design contractors for the National Waste Terminal Storage (NWTS) Repositories No. 1 and No. 2, respectively.

The process of emplacement described in this section identifies those features common to all disposal forms. Generally, the Kaiser Engineer design is used with some support from the SRENCO conceptual design and other sources. In the later sections which describe specific operations in the facility, those features unique to the disposal form being considered are identified.

Figure 17.1 shows a simplified flow diagram of the disposal processing, starting from the construction of the repository to final emplacement of the waste in the mine. Repository construction activities consist of the excavation of the mine, including disposal rooms and underground airways, as well as the construction of shafts to provide access to the mine. Once the repository has been completed to some initial functioning level, waste receipt and handling activities begin.

Upon receipt of the waste shipping cask, or other packaging (e.g., for CH-TRUW), inspections for contamination are performed on the cask exterior and interior, the canister exteriors, and the cask atmosphere. Necessary decontamination and cask and canister repairs are performed and the shipping casks are returned to service. These operations are performed in hot cell facilities. After removal from the shipping casks and, if necessary, decontamination and repair, the canisters are placed in a transfer cask. The transfer cask is shielded to make handling possible without high radiation exposure of operating personnel. The transfer cask containing the canister is then shipped to the main waste shaft and lowered to the mine level. From here, the transfer cask is transferred to a mine storage room. The waste canister is then lowered from the transfer cask into holes drilled into the salt floor of this room. After emplacement, a concrete plug is placed into the hole for shielding and the hole is covered with salt to allow personnel and transport vehicles to move freely over the emplacement.

FIGURE 7.1. Overall Process Flow Scheme for the Representative Repository Facility

Retrievability is postulated for the representative repository until 8 years after the final waste emplacement, during which time no backfilling will occur. After this period, however, storage rooms are backfilled with salt and sealed. Onsite generated low-level waste (LLW) is incinerated at the facility, immobilized in concrete, and placed in 210-liter (55-gal) drums. Such onsite-generated wastes are then shipped offsite to a LLW repository. Any waste onsite that has enough transuranic nuclide content to be declared TRUW is disposed of at the repository.

More details of some of the major activities in the representative repository facility process flow scheme are discussed below.

Mine Excavation

Mine excavation operations include the mining activities of the main airways, the manifolds, the emplacement rooms, and the storage hole drilling for SURF and HLW.

The main airways are driven by two methods. Blasting is used during initial Phase I construction. Machine excavation is utilized once the main airways in the shaft pillar are driven to a point where the continuous mining machine can complete the work. The main airways are 3.3 m (11 ft) high by 5.6 m (18 ft - 6 in.) wide.

Salt is excavated from the mine face to provide storage rooms. The rate of excavation corresponds to projected canister-receiving rates. All mine disposal rooms are mined in two passes with a continuous mining machine. The advance rates are calculated on multiples of 42.4 m (140 ft) per day per eight-hour shift, one mining unit, and 250 working days per year. A crosscut tunnel is excavated every 303 m (1,000 ft) along the 1212-m (4,000-ft) long room to connect the parallel storage rooms. These provide ventilation and alternate escape routes for personnel and for maintenance and support vehicles.

Shafts and Boreholes

Four mine shafts give access from the surface facilities to the mine and are located at 121.2 m (400 ft) intervals in a north-south line parallel to, and 606 m (2,000 ft) from, the west end of the mine workings. In order from south to north, these shafts are for 1) men and materials, 2) salt hoisting, 3) waste handling and 4) mine confinement ventilation exhaust.

The men and material shaft (Shaft No. 1) functions as the fresh air intake for the development side of the mine and enables the movement of air, men, materials and equipment between the surface and the shaft station at the mine working level.

The salt hoisting shaft (Shaft No. 2) is used for allowing access for initial mine development, exhausting air from the mine development area, hoisting salt at a maximum of 5700 Mg per day (6365 tons per day) and lowering salt for backfill operations.

The waste handling shaft (Shaft No. 3) functions as the main fresh air intake for the confinement side of the mine and has a compartment for hoisting SURF, HLW, and RH-TRUW canisters, and CH-TRUW drums from the waste handling building to the shaft station for further transport to disposal rooms.

The confinement ventilation exhaust shaft (Shaft No. 4) has the single function of exhausting spent air from the confinement side of the mine. The mine development and confinement sides of the mine are separated by an airlock and the confinement side is maintained at a small negative pressure relative to the mine development. Thus any leakage flows to the confinement side and prevents any radioactivity from being vented since the confinement side is exhausted through a filtration train at the head of Shaft 4 when radioactivity is in the air.

Some boreholes may be found in the salt formation. These holes may have been used to remove material samples for site selection studies, for salt mining, or may have been created from previous oil and gas drilling explorations in the area. Since salt is highly soluble in water, it is necessary that these holes are completely sealed before the repository starts receiving wastes for underground emplacement.

Waste Receipt and Handling

Waste receipt and handling facilities include the surface facilities that handle the incoming SURF, HLW, RH-TRUW and CH-TRUW for transfer to the underground facilities, and those underground operations that move canisters and drums to the storage rooms for burial in emplacement holes.

The functions of the surface waste handling facilities include:

- unloading of shipping casks and cargo carried by rail or truck
- transferring of canisters and drums to inspection stations
- decontaminating canisters and casks
- overpacking of canisters and drums, as required
- transferring canisters and drums to underground facilities
- treating and handling of onsite-generated wastes.

Underground operation covers the transfer, handling, and emplacement of SURF and/or waste canisters and CH-TRUW drums after processing these canisters in the receiving building. More detailed description of the surface facilities is given in Section 17.4.3.

Shipping casks arrive by rail and truck, as described in Section 19. Up to seven canisters, containing one PWR or two BWR assemblies each, may arrive in a single shipping cask, either by rail or truck. Up to thirteen HLW canisters or three RH-TRUW canisters may arrive in a rail shipping cask (Bechtel 1979). All canistered wastes entering the repository are aged at least 10 years from the time of removal from the reactor. Contact-handled TRUW drums are packed in TRUPACT shipping containers designed for fire and crash resistance. The CH-TRUW is assumed to have been properly packaged for disposal and is not treated further at the repository.

From the receiving gate, the shipment is directed through an inspection pit area for visual observation of the vehicle and shipping cask package. Any visible repairable damage that could create a problem in unloading is remedied. If there is evidence of abnormal conditions apart from road damage, the vehicle is dispatched immediately to a suspect storage area for detailed investigation. Shipments that pass inspection are sent either directly to the covered washdown area adjacent to the waste handling building or to the cask storage area for temporary holding.

The shipping cask exterior is tested for external contamination to determine if there is any possible leakage of the waste canisters. In any event, the cask must be transferred to a hot cell for dry unloading, washdown, decontamination, and possible repair before the decision to return it to service can be made.

Drums of CH-TRUW that are damaged or do not respond to decontamination procedures are placed into 319-liter (83-gal) overpack drums. The drum transfer fixture may carry up to five 210-liter (55-gal) drums or four 210-liter (55-gal) and one 319-liter (83-gal) drum. The design is similar to that of the transfer cask for spent fuel assemblies so that both may be handled with the same equipment. Shipping containers are cleaned and decontaminated if necessary and returned to service.

After testing, the railcar or truck trailer is moved by a cable car puller through the airlock door into position under the hot cell unloading port, where waste canisters are unloaded from the shipping cask and inspected for visible damage. Instrumentation in the hot cell also measures the radiation level and surface temperature of the canisters. A hoist is attached to the hoisting trunnions at one end of the cask and the cask rotated to the vertical while the other end is supported by the pivot trunnion. The hoist holds the cask in the vertical position while it is latched and locked into place at the hot cell unloading port.

After inspection, undamaged canisters are either placed into a transfer cask or put in interim storage in the canister storage rack. Damaged canisters are placed in overpacks before they are moved into the transfer cask loading port or put into interim storage, depending on availability of transfer casks.

Transfer Cask

Transfer casks (Figure 17.2) are used to transfer canisters from hot cells to storage holes in the mine. The spent fuel transfer casks are sealed and provided with neutron shielding (53 mm [2.1 in.] of cast organic polymer) and gamma shielding (156 mm [6.16 in.] of lead) to reduce radiation dose to 25 mR/h, or less, at the outer surface.

The RH-TRUW canister transfer cask provides a shielded container for movement of three canisters from the hot cell to the underground emplacement hole. Overpacked RH-TRUW canisters are accommodated by a separate, single-canister-overpack transfer cask due to infrequent need for handling overpacked canisters.

FIGURE 17.2. Onsite Transfer Cask for Spent Fuel and High-Level Waste

The spent fuel transfer cask weighs 21.3 Mg (23.5 ton) empty and 22.7 Mg (25 ton) loaded. Operators of the transfer cask carrier and the mine transporter are located in shielded cabs designed to reduce the dose rate to 2.0 mR/h. The cask is provided with two shuttle doors at one end and an interim electro-magnet suspended from the other. The magnet is served by two hoisting cables and one electric power and control cable. The hoisting cables are wound on an electrically driven winch, and the electric cable on a spring takeup drum. The winch and drum are contained in a sealed housing and mounted on the exterior of the cask. Automatically actuated mechanical locks will prevent inadvertent door opening. The waste hoist has two cages having a 2.5-Mg capacity each.

The SRENCO design specifications for a HLW transfer cask are as follows:

Mass: 30 Mg (33 ton) unloaded, 31 Mg (34 ton) loaded

Size: 3.9 m (12.7 ft) high x 1.2 m (3.9 ft) nominal diameter

Shielding Material: Depleted U and borated water equivalent shielding encased in steel liner. Shielding provides nominal contact dose rates of 1.5 mR/h along the outside circumference of the transfer cask and 10 mR/h at the top and bottom of the cask.

For the RH-TRUW transfer cask, the SRENCO design specifications are:

Mass: 27 Mg (29.7 ton) unloaded, 33.6 Mg (37 ton) loaded (transfer cask); 10 Mg (11 ton) unloaded plus additional ballast as needed for friction hoist operation.

Size: 3.8 m (12.5 ft) high x 1.1 m (3.6 ft) nominal diameter (overpack transfer cask)

Shielding Material: 10.2 cm (4 in.) lead and 5.1 cm (2 in.) borated water equivalent shielding encased in steel liners. The transfer cask provides shielding to limit the average contact dose rate to 2.5 mR/h.

The heavier weight of the HLW and RH-TRUW canister transfer casks as compared to the SURF transfer cask required an increased capacity of the waste hoist.

Waste Cage Loading/Unloading Interface

The waste cage loading/unloading operation is accomplished by means of a transfer cask mover (Figure 17.3). The machine is a modified form of underhung crane. A wheeled, powered carriage is mounted on a A-frame structure. Suspended from the carriage is a track upon which a hydraulically actuated saddle is mounted. Descending from the saddle is a pedestal with an electrically driven rotatable member. The saddle and track provide a transverse motion capability which permits the mover to service either of two

FIGURE 17.3. Transfer Cask Handling Equipment at Waste Cage

waste cages. The forks are designed to be raised or lowered independently. If an empty transfer cask is in the waste cage and the mover is empty, the operation includes an automated procedure in which the carriage advances toward a waste cage with its lowered forks entering the waste cage and stopping with the center of the saddle trunnion in line with the trunnion on the transfer cask. At a signal, the forks raise, the transfer cask trunnion latches release, the fork saddles engage the cask trunnions, the cask is lifted clear, a signal causes the carriage to withdraw, and the transfer cask is removed from the waste cage.

If a transfer cask carrier is waiting in position, the mover lowers its empty forks and continues its motion away from the cage until the saddle trunnions of the forks are under the trunnions of the transfer cask in the transfer cask carrier. The same cycle is used in removing the cask from the cage.

Waste Emplacement

Emplacement of SURF canisters is shown in Figure 17.4. The carbon steel SURF canisters are handled by electromagnetics. The disposal holes, designed for a 25-year retrievability period, have metal sleeve linings to keep the holes clear of sloughing salt. The holes are drilled with an auger to remove salt cuttings. The top part of the hole is countersunk to allow for placement of the concrete canister shield plug. In addition to the rotary drill, one load/haul/dump vehicle is provided to clear away cuttings from each vertical hole.

Each completed hole has a round steel hole cover to protect the hole and to maintain safety while the transporter and other equipment travel through the rooms and over the holes. Disposal holes in the salt mine are 0.5 m (20 in.) in diameter by 6.7 m (22 ft) deep, drilled within 1 degree of vertical, and counterbored 0.6 m (24 in.) in diameter by 1.5 m (5 ft) deep. The disposal hole is drilled to provide a 5-cm (2-in.) annular clearance to allow for closure of the hole around the sleeve. This clearance volume around the sleeve is vented to allow the escape of water vapor produced as brine inclusions migrate toward the heat-generating wastes.

Because there are expected to be few overpacked canisters, standard holes are drilled at 0.5 m (20 in.) i.d., and standard sleeves at 0.4 m (16 in.) o.d. are provided. On the infrequent occasions when an overpacked canister is to be stored, an oversize hole and sleeve will be provided. The end of the sleeve is brazed to a cast iron alignment ring which centers and suspends the sleeve portion from the counterbore in the storage hole. With the sleeve suspended from the top, it will not initially contact the corrosive salt. The ring design includes a tapered inside diameter that acts as a centering seat for the concrete shield plug and as a centering guide for the electromagnet grapple used for canister deposit and retrieval. The concrete plug and floor cover prevents loose salt from entering the steel sleeve before canister storage.

1 ft = 0.305 meters; 1 in. = 2.54×10^{-2} meters)

FIGURE 17.4. Spent Fuel Canister Emplacement in Geologic Repository

The shield plug is designed to reduce the radiation level at the top of the plug to less than 1.2 mR/h. The plug mates with its alignment ring, which is cast iron and brazed to the canister sleeve and serves as the streaming shield for the annular gap between the shield plug and the counterbored hole in the salt.

The flanged floor cover, with an annular rib sized to fit the storage hole, is fabricated from heavy plate. The flange of the cover is of sufficient diameter to permit the transporter and other vehicle traffic to pass over the cover without breakdown of the counterbore walls.

The HLW and RH-TRUW emplacement follows the same procedure described for spent fuel canisters. However, the size and depth of the holes are changed to accommodate changes in the dimensions of the HLW and RH-TRUW canisters. The different heat generation rates of these wastes also require changes in the hole patterns. Section 17.4.2, which discusses the mine layout, will cover these changes in more detail. It is also possible that waste canisters will be stainless steel, rather than mild steel containers, which cannot be handled magnetically. In this event, a mechanically-actuated grappling system can be used for handling HLW and RH-TRUW canisters.

Emplacement of CH-TRUW is accomplished using a drum transfer fixture that is designed to accommodate a stack of five drums. The drum transfer fixture is designed with the same handling aids as specified for the transfer cask, so that waste cage loading and unloading is conducted in the same manner for either item. The fixture is loaded into the waste cage and lowered into the mine just as though it were a transfer cask. At the mine level, the drums are taken from the fixture and palletized in a 2 x 3 pattern two drums high, with the layers separated by a metal plate. These pallets are placed in a separate storage room, two pallets high.

Retrieval

The protection sleeve in the emplacement hole ensures that the SURF canister remains free. The retrieval operation, if needed, will thus be the reverse of the emplacement operation for the great majority of the canisters. An empty transfer cask is removed from the waste shaft cage, loaded onto the transporter, and moved to a storage room. After the transporter has been positioned over a storage hole, the canister is raised into the transfer cask. The transporter returns the transfer cask to the waste hoisting shaft where the cask is raised to the waste handling building on the surface.

Retrieval of an abnormal canister (one that cannot be freed from the enclosing salt) will require that a salt overcore be cut around the emplacement and the entire salt-encapsulated emplacement be removed. A device to perform this operation is described in Stinebaugh (1979) for a waste emplacement of different dimensions than those assumed for this facility. It is assumed that a similar piece of equipment would be provided for this facility.

Backfilling

No backfilling will occur during the 25-year retrievability period. After this period, the decision may be made to begin backfilling the mine rooms. If this decision is made before the mine development phase is completed or if handling schedules should fall behind, salt from room development areas may be used as backfill rather than being hauled to the surface.

Ventilation

Underground facilities are ventilated with fresh air from the surface. There are two distinct ventilation systems to prevent air from the waste confinement areas from mixing with air ventilating the mine development area.

During initial mine development, only one system is in use. Flexible ventilation ducts are used during this period to accommodate variable work locations. Once the ventilation system is complete, air is directed down shafts No. 1 and 3 and split in parallel circuits to the various working places. The airflow is regulated through low resistance workings to maintain sufficient airflows throughout the mine and finally is exhausted up shafts No. 2 and 4. Air from the confinement side is filtered through high-efficiency particulate air (HEPA) filters before being discharged to the atmosphere.

Interface between the mine and confinement ventilation systems exists through the two access routes in the pillar area by way of multiple air-lock doors and through barricades in the mine storage area where minor leakage takes place. At all such points, mine air leaks into the confinement air system (rather than vice versa) because of the negative pressure differential from the mine ventilation to the confinement ventilation system.

Other important interfaces exist between the mine confinement ventilation intake and the canister hoisting compartment and between the canister hoisting compartment and the waste handling building. The canister hoisting compartment, which is a potential contamination area, is maintained at a negative pressure in relation to the mine confinement intake and the waste handling building.

Ventilation systems are constructed with redundant components and backup systems are available in the event of failure of a given fan. Emergency generators are also maintained for events resulting in the loss of offsite power.

17.2.3 Operating Schedule

The operating activities at the repository may be divided into five major categories:

- surface facility construction
- initial mine development
- storage room development
- waste emplacement
- backfill and decommissioning.

The gross schedule for operations in a SURF repository facility is shown in Figure 17.5. As shown, activities 1 and 2 and activities 3 and 4 will overlap.

The mine is designed to remain open for 25 years after initial waste emplacement to facilitate retrieval of the wastes, if such action should be desirable.

Mine development/operation is divided into three phases as shown in Figure 17.5. Phase I extends over the first five years of operation. During Phase I operation, 20,025 canisters of SURF will be emplaced. The retrievability period is measured from the end of this period. A 12-year Phase II operation begins after the first five years of emplacement. During this period, another 114,655 SURF canisters will be emplaced. Following this phase, a five-year Phase III operation, which requires further mine-drift development, occurs during which another 69,264 SURF canisters are emplaced. Decommissioning of the facility will begin 30 years after the first canisters are emplaced.

The representative repository has a maximum receiving rate of 15,000 canisters per year. Assuming 250 working days per year, a maximum of 60 canisters may be handled during a work day. These are handled by up to four transporters working 15 hours per day on the average. In the case of a SURF-only repository, the canisters consist of unreprocessed spent fuel, aged at least 10 years from reactor discharge. In a scenario that allows for spent fuel reprocessing, it is assumed that three types of canistered wastes could be stored in the representative repository: HLW, RH-TRUW and CH-TRUW. The HLW and RH-TRUW are in canisters, 0.3 m (1 ft) in diameter and 3.1 m (10 ft) long, and 0.6 m (2 ft) in diameter and 3.1 m (10 ft) long, respectively. It is assumed that 3.6 RH-TRUW canisters are produced for every 10 HLW canisters. CH-TRUW will also be deposited in the repository. It will be packaged in 210-L (55-gal) drums.

It is assumed that the repository could accommodate the amount of HLW that would result from the recycle of 86,000 Mg of heavy metal (MgHM) contained in PWR and BWR spent fuel assemblies. With 2.5 MgHM per canister of HLW, this would amount to 34,400 HLW canisters deposited in 22 years of repository operation (assuming only HLW is deposited). An equivalent of 12,384 RH-TRUW canisters is deposited over the same period. Over 50,000 drums of CH-TRUW will require disposal during the 22 years of repository emplacement operation.

Changes in emplacement procedures and mine layout to accommodate these waste forms are discussed further in Section 17.4.2. It will be seen that HLW canisters, even with their higher heating rate, would require fewer rooms than the spent fuel canisters. This releases more disposal areas for RH-TRUW.

FIGURE 17.5. Operating Schedule for the Deep Geologic Repository

17.3 DESCRIPTION OF EFFLUENT PROCESSING AND WASTE MANAGEMENT SYSTEMS

This section describes the treatment of radioactive and nonradioactive effluents and wastes generated at the representative repository facility.

17.3.1 Effluent Processing

The various processes in the repository generate solid, liquid, and gaseous materials which are deliberately discharged to the environment. Solid effluents consist mainly of excavated salt which releases radon, a naturally occurring radioactive gas. Radiation from stockpiled salt is estimated to be very low and should not be a hazard for the public. Liquid effluents are generally nonradioactive and are made up of uncontaminated water from certain processes. Gaseous effluents result from leaking fuel assemblies and from incineration of the solid low-level radwaste generated at the repository.

17.3.2 Waste Management

The operation of the repository facilities results in airborne releases of small amounts of radioactivity and also generates low-level radioactive waste in solid and liquid form. Solid radwaste consists of filters, ion-exchange resins, and contaminated general trash. Liquid radwaste comes from decontamination operations.

These wastes are managed in a similar manner to the low-level wastes routinely handled at other nuclear facilities. The gaseous waste is discharged, after filtration, through the stack of the waste-handling building. Contaminants in the liquid waste are extracted and converted to a solid form by immobilization in concrete. The remaining water is recycled for decontamination operations. The demineralized water is directed to the retention basin or may be used to make concrete. Combustible solid waste is incinerated, with the residue immobilized in concrete, packaged in steel drums. Drummed LLW is shipped to a LLW facility.

Four specific subsystems are used to collect and treat radioactive waste generated at the disposal facility:

- radioactive waste incinerator system for burning HEPA filters, ion-exchange resin beads, PVC bags, smear cloths, etc. (Figure 17.6)
- solid radwaste drumming system for immobilizing radwaste with concrete in drums (Figure 17.7)
- liquid radwaste system, which processes collected liquids containing radioactivity (Figure 17.8), and
- gaseous radwaste system, which treats process radioactive gases (Figure 17.9).

17.4 PLANT LAYOUT

This section provides a description of the representative site, the mine layout, the surface handling facilities, and the shafts and hoists. A conceptual sketch of the entire facility is shown in Figure 17.10.

FIGURE 7.6. Radioactive Waste Incinerator

FIGURE 17.7. Solid Radwaste Drumming System Using Concrete Immobilization

FIGURE 17.8. Liquid Radwaste System

FIGURE 17.9. Gaseous Radwaste and Effluent System

FIGURE 17.10. Artist's Conceptual Sketch of a Representative Geologic Repository

While the surface and mine level activities described herein were designed specifically for the disposal of SURF, they are also applicable to HLW, RH-TRUW and CH-TRUW. Features unique to these other waste forms are identified, as well as procedures required for handling CH-TRUW.

17.4.1 Site Description

The representative site chosen for storage of either SURF or HLW and TRUW includes a bedded salt geologic formation. However, other geologic media are under consideration: domed salt, basalt, granite, tuff and shale. Much of the designs developed for SURF-bedded salt and HLW-domed salt are expected to be applicable to repositories in other media.

The representative site is entirely hypothetical, yet its physiographic setting and geologic and hydrologic properties are analogous to several regions of the continental USA. The site includes a 14.5 km² (3576-acre) mine area located underneath and approximately within 34.74 km² (8590-acre) of outer controlled area. A detailed site description was provided in Section 6 of this report. The perimeter of the 14.5-km² inner controlled area is enclosed by a three-strand barbed wire fence. The central process area is enclosed by a double chain-link security fence fully instrumented with intrusion detection and alarm equipment. Within the central process area, the railroad cask storage yard and the substation will be enclosed within a single chain-link security fence. Further description of the surface facilities for the representative repository is provided in Section 17.4.4.

17.4.2 Mine Description

The creep closure rate of the rooms in the salt bed places some restrictions on the gross features of the mine. Creep closure accelerates as the temperature of the salt and the salt extraction ratio^(a) increase. Hence, to ensure that room closure is acceptable over the required 30 years, the mine design has the following characteristics:

- none of the salt exceeds a temperature of 135°C (275°F)
- when filled to capacity with 10-year-old wastes as described in Section 17.2.1, the areal thermal load is 7500 kW/km² (30 kW/acre)
- the extraction ratio is 8.1%(a).

The mine contains disposal rooms connected at right angles to branch corridors. Entrance to these rooms can be from either end. Specific mine layout requirements and waste emplacement for SURF and HLW and TRUW are discussed below. Thermal and physical constraints ultimately determine the geometry used to emplace canisters.

(a) The extraction ratio is the ratio of the excavated floor area to the area enclosed by the perimeter of the mine.

Arrangement for SURF Disposal

The master floor plan for the mine development is shown in Figure 17.11. The mine layout roughly resembles a large rectangular pan and its small handle. The shaft pillar (the "handle") provides access to the mine proper and encloses the four in-line shafts. This line of shafts is transverse to both the long axis of the mine (the "pan") and the five main airways connecting the shafts to the mine area. On either side of and at right angles to the central airways, two independent storage areas of equal size are developed. Two peripheral confinement return airways are also provided. The total mine area contains 106 rooms. The initial crosscut is widened to 11.3 m (37 ft) to form a manifold area. The manifold is used for miscellaneous storage, equipment passing, backfill activities, and ventilation control.

The elevation of the floor of the disposal rooms is 625 m (2050 ft) below the surface, with excavations within the shaft pillar ranging between 570 m (1870 ft) and 674 m (2210 ft) deep. A disposal room is 5.3 m (17-1/2 ft) wide by 5.8 m (19 ft) high by 1220 m (4000 ft) long. The center-to-center distance between rooms is 88 m (290 ft). There are 54 rooms in each half of the mine. Fifty-three rooms are used for disposal, and one remains open for ventilation.

The disposal rooms are crosscut every 305 m (1000 ft) for safety purposes, including alternate escape routes. Crosscuts also serve to route ventilation during canister emplacement and backfill operations. Each 305 m (1000 ft) room section (four to each disposal room) holds an average of 481 spent fuel canisters in two parallel rows spaced 1.7 m (5.5 ft) apart. Canister pitch in each row is 1.2 m (4 ft). These and other features are shown in Figure 17.12. Thus, each disposal room holds 1924 SURF canisters. Each spent fuel canister contains one PWR or two BWR fuel assemblies. Considering the thermal loading limits per room, about five rooms (1,924 canisters per room) of the 106-room underground disposal area are used per year for spent fuel emplacement.

The mine is designed to accommodate 203,944 spent fuel canisters containing 86,000 MgHM. Assuming an average burnup of 33,000 MWd/MgHM and a conversion efficiency of 33%, 86,000 MgHM is equivalent to 2567 GWe-yr.

Arrangement of Mine for HLW and RH-TRUW

The mine layout described for disposal of spent fuel assemblies must be modified to consider specific features of the HLW and RH-TRUW. Since the HLW canisters and transfer casks are 1.5 m (5 ft) shorter than those used for spent fuel, the height of the disposal room and emplacement gantry could be reduced by 1 m (3 ft). Thermal characteristics of the HLW require that each 1220 m (4,000 ft) room hold only one row of HLW canisters (instead of two parallel rows as for SURF) and that they be spaced further apart.

(1 ft = 0.305 meters; 1 acre = 4047 square meters)

FIGURE 17.11. Plan View of the Mine

(1 ft = 0.305 meters; 1 in. = 2.54×10^{-2} meters)

FIGURE 17.12. Plan View of Room, Pillar and Emplacement Geometry

With 2.5 MgHM equivalent and a heat generation rate of 2.3 kW in each waste canister, the total heat generated by HLW canisters (1563 canisters/year) is 3600 kW/year. With an areal thermal loading limit of 900 kW/room, four rooms are required per year, or 391 HLW canisters per room. This requires 88 rooms. At the end of 22 years all HLW canisters emplaced would be generating a maximum of 59,400 kW of heat.

The maximum heat generated by all RH-TRUW canisters (12,384) in 22 years is 850 kW. Thus, all RH-TRUW canisters are placed in one or two rooms. For the RH-TRUW disposal in one or two rooms, emplacement hole patterns are changed since the canister's diameter of 0.61 m (2 ft) is larger than those of either SURF or HLW canisters. But because of the lower heating emplacement rate, holes can be drilled closer to each other. Three or four parallel rows of holes could be drilled in each room section. The SRENCO conceptual design (Jones 1979) suggests stacking the canisters two-high (piggy-back) in the disposal wells, without steel sleeves or concrete plugs. Since disposal of this waste is not designed to be retrievable, the well is filled with crushed salt after the canister is in place.

The 106 disposal rooms in the representative repository, therefore, seem adequate to accommodate RH-TRUW with combinations of SURF and HLW for the specified capacity of this system.

Arrangement for CH-TRUW

Assuming a fuel cycle that utilizes reprocessing, another waste item to be stored at the representative repository is CH-TRUW generated from the reprocessing facilities. It is anticipated that over 50,000 drum-type canisters of CH-TRUW will require disposal during the 22 years of emplacement operations. The repository design assumes that all such waste is deposited in separate rooms. Drums are stacked two high and seven wide to minimize floor space and disposal requirements. This requires two disposal rooms for CH-TRUW. Since 90 out of 106 rooms are expected to be filled with HLW and RH-TRUW canisters, the 16 remaining rooms will be sufficient to dispose of CH-TRUW drums.

17.4.3 Shaft and Hoist Descriptions

Access between the mine and the surface facilities at the representative repository is provided by a series of four shafts equipped with hoists. The characteristics of these shafts were discussed in Section 17.2.2 and are summarized in Table 17.1.

The hoisting equipment utilized at the representative repository is located at the top of the head frame structures of Shafts No. 1, 2, and 3. (a) In addition to the main hoist in Shaft No. 1, there is a smaller service cage for personnel entry into the mine without disturbing production operations. The Koepe hoist concept (Cummings 1973) is anticipated to be used because its light compact design allows a lighter, less expensive head

(a) Shaft No. 4, which has the single function of venting exhaust from the confinement side of the mine, is not equipped with a normally-used hoist.

frame, less mass to accelerate and retard, and minimal rope wear due to bending. The salt skip system can be set for automatic hoisting after a signal is given from the door closing off the measuring pocket. The other hoisting systems are operated by signals from the stations being used. No men are thus required to operate any of the hoists. Specific characteristics of the hoisting systems at each shaft are presented below. Each of the three head frames is equipped with a monorail and 2.7-Mg (3-ton) hoist to service the shaft hoisting equipment.

TABLE 17.1. Characteristics of Representative Repository Shafts

<u>Shaft Number</u>	<u>I.D. (m)</u>	<u>Depth (m)</u>	<u>Equipment</u>	<u>Construction and Equipping Time Required (yr)</u>
1	6.7	654.6	27.5-Mg - cage, 909-kg service cage	2
2	3.6	615.2	Two 7.4-Mg bottom-dumping skips, clam-shell mucking hoist	2
3	4.8	648.5	Two 25-Mg cages and two inspection cages	2
4	3.2	593.9	Inspection cage and hoist	2

Shaft No. 1

- Men-and-material hoist: In the shaft there is one cage with a capacity of 27.4 Mg (30 ton) in balance with a counterweight. The hoist is rated at 670 kW (900 hp) and hoists at 4 m/s (800 ft/min).
- Service hoist: There is one cage with a capacity of 4.09 Mg (9,000 lb) in balance with a counterweight. The hoist is rated at 56 kW (75 hp) and hoists at 4 m/s (800 ft/min). The large cross-section of the men-and-material shaft allows large, assembled pieces of equipment to be handled in the shaft. This will save many manhours compared to the time otherwise required to disassemble and assemble equipment because of a smaller diameter shaft. For example, the deck is 2.9 m (9 ft 6 in.) x 5.8 m (19 ft), large enough to drive on a 2.4 m (7.9 ft) x 3.9 m (13 ft) D-6 dozer weighing 17.0 Mg (18.8 ton) and lower it into the mine.

Shaft No. 2

- Salt hoist: Two bottom-dumping skips, each with a capacity of 7.4 Mg (8.1 ton) are in balance in the empty condition. The weights of the empty skip at the surface plus the tail rope hanging below it amount to 50% of the gross weight of the loaded salt skip at the shaft station. The hoist is rated at 830 kW (1,114 hp) and hoists at 10 m/s (2,000 ft/min).
- Mucking hoist: A clamshell mucker to clean out the salt spilled to the sump is located in the shaft station.

Shaft No. 3

- Waste hoist: There are two cages in balance with a capacity of 24 Mg (27.5 ton) each. The downcast full transfer cask approximately balances the upcast empty cask. The hoist is rated at 1,280 kW (1,714 hp) and hoists at 10 m/s (2,000 ft/min).

17.4.4 Surface Facilities

This section describes the surface facilities for handling SURF. Figure 17.13 shows a plan view of the surface facilities. These facilities are also capable of handling HLW, RH-TRUW and CH-TRUW, although certain modifications may be required.

The structure and floor plan for the hypothetical waste handling facility are shown in Figures 17.14 and 17.15. The various activities in this building require heavy machinery, hoists, and transport equipment to handle shipping and transfer casks and canisters. Heavy shielding is required to reduce exposure rates to levels acceptable in working environments. Control of ventilation and water flow are necessary to prevent the possible spread of contamination. The facility is designed to withstand such natural phenomena as tornadoes and earthquakes. Fire prevention is also an important design consideration, and emergency power supplies must be maintained in the event of loss of offsite power. Specific subsystems of the facility are described below.

Decontamination Systems

Systems are provided to prepare decontamination solutions and to decontaminate building surfaces, hot cells, canisters, shipping cask interiors and exteriors, transfer cask interiors and exteriors, and interiors and exteriors of equipment and piping which may become contaminated.

To facilitate decontamination, special materials and coatings are used. Examples of these are the stainless steel walls, floors, and ceilings of the hot cells and epoxy- and phenoline-coated floors and walls in areas where contamination is possible.

FIGURE 17.13. Plan View of Surface Facilities for Geologic Repository

FIGURE 17.14. Orthographic View of Waste Handling Building

FIGURE 17.15. Key Plan for the Waste Handling Building

Waste Handling Process Cooling Water System

A two-loop cooling water system is provided for equipment which may contain radioactive material. Two 560 l/m (150 gpm), 180 kPa (60 ft of water total developed head) stainless steel, mechanically sealed horizontal centrifugal pumps (one standby) circulate chemically treated cooling water through welded, standard-weight, stainless steel piping to the equipment. The cooling water then passes through a plate heat exchanger, which also receives cooling water from a closed-circuit evaporative cooler.

Waste Handling Building Ventilation

Confinement and nonconfinement zones are ventilated by independent exhaust systems with independent controls. The building is operated at a negative pressure with relation to the atmosphere. Ventilation flows are controlled by pressure differential from areas of no contamination and low contamination potential to areas of successively higher contamination potential.

Three ventilation zones in the handling building are defined according to the contamination potential present. Each zone is ventilated by an independent system. The three systems are described as follows:

1. Primary confinement system: Those areas with highest contamination potential are maintained at the lowest pressure. Activities performed in those areas would be, for example, hot cell transfers, waste incineration, and waste immobilization.
2. Secondary confinement system: Those areas adjacent to primary confinement areas through which waste passes, in which contaminated equipment is cleaned and serviced, or in which contamination by leakage is possible are maintained at a pressure above that of the primary confinement area.
3. Nonconfinement system: Those areas used for administrative activities, building maintenance, and other nonhandling activities are maintained at the highest pressure.

CH-TRUW Handling

This system receives CH-TRUW, performs operations at surface level as required, and emplaces the CH-TRUW in the mine. The waste is received in shipping containers carried by trucks or railcars. After inspection and washdown, the vehicle with container(s) is moved into the receiving area of the waste handling building. CH-TRUW is received in containerized 210-liter (55-gal) steel drums. The CH-TRUW is assumed to have been immobilized in a concrete matrix by the CH-TRUW originator.

17.5 SAFETY-RELATED INFORMATION

This section discusses the pertinent safety-related information that can be used to help identify the risks associated with operation of the representative repository.

17.5.1 Operating and Maintenance Requirements

Table 17.2 gives estimates of requirements for operations personnel per day (for two shifts) for the Phase I and Phase II operations of the facility including room backfill and decommissioning. Waste receiving rates during Phase III operations are somewhat higher than those during Phase II and could require additional manpower, shifting of manpower from mine development to emplacement activities, adjustment of the receiving schedule, or some combination of these. The assumption of the existence of another facility during the latter years of operation makes the exact scheduling of activities flexible. In postulating scenarios leading to release of the radionuclides, the numbers of personnel involved in various activities may be important.

Waste handling and storage operations involve both contact and remote operations. Shipping casks are contact-handled until they are transferred to the hot cell for inspection and decontamination of canisters. Maintenance operations are primarily performed with crane-mounted tools and hand tools. A crane repair bay provides access to crane for general maintenance.

Equipment used in the hot cell is engineered to rugged, heavy-duty standards. All lubrication needs are supplied through the use of solid materials such as carbon-graphite, sintered bronze, etc., and oils and greases are specifically developed for use with radioactive materials. Compared to production rates in private industry, periods of activity and operating cycles will be so low that maintenance and repair will be a "hands-on" operation. When contact work is required within the cell, any canister in the cell is removed and put in transfer or shipping casks. Decontaminating sprays are expected to reduce any contamination to a level that will permit suitably attired personnel to perform needed tasks. Maintenance and repair is scheduled for nonoperating periods.

The various areas in the mine and waste handling facilities have decontaminable surfaces for ease of decontamination and cleaning. All HEPA filters are planned to be replaced annually. Maintenance of equipment and instrument control systems is performed periodically.

17.5.2 Input and Output of Major Materials

This section discusses the input and output of major materials in the repository. They include feeds (i.e., packaged wastes received for storage), effluents, onsite generated wastes and nonradioactive input materials.

Feeds

The facility receives an average of 3909 MgHM of spent fuel or its equivalent of vitrified HLW plus TRUW annually (over 22 years of receipts). Incoming waste is assumed to have decayed 10 years prior to shipment to the repository. The waste is intended to be permanently disposed of in the repository. Spent fuel or HLW may be retrieved during the first twenty-two years, after which the emplacement holes are backfilled with salt and the waste is stored permanently. The specific isotope inventories for SURF, HLW and TRUW were given in Sections 16A, 16B and 16C, respectively, of this report.

TABLE 17.2. SURF Personnel Requirements

	<u>Phase I</u>	<u>Phase II</u>	<u>Room Backfill and Decommissioning</u>
Waste Handling			
Supervisors	3	5	1
Support Staff	12	12	6
Technicians	9	14	7
Craftsmen	<u>21</u>	<u>43</u>	<u>12</u>
Subtotal	45	74	26
Mine Level (all craftsmen)			
Mine Development	113	97	0
Room Maintenance	8	8	0
Canister storage	20	80	0
Backfill	0	0	54
Demolition	0	0	36
Shaft Decommissioning	<u>0</u>	<u>0</u>	<u>120</u>
Subtotal	141	185	210
Total	186	259	236

Effluents

Effluents are either radioactive or nonradioactive. Radioactivity in effluents is from radionuclides released to the ventilation system by leaking canisters and fuel assemblies and through the stack of the incinerator during processing of solid low-level waste. Salt excavated from the mine and stockpiled at the surface releases radon, but the radioactivity is estimated to be very low. Nonradioactive effluents consist of 1) uncontaminated water from certain processes, 2) combustion products from burning diesel fuel during underground mining operations, and 3) combustion products from surface burning of coal. Nonradiological pollutants released to the environment during the repository's operational life are given in Table 17.3 for a salt disposal medium.

TABLE 17.3. Total Quantities of Effluents Released to the Atmosphere During Operation of Salt Repository(a)

<u>Effluent</u>	<u>Quantity</u>	
	<u>Annualized</u>	<u>Total</u>
Particulates, Mg	20	510
SO _x , Mg	550	12,000
CO, Mg	130	2,900
Hydrocarbons, Mg	50	1,000
NO _x , Mg	770	17,000
Heat, MW	5.8E7	1.3E9

(a) Taken from (DOE/ET-0029 1979).

Wastes

Combustible solid radwaste consisting of filters, ion exchange resin and general contaminated trash is incinerated. The ash is delivered to the LLW drumming system for immobilization in concrete. The scrubbed off-gases are routed to the gaseous radwaste system. Liquid radwaste arises from cask, canister, hot cell and general decontamination operations. The contaminated water is purified in the deionizer system and the cleaned effluent is discharged to surface drainage via the retention pond or returned to the decontamination water storage tank for retreatment. Gaseous radwaste is generated by controlled venting of casks and leaking canisters and by the incinerator off-gases. The purified gases are vented to the mine confinement exhaust system and exhausted from the stack. Scrubber liquid is drained to the liquid radwaste system. Condensate from the entrainment separator is drained to the LLW drumming system and incorporated into concrete. Non-TRU LLW generated at the repository during the handling and emplacement of wastes is treated and sent to a shallow ground burial site for disposal. Approximately 220 drums of this waste is generated each year of representative repository operation at the operating rates given earlier in this section.

Nonradioactive Input Materials

Other material utilized at the repository includes the ion exchange resin beads, acid and caustic solutions for resin bed regeneration, detergent for decontamination and cleaning of rooms and equipment, and chemical agents for fire suppression. Annual requirements of these materials are not available but it is expected that they would fall in the same range given for the ISFSF (Table 16.A.5).

17.5.3 Inventory of Major Hazardous Materials at Plant Locations Having Safety-Related Significance

The major hazardous materials other than the radioactive materials already identified are the gasoline and fuel oil needed for various operations of the facility. These items are summarized in Table 17.4. Number 2 diesel fuel is used by the transfer-cask transporter and other mine diesel equipment, locomotives, surface vehicles, standby generators, fire water pumps, and startup for the boilers. Gasoline fuel is needed for vehicle refueling. The gasoline fuel consumption is estimated to be 19 m³/month (5,000 gal/month). Average consumption of number 2 diesel fuel oil is estimated to be 136 m³/month (36,000 gal/month) for the operation of Phase I and Phase II facilities.

TABLE 17.4. Inventories of Hazardous Materials at Various Plant Locations

<u>Location/Material</u>	<u>Form</u>		<u>Quantity</u>	<u>Potential Hazard</u>
	<u>Chemical</u>	<u>Phase</u>		
Surface Storage Area				
- Excavated salt	NaCl	S	5775 Mg/day	Radioactive
- Coal	--	S	65 Mg/day	Combustible
Waste Receiving and Transfer Areas				
- Packaged Wastes	Many	S	15-50 canisters/day	Radioactive
- Radioactive effluents (leaks)	Many	G,L,S	NA	Radioactive
Waste Processing Area				
- Packaged Wastes	Many	S	0.9 drums/day	Radioactive (low-level)
- Effluents (leaks)	Many	G,L,S	NA	Radioactive
Storage Tank Areas (see Table 17.9)				
- Gasoline	Hydrocarbons	L	0.9 m ³ /day	Flammable
- Diesel fuel	Hydrocarbons	L	6.5 m ³ /day	Flammable

Gasoline fuel is stored in a buried storage tank which holds a 30-day inventory. Number 2 diesel fuel is stored in below-ground storage tanks at maintenance shops, at the locomotive maintenance area, the mine operations building, the standby generator building, the boiler house, the mine maintenance area, and the surface refueling facility. Table 17.5 provides a summary of individual fuel storage tank volumes and operational durations.

TABLE 17.5. Diesel Fuel Storage Tank Volumes
and Operational Durations

<u>Facility Name</u>	<u>Storage Volume (m³/gal)</u>	<u>Storage Duration (days)</u>
Surface vehicle refueling	75.7/20,000 (plus 75.7/20,000, Phase II)	30
Mine vehicle refueling	15/4,000 (mine level at 610 m)	30
Locomotive refueling	19/5,000	30
Standby generators fuel system	113.6/30,000 (two 56.8/15,000 tanks)	7
Standby generator man/material hoist fuel system	9.5/2,500	7
Boiler startup fuel system	37.9/10,000	1
Fire water pump houses No. 1 and No. 2 fuel system	1.9/500/pump house	1

Radioactive materials at the representative repository site include excavated salt at the site surface, radioactive effluents at the waste receiving and processing area, and the SURF canisters (1 PWR or 2 BWR assemblies per container) or HLW and TRUW packages, whose inventories are detailed in Tables 16.A.1, 16.B.1, and 16.C.2-6, respectively.

17.5.4 Unit Operations Involved

Unit operations involved for each step in the process flow scheme discussed in Section 17.2.2 are listed in Table 17.6.

TABLE 17.6. Unit Operations Involved for the
Repository Process Flow Scheme

<u>Process Step</u>	<u>Unit Operations</u>
Repository Construction	Solids materials handling Solids drilling/blasting Liquid materials handling (pumping)
Waste Handling and Storage	
Waste Receipt/Handling	Solids materials handling
Decontamination	Solids materials handling (packaging) Solids materials melting (welding) Liquid materials handling (pumping)
Waste Emplacement	Solids materials handling (conveyance) Solids drilling/blasting Solids materials burial
Backfilling	Solids materials handling (conveyance) Solids materials burial
Waste and Effluent Management	
Ventilation and Process Gas Treatment	Gas Pumping Solid-gas separation (filtration) Solid sorption (I ₂ adsorption)
Waste Treatment and Concentration	Liquids distillation (evaporation) Solid-liquid separation (filtration) Solid-liquid sorption (ion exchange) Liquids pumping, vapor condensation Solid materials size reduction (shredding, compaction) Incineration
Solid-Liquid Solidification	Cementing
Drum handling	Solids materials handling (packaging, conveyance)

17.5.5 Accident Driving Forces

This section identifies the driving forces that could cause accidents at the repository facility and the potential radionuclide risks from the occurrence of an accident. Table 17.7 lists the driving forces, their locations, and their characteristics.

TABLE 17.7. Accident Driving Forces at the Representative Repository Facility

<u>Accident Driving Force</u>	<u>Location</u>	<u>Characteristics</u>
Potential energy	Canister receiving and handling area	Air crashes, transporter collisions, cask drop
	Waste cage unloading area	Air crashes, transporter collisions
	Underground storage rooms	Transporter collisions, crane malfunctions
Nuclear reactivity	Canister handling area	$K_{eff} \leq 0.95$ by administrative control
	Underground storage rooms	"
	Surge storage cells	"
	Ventilation system	"
Chemical reactivity	Diesel fuel oil storage in mine	Flammable
	Hydrogen from vehicle batteries	Flammable
	Transformer oil storage	Flammable
	Water in mine salt	Corrosive
Temperature	Radwaste evaporator	50°C-150°C
	Incinerator system	>150°C
Pressures	Process cooling water system	0.1-1.0 MPa (1-10 atm)
	Incinerator system	NA

17.5.6 Containment and Shielding Systems

Containment barriers at the representative repository basically consist of the high-integrity waste form, the waste canister and overpack (i.e., steel sleeves, concrete hole liner, clay or sand packing), where applicable, and the naturally occurring barrier consisting of the properties of the repository rock and the surrounding rock materials and hydrologic systems. The lithologic system is generally considered to be the highly dominant feature for long-term isolation of the wastes from man's environment. Canister life-time is determined by the corrosion rate of the canister in the waste and

repository environment. Considerable study has been made of the corrosion resistance of candidate canister materials (Braithwaite 1978), and this work is continuing.

To retard radionuclide transport from the waste site in the event of canister failure, sorptive barriers have also been proposed. A candidate substance which acts as a sorptive barrier is a combination of 10% bentonite clay in 90% sand (DOE/EIS 1979). This mixture can significantly retard movement of the radionuclides from the area of emplacement in the presence of flowing water. The mixture would likely be placed around the waste canisters in the repository.

Low material-corrosion rates and sorptive barriers combine to delay the release of radionuclides in the event of intrusion of flowing groundwater in the future. The effects of such delays are being studied. A preliminary study indicates that a delay greater than a few hundred years contributes little to risk reduction (Cloninger 1979).

The shielding design of the waste handling facilities and equipment (including the hot cells, shipping casks, transfer casks, mine transporters, and shield plugs for the mine storage holes) is based on the neutron and gamma-ray source strengths associated with the waste form being emplaced. For HLW and spent fuel, a typical PWR fuel assembly having an exposure history of 33,000 MWd/MgU, an average specific power level of 30 MW/MgU, and a uranium loading of 0.461 MgU was assumed as the neutron and gamma-ray source requiring shielding. For RH-TRUW intermediate levels of radiation are involved, and shielding needs are designed accordingly. The CH-TRUW handling facilities and equipment, on the other hand, are designed to handle CH-TRUW containing sources of penetrating radiation equal to or less than 200 mR/h at the surface of the waste drum.

Facility shielding is designed so that the exposure to personnel is maintained as low as reasonably achievable (ALARA) given the magnitude of dose rates from radiation sources in conjunction with facility layout and time for operations, maintenance, surveillance, and testing. In addition, the radiation shielding is designed to ensure that the radiation levels specified in 10CFR20 are not exceeded during normal operation or during anticipated abnormal operational occurrences. Dose rates at various locations within the facility boundary have been categorized and access time is limited as shown in Table 17.8.

The following paragraphs describe the shielding design developed for the handling of spent fuel assemblies and for high-level wastes.

Hot Cell Shielding

The hot cell is shielded by 1.4 m (4.5 ft) of ordinary concrete to reduce the radiation level in operating galleries to less than 0.25 mR/h. The canister holding stations are located along both sides of the inner hot cell walls in a single row of ten stations each to hold a total of 20 canisters. This accommodates surge and unexpected blockage in the canister

handling and storage operations. For those areas in the spent fuel handling facilities requiring only periodic access (such as the service gallery, the cask unloading area, and the crane canyon area) radiation dose rates are between 0.25 and 2.5 mR/h.

TABLE 17.8. Facility Zones and Access Times for the Representative Repository

<u>Zone and Access Description</u>	<u>Dose Rate (mR/h)</u>
Zone 1: Continuous unlimited access	0.25
Zone 2: Controlled unlimited access for 40 h/week	0.25 - 2.5
Zone 3: Limited access between 10 and 40 h/week	2.5 - 10
Zone 4: Controlled limited access between 1 and 10 h/week	10 - 100
Zone 5: Normally inaccessible	>100

Canister Transfer Cask Shielding

The canister transfer cask has a photon shield made of 15.6-cm (6.16-in.) thick lead and a neutron shield made of cask organic polymer material, 5.31 cm (2.09 in.) thick. The cask o.d. is limited to 0.9 m (3 ft) by the waste cage and shaft dimensions. The resulting dose rate was calculated to be 35 mR/h at the surface for a 3-year-old PWR spent fuel canister (although the normal age is planned to be at least 10 years). Since the operators of the transfer cask carrier and the mine transporter are located in shielded cabs designed to reduce the dose rate to 2.0 mR/h or less, no overexposure of personnel is anticipated for this design.

Shipping Cask Shielding

The shielding configuration chosen for the shipping casks that transport canistered wastes to this facility was discussed in Section 16.

Shielding Design of Mine Transporter

Normal transporter operations such as loading of the transfer cask onto the transporter, alignment with the storage hole, shield plug placement or removal, and canister storing, etc., are all remotely performed by a transporter operator located in a shielded cab. The transporter is designed to limit radiation exposure during the canister emplacement operation when the canister is partially or completely out of the transfer cask. The operator's compartment is shielded by 4.3 cm (1.7 in.) of lead or equivalent shielding material to protect the operator from the transfer cask that contains the spent fuel canister and from the canister storage operations. The operator will not receive a dose rate in excess of 2.0 mR/h.

After the transporter reaches the storage hole and cask alignment with the storage hole is completed, the floor shield plate is lowered to cover the floor immediately around the hole. The floor shield plate consists of an 8-cm (3-in.) thick depleted uranium slab. During the canister emplacement operation, the shield plate reduces the radiation level around the transporter periphery to less than 10 mR/h. The presence of this floor shield plate allows personnel access around the transporter, if required. The shield plug lifter housing is especially designed to minimize radiation streaming during the shield plug placement operation.

Shield Plug Design and Storage Hole Spacing

The shield plug has dimensions of 0.5 (20 in) o.d. and 1.1 m (44 in.) long and is made of ordinary concrete with a cast-iron bottom. The shield plug is designed to reduce the radiation level at the top of the plug to less than 1.2 mR/h.

A large savings in the cost of mining can be realized by placing the disposal holes as close together as possible. This results in 1.2-m (4-ft) center-to-center storage spacing for this design. This spacing leaves only 0.6 to 0.7 m (24 to 28 in.) of salt web thickness between the 0.6-m (25-in.) i.d. top counter-bore sections, and 0.7 m (28 in.) of salt between 0.5-m (20-in.) i.d. lower canister hole sections, respectively. The 0.6 to 0.7 m (24 to 28 in.) of remaining salt acts as a shield between a loaded hole and the adjacent empty hole. The thickness of salt, however, is not sufficient to shield the radiation penetrating into the adjacent empty hole, since this radiation will scatter and contribute to the radiation level in the mine room. The radiation level in the empty hole opposite the stored canister midplane is estimated to be approximately 3000 mR/h and the radiation level at the mine floor level without the shield plug is approximately 150 mR/h. Therefore, a shield plug is to be placed temporarily in the empty hole adjacent to the loaded hole to limit the radiation level in the storage room.

Criticality Safety and Storage Hole Spacing

The effective neutron multiplication factor (K_{eff}) for two infinite rows of fuel assemblies stored in the salt at different hole spacing has been investigated by Kaiser Engineers (Jones 1978) to determine the sensitivity of K_{eff} to fuel assembly separation distance. The 1.2-m (4-ft) spacing in the same row and 1.7-m (5.5-ft) spacing between two rows, as selected for the design, gives a K_{eff} value of only 0.162; therefore, no criticality safety concern is anticipated. Even when placing the PWR canisters side by side in one row, it is estimated that the K_{eff} value is considered to be due principally to a relatively large thermal neutron absorption and scattering cross-section associated with chlorine, i.e., $3.4E-23 \text{ cm}^2$ and $1.6E-23 \text{ cm}^2$, respectively. The fuel assembly was modeled as an equivalent cylinder surrounded by salt having the density of 2.16 g/cm^3 and 0.1 weight-percent moisture.

17.5.7 Relative Degree of Hands-On Operation

This section identifies the relative degree of hands-on operation required for each element of the mainline process scheme. The approximate frequency and types of required handling and the amount of direct contact required for each unit operation is shown in Table 17.9 for normal operations and for maintenance requirements.

17.5.8 Fire and Flooding Protection

Mine fire protection for the repository complies with MSHA standards and consists of four basic types:

- a mobile fire protection unit
- built-in vehicle fire protection system
- belt drive fire protection units
- portable fire extinguishers.

With the exception of potable water, a basic criterion of the repository design is that no waste is allowed in the mine. Therefore, the mine fire protection system is limited to the use of dry chemicals. Waste handling building fire protection requirements are summarized in Table 17.10.

There is some potential for flooding of the mine from unplanned surface water overflow into the mine shaft. The mine shafts are also expected to penetrate water-bearing strata and must be sealed to prevent water from entering the mine. Condensation of moisture from incoming humid ventilation air is also a potential source of water. Seepage of water into the mine from overbearing aquifers and, in the extreme, the collapse of strata above the mine level could potentially lead to flooding of the mine. Flooding of the mine would be both hazardous to the operating personnel and injurious to the salt bed. Flooding during the operating phase of the repository could accelerate corrosion of the waste canisters and lead to some failure of barriers to radionuclide release into the normal access areas of the mine.

17.6 ALTERNATIVE PROCESS SCHEMES FOR GEOLOGIC WASTE REPOSITORIES

The exact scheme for disposal of spent fuel or high-level radioactive wastes in a geologic repository has not yet been determined. Many technological and political factors will enter into the final design of a repository system. This section contains brief descriptions of some of the alternative techniques or processes that may be utilized in the representative geologic repository.

17.6.1 Alternate Geologic Media

The effects of other geologic media such as basalt, granite, tuff and shale on safe and effective waste isolation have been evaluated previously. The important characteristics of these media from the standpoint of repository requirements are summarized below (Bechtel 1979).

TABLE 17.9. Relative Degree of Hands-On Operations
in Waste and Spent Fuel Disposal

<u>Process Element</u>	<u>Operations Required</u>	<u>Type of^(a) Contact</u>	<u>Relative Frequency During this Operation</u>
Repository Construction	Solids materials handling	Semi-remote	Continual
	Solids Drilling/ blasting	Semi-remote/ Direct	Continual
Waste Handling and Storage	Waste receipt/ handling	Semi-remote/ remote	Continual
	Decontamination	Semi-remote/ remote	Continual
	Waste Emplace- ment	Semi-remote	Continual
	Backfilling	Semi-remote/ Direct	Continual
Waste and Effluent Management	Ventilation	Remote/Contact	Continuous
	Water treatment (filtration/ sorption)	Remote	Continuous
	Solids compaction	Semi-remote/ remote	Continual
	Cementing	Semi-remote/ remote	Continual
	Solids materials handling	Semi-remote/ Direct	Continual
Maintenance	Welding	Semi-remote/ Direct	Periodic
	Parts Replacement	Semi-remote/ Direct	Periodic
	Inspections	Semi-remote/ remote Direct	Periodic

(a) Types of contact are defined as follows:

- Direct = Immediate hands-on contact with the material by personnel (with personnel wearing protective clothing).
- Semi-Remote = Use of major piece of equipment for operation; personnel required to directly operate equipment, but do not directly contact material.
- Remote = Personnel are far removed from process step; operation of required equipment takes place from a remote control station.

TABLE 17.10. Waste Handling Building Fire Protection Requirements

Facility	Fire Control Systems	Hazard Classification	Fire Demand Sprinkler and Hose	Comments
Waste handling building	Water sprinkler Halon gas--glove box Standpipe and hose stations--in confinement areas Smoke detectors Temperature detectors (Halon gas system and air intake and head frame) Dry chemical agents (ground floor near entrance to waste shaft and at air intake and head frame)	XI--Exhaust filter area incinerator OII--Cask unload and preparation LLW receiving and handling Mechanical/electrical rooms Office and change areas	0.13 m ³ /sec (2000 gpm)	Building important to operations and protect public from radioactive release
Confinement exhaust building	Water sprinklers Standpipe hose station Smoke detectors	XII--HEPA filter banks and storage OII--Mechanical and maintenance areas	0.3 m ³ /sec (2000 gpm)	Building important to operations and for public safety
Mine exhaust building	Water sprinklers Smoke detectors Dry chemical agents (salt loadout and head frame) Temperature detectors (salt loadout and head frame)	OII--Exhaust fans	0.02 m ³ /sec (1200 gpm)	Smoke detectors for early alert in unmanned building
Standby generator building	Water sprinklers	OII--Entire building	0.03 m ³ /sec (1200 gpm)	Building important to safety and maintenance of confinement systems
Mine operations building	Water sprinklers Smoke detectors Dry chemical agents (ground floor at mine shaft head frame) Temperature detectors (ground floor at mine shaft head frame)	OII--Entire building	0.03 m ³ /sec (1200 gpm)	Building important to operations and safety

Basalt is considered a favorable geologic medium for spent fuel waste disposal. It shows a very high confining capacity because of its low porosity, low permeability, high mechanical strength, and tolerance of high thermal loading.

Granite is considered a potential geologic medium because it has the following characteristics:

- very good strength and structural stability
- very low water content
- good sorptive properties
- good thermal properties
- very low permeability.

While granite is sensitive to fractures that affect permeability, it would be very difficult to locate a repository in any rock mass without fractures. Construction and operating costs are higher in granite because of the characteristics of crystalline homogenized rock.

Shale is the product of compaction and cementation of mud. It has positive attributes such as low permeability, good sorptive characteristics, and low solubility in water. However, thermally-induced changes at temperatures above 100°C increase the permeability and porosity of shale. This tends to increase radionuclide transport rates. Furthermore, inhomogeneities in shale rock affect its structural characteristics and present mining difficulties.

Tuff is of interest for use as an isolation medium for high heat producing wastes because it provides highly sorptive minerals and suitable thermomechanical properties. Furthermore, tuff is widespread in areas that offer long and deep groundwater flow paths. More information on the occurrence and geologic/hydrologic setting of tuff may be found in Tyler (1979). Studies are currently being pursued to determine the general characteristics of repositories in tuff as part of the Nevada Nuclear Waste Storage Investigations Project.

Table 17.11 lists the typical media properties and qualitatively classifies these properties relative to waste isolation. These properties affect repository capacity, spent fuel placement geometry, rate of migration of radionuclides, repository temperature, and drilling methods. The primary impacts of different geologic media on safety-related characteristics of the representative repository during mining and waste emplacement are in the mining methods and amount of ground support required. During long-term disposal, the basic need is a high integrity of the containment barrier provided by the host rock. Granites, basalts and tuffs exhibit high compressive strength and usually a relatively competent rock structure. Shales are subject to slacking (due to cycles of wetting and drying) and require more underground wall support. Salt exhibits some plasticity under stress and at high temperatures, and may require extensive underground support systems.

TABLE 17.11. Typical Properties of Geologic Media

Item	Alternative Geologic Media				
	Salt	Granite	Shale	Basalt	Tuff
Rock mechanical strength	Poor	Very Good	Good	Very Good	Very Good
Porosity	Very Good	Very Good	Poor	Very Good	Good
Permeability	Very Good	Very Good	Very Good	Very Good	Good
Solubility	Poor	Neg.	Neg.	Neg.	Good
Thermal conductivity	Very Good	Good	Good	Poor	NA
Heat capacity	Very Good	Good	Good	Good	Good
Chemical properties	Vary greatly; range of possible chemical reactions between wastes and rock is great				

In regard to the containment characteristics of the rock, shale, granite, tuff, and basalt are all relatively impermeable. Although salt is also impermeable, it is highly soluble in water. Thus, any mechanism that would cause water to reach the salt rock could potentially breach the waste containment barrier.

The heat released from emplaced waste and its induced thermal stresses could have a significant effect on the design of the repository. The thermal conductivity, mass density, and heat capacity of the host rock determine the rate of heat dissipation throughout the formation. It has been determined that the thermally induced stresses in hard rocks like granite, basalt and tuff are small compared to the strength of the rock in the pillars. Since shale and salt are generally weaker than these rocks, the combined stresses due to heat and gravity may require additional supports, increased spacing between canisters, etc.

17.6.2. Self-Shielded Canisters

The representative repository design has assumed that all waste forms are contained in stainless steel or carbon steel canisters prior to emplacement into vertical boreholes in the mine, which is 625 m (2,050 ft) below the surface. This design requires that all waste handling operations before and during emplacements use shielded or remotely controlled equipment.

A self-shielded package design (WE/AESD 1981) provides each waste package with its own shielding. Such design allows operations to be performed with little or no additional shielding, thereby eliminating the need for an elaborate shielded waste handling facility, although a smaller shielded facility will still probably be required to handle damaged drums or other abnormal conditions. It also eliminates the need to have individual emplacement holes for each canister. The self-shielded canister could be placed horizontally in the floor of the mine with relatively low height and

subsequently backfilled. The self-shielded canister design employs a primary containment barrier of a cast steel overpack of sufficient thickness to limit the radiation at the surface of the package, thus allowing limited hands-on operation. The waste form is conceived to be placed into the casting at the waste generating facility and a top shield plug is welded into place to form a totally leak-tight barrier.

Compared to the emplacement hole design, the self-shielded design could potentially offer the following safety-related advantages:

- reduces the need for shielded hot cell and surface storage facilities
- simplifies package emplacement operation and eliminates the need to drill emplacement holes in the mine floor
- minimizes the handling of unshielded material and recovery from accidents may be simplified
- reduces mining efforts because of having smaller mine tunnels
- eliminates concerns regarding the effects of radiolysis of groundwater and radiation effects on the host rock and tunnel backfill materials
- simplifies retrieval of waste packages.

17.6.3 Reduced Thermal Loading

A significant factor in the isolation of radioactive waste is the heat the waste generates. The heat flows from the waste through the canisters and emplacement hole sleeve, into the host rock formation and its surroundings, and eventually out into the atmosphere. The heat can have definite impacts on:

- integrity and retrievability of the waste canisters
- stability of room and pillar
- integrity of the waste form over long periods of time
- integrity of the host rock and surrounding formations.

Thermal impacts on the repository can be minimized by limiting the thermal loading and, therefore, limiting temperatures in the repository. Reduced thermal loadings are accomplished by emplacing fewer canisters per disposal room than are presently considered in the design. This would mean either having a larger mine area or having more than one repository to accommodate all waste canisters that are scheduled for permanent disposal.

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18.0 LOW-LEVEL AND INTERMEDIATE-LEVEL WASTE DISPOSAL

The representative facility selected for low-level and intermediate-level waste disposal is representative of current technology and is primarily modeled after the shallow land waste burial ground developed by the NRC to evaluate environmental impacts associated with land disposal of radioactive wastes (NRC 1981). The facility is to dispose of nontransuranic wastes with low- or intermediate-levels of radioactivity that are acceptable for shallow land burial. The representative facility is assumed to be located on the mid-continental site as described in Section 7. It is estimated that at least four such facilities will be required for low level waste disposal to the year 2000 for the United States.

The overall characteristics of the representative shallow land waste burial ground facility (SLWBGF) in this study include the following:

- The facility receives a variety of low- and intermediate-level wastes from nuclear, industrial, and institutional facilities.
- The facility receives a waste volume of 50,000 m³/yr for 20 years.
- The vast majority of wastes are placed in trenches as received. Processing consists only of repackaging wastes not conforming to standards, and equipment decontamination.
- The facility is assumed to operate 8 h/d, 250 d/yr, being closed weekends and holidays.

18.1 SUMMARY

The representative SLWBGF is designed primarily to accept wastes transported to the facility for immediate burial. Processing, which consists primarily of overpacked, failed containers or modest decontamination, is only expected for a very small fraction of the wastes received. An assumed volume of 50,000 m³/year with a total capacity of 1,000,000 m³ gives the facility a 20-year lifetime.

Facility operations consist primarily of inspection, storage, and disposal. The vast majority of shipments are inspected and transported in the same vehicle to the burial site for immediate off-loading and placement in trenches.

No large storage facilities exist at this site. Packages from common carriers may be stored only temporarily (e.g., overnight). Most shipments arriving by private carrier are temporarily stored on the vehicle if necessary.

Processing at the facility consists primarily of repackaging unacceptable shipments, and routine decontamination of equipment and personnel, and minimum processing of liquid decontaminating agents to packaged solids for onsite burial. This is carried out in the waste activities building. Details of the decontamination processes are not fully developed at

this time. Potential safety hazards include high temperature and pressure in liquid evaporators, and high mechanical pressure in compactors. Radiation exposure in package handling is expected to be minimal. The primary hazards center on the normal industrial hazards of lifting and conveying packages. Operation of heavy equipment including cranes and earth moving equipment for trenching and backfill operations also experiences routine industrial hazards. Effluent discharges of radionuclides from evaporator stacks or waste water are low, with any significant concentration of radionuclides being packaged and buried at the site. It is estimated that 1% of daily receipts in 2m³/d will require repackaging for burial.

The facility employs 70 persons. It occupies an area of 0.6 km² (148 acres) with 0.35 km² committed to waste disposal, 0.21 km² to operational and buffer zones, and 0.04 km² to administrative areas.

Waste volumes and radionuclide contents are based on projections of current waste streams from nuclear, industrial, and institutional sources.

18.2. MAINLINE PROCESS DESCRIPTION

The primary functions of the SLWBGF are to receive and dispose of low- and intermediate-level wastes that are packaged and shipped from offsite. The vast bulk of the wastes received are expected to move immediately to the trenches for disposal. Packages not acceptable may be repackaged and disposed of or returned to the shipper. The steps are described below and shown graphically in Figure 18.1.

18.2.1 Routine Waste Disposal

The routine functions of waste disposal at the SLWBGF consist of receipt and inspection, storage, and disposal, as shown in Figure 18.1. Arriving shipments of waste are first inspected for compliance with applicable state and federal regulations and waste acceptance criteria established as conditions in the facility operating license. Shipments found in compliance with regulations proceed to the disposal area. Off-loading from the vehicle occurs at the disposal area, where the wastes are placed immediately in the burial trench. Little or no processing is expected for the bulk of the wastes entering the facility.

In general practice, the wastes received at the site are disposed of within one to a few days. Wastes are stored temporarily, if needed, in the transport vehicle (generally) it arrived in. However, a small designated storage area is provided for waste packages which will require temporary storage.

An added storage requirement exists for wastes containing special nuclear material. License conditions require that any single shipment of special nuclear material be stored at least 3.7 meters (12 feet) from any other package containing special nuclear material. At this representative facility, special nuclear material is limited to uranium-223 and -235; only quantities of plutonium less than that considered to be transuranic waste are accepted.

FIGURE 18.1. Overall Process Flow for Shallow Land Waste Burial Ground Facility

Waste packages are then transported to the burial site by the cross-country vehicle. Upon unloading, waste is randomly emplaced in the trench using cranes and forklifts, and backfilled with dirt removed during trench excavation. Random waste emplacement results in a trench-volume-use efficiency of about 50%. License conditions prohibit uncovered waste from extending more than 100 feet beyond the backfilled portion of the trench, so that backfilling is done continually. License conditions also require that backfill operations commence immediately if radiation readings greater than 100 mR/h are recorded at the trench boundary, and continue until radiation levels are reduced below 100 mR/h. License conditions prohibit waste packages from being placed or left standing in water, so waste disposal commences at the high end of the trench and works down towards the lower end. Rainwater falling within the open trench and contacting the uncovered waste packages drains away to the lower end of the trench. Rainwater collecting in the lower end of the trench is then removed as necessary and treated by such methods as solar evaporation or solidification.

Waste is emplaced to within one meter of the top of the trench. Earthen fill is then backfilled into the trench. An additional one meter thick cap composed of soil is then placed upon the backfill and is mounded. No special compaction is performed on the backfill and cap other than that provided by trucks and heavy earth-moving equipment driven over the top of the cap. The cap is then covered with natural overburden material as necessary to provide good drainage characteristics according to the final contours planned for the site surface. The overburden is then reseeded to promote growth of a shallow-rooted grass cover.

Following trench capping, the disposal trenches are each marked with a monument which is inscribed with the following information:

- a trench identification number
- total trench activity of radioactive material in curies, excluding source and special nuclear material; mass of source material in kilograms; and mass of special nuclear material in grams
- date of completion of waste disposal into the trench
- Volume of waste in the trench.

In addition, each of the four top corners of the disposal trench is marked with a marker stone.

Beyond the disposal requirements discussed above, an additional requirement exists for disposal of wastes containing special nuclear material. License conditions require that each package of waste containing special nuclear material be disposed of in such a manner as to have a minimum of twenty centimeters (eight inches) of earth (or wastes not containing special nuclear material) in all directions from any other package containing special nuclear material.

During waste handling and disposal, operations are monitored to ensure radiation safety. After the transport vehicle is unloaded, it is again surveyed for contamination and decontaminated, as necessary, prior to leaving the restricted area.

18.2.2 Off-Normal Waste Handling

During receiving and inspection, a few shipments are assumed to be found that are not in compliance with the appropriate standards and regulations the facility operates under. Some of these shipments are returned immediately to the shipper, others may be off-loaded and moved to the waste activities building for intermediate processing. They then may be returned to the shipper or accepted for burial.

Packages moved to the waste activities building consist primarily of those found to be damaged or leaking. Depending upon license conditions, damaged or leaking waste containers may be overpacked or repackaged, and either accepted for disposal or returned to the sender. Free-standing liquids detected are absorbed and solidified on an appropriate medium (molecular sieve, etc.). Output from the waste activities building consists of packages suitable for disposal or shipment back to the sender.

The remaining function of the waste activities building is decontamination. The decontamination of equipment, (vehicles, etc.) materials, and personnel is carried out in this building. Output from these functions is again waste in suitable packages for immediate disposal.

18.3 DESCRIPTION OF EFFLUENT PROCESSING AND WASTE MANAGEMENT SYSTEMS

The main process flow at this facility consists of off-loading waste packages as they are received for immediate placement in the burial trenches. As such, no effluents other than dust are expected for these operations, and no effluent processing is provided for at that point. Activities in the waste activities building, however, involve some effluent control and result in a small amount of airborne and liquid releases to the environment.

18.3.1 Airborne Effluent Processing

Airborne effluents result from processes in the waste activities building, and at the trench sites themselves.

In the waste activities building, trace quantities of volatile gases and water vapor can be expected from the filtered evaporator stack. Being part of a routine decontamination system, handling only trace inputs, airborne releases are expected to be extremely low. Building atmosphere is also routinely exhausted through ventilation ducts, and is expected to have naturally-occurring amounts of pollutants.

At the disposal site, earth moving and vehicle moving operations are sources of airborne dust and exhaust emissions from heavy equipment. At times when fugitive dust from trenching operations or overburden could be a problem,

it is controlled by water sprays. No control beyond original manufacturer's equipment is expected for equipment emissions.

18.3.2 Liquid Effluent Processing

Decontamination processes in the waste activities building include liquid washing of equipment, personnel, and vehicles which become contaminated during waste handling activities in the building, or while driving in the disposal area. Liquids from washing are of low activity. They are collected and stored in a holding tank for monitoring. If radioactivity levels are acceptable, the liquid may be discharged to the sewer system with no further processing other than passage through a filter.

The liquid effluent system is shown in Figure 18.2. It has the capability of further waste processing through an evaporator if necessary. Concentrated wastes are solidified as described below.

Liquid effluents at the trench sites consists only of rain water or melted snow. License conditions prohibit placing packages in standing water. Rain water falling within an open trench or on packages is drained away to the lower end of the trench where it is collected, removed, and treated by evaporation as necessary. No untreated liquid off-site releases are expected from water that gets into the trenches. Estimates of liquid effluent control performance are not available.

18.3.3 Solid Waste Management

Concentrated liquids generated in the waste activities building are pumped from the trenches and are stored in a separate holding tank, as shown in Figure 18.3. They are processed in batch mode in a cement solidification process, then placed in drums suitable for immediate burial at the site. Solidification with other media such as vermiculite, molecular sieve, or bitumen could also be used.

Other solid wastes generated as routine operation (paper, gloves, rags, etc.) are compressed and baled, then packaged for immediate disposal at the site.

18.4 PLANT LAYOUT

The description of the representative shallow land waste burial ground facility is divided into two sections: site and facilities descriptions. These are presented here. The site environment was discussed in detail in Section 7.

18.4.1 Site Description

The representative facility occupies a total of 0.6 km² (148 acres), including the disposal area, operational area, and administration area. As is the case at existing disposal facilities, however, considerably less than the total site area is used for waste disposal. For example, specific areas of a particular disposal site may not be suitable for waste disposal due to geohydrological or topographical reasons. The area of the land committed for

FIGURE 18.2 Liquid Effluent Control for the Shallow Ground Waste Disposal Facility

FIGURE 18.3. Solid Waste Management for the Shallow Ground Waste Disposal Facility

waste disposal--that is, the land actually containing disposed radioactive waste--covers 0.35 km^2 (87 acres) at the representative facility. This area was calculated assuming random disposal (50% utilization of disposal space) of $1,000,000 \text{ m}^3$ of waste into trenches. The committed land use rate--that is, the unit volume of waste disposed per unit area of land--is estimated to be $2.88 \text{ m}^3/\text{m}^2$ ($9.45 \text{ ft}^3/\text{ft}^2$). The administration area occupies 0.04 km^2 (9.1 acres). The remaining 0.2 km^2 (53 acres) includes the operational area and the 30 m buffer zone inside the perimeter of the disposal area as well as any excess land within the disposal area used for roads, working areas, and so forth.

The entire site is surrounded by a 2.4-m (8 ft) chain-link fence topped with three strands of barbed wire. A 2.4-m fence also separates the administration area from the restricted area. Access to the disposal site is via a state highway running close to the site from which two short gravel roads lead onto the disposal facility. There are no rail facilities at the site. Incoming waste delivery and employee vehicles enter the site through one of two gates located in the administration area. These gates are locked at night and at other times such as holidays when the site is not being operated. Access to the restricted area is controlled by security check points near the gates in the fence separating the administration area and the restricted area.

For security purposes, a narrow gravel road runs alongside the inside of the fence surrounding the restricted area. Other onsite gravel roads, wide enough to accommodate two small vehicles, lead to the active disposal areas and are constructed by the licensee as needed. A lighting system is provided in the operational and administration area. There are no other lights installed in the interior of the restricted area.

18.4.2 Facility Description

A conceptual layout of the representative disposal facility is illustrated in Figure 18.4 and 18.5. As shown in the figures, the disposal facility is divided into two basic areas: a "restricted area" and an "administration area". Pursuant to 10CFR20, the restricted area is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials. The restricted area includes a "disposal area", in which disposal of radioactive waste takes place, as well as an "operational area". The restricted area includes a buffer zone between the disposal trenches and the restricted area fence of 30 m (100 ft). As shown in Figure 18.5, the operational area is located along the eastern side of the disposal facility and is also used for cask storage and for other miscellaneous functions. The operational area includes two facilities, a waste activities building and a garage, which are used to support waste disposal operations. The administration area is located near the eastern corner of the disposal facility and is uncontrolled with regard to radiation protection. The administration area includes support facilities plus parking space for employees as well as for incoming waste delivery vehicles. A more detailed discussion regarding the functions of the support facilities and structures is provided later in this section.

FIGURE 18.4. Layout of the Representative Shallow Land Waste Disposal Facility

FIGURE 18.5. Support Facilities for the Shallow Land Waste Disposal Facility

Disposal Facility

The disposal facility is assumed to have a total burial capacity of up to 1,000,000 m³ (35.3 million ft³) of waste which is delivered to the disposal site at an annual average rate of 50,000 m³ (1.77 million ft³) per year. The waste is randomly disposed into shallow land burial trenches having a design which is typical of current practices. Alternatives to this concept are considered in Section 18.6 and vary in the amount of land committed for waste disposal. For example, increased processing and volume reduction of waste decreases the amount of land needed for waste disposal.

The disposal area at the reference facility includes 58 disposal trenches. The average disposal trench is assumed to be 180 m (591 ft) long, 30 m (100 ft) wide, and 8 m (26 ft) deep (See Figure 18.6). However, the length and width of the disposal trenches may vary somewhat (about ± 10 m). The rather large trench sizes assumed are representative of recent trends at existing disposal sites.

The site soils are cohesive, but not cohesive enough to allow vertical-walled excavations. Therefore, trench wall slopes of 1 horizontal to 4 vertical (1:4) are employed. In some circumstances, (e.g., when extensive sloughing occurs) more gentle slopes are employed. The trenches are separated by natural soil walls that are 3.0 m thick at the top. These inter-trench walls are able to support only light vehicles. Other vehicles, including heavy construction and transport vehicles, require a more substantial substrate.

As a trench is constructed, the locations of the four corners of the trench are surveyed and referenced to a benchmark. An approximate one degree slope is provided in the bottom of a trench from end to end and from one side toward a small, 0.6 m x 0.6 m (2 ft x 2 ft) gravel-filled French drain, allowing the water to percolate into the soil. The French drain runs the entire length on the lower elevation side to provide for collection of liquid drainage that might occur. A gravel-filled sump is located at the low corner of the trench. Standing water in excess of the drains capability would be pumped and monitored, transferring it to the waste activities building, or the holding pond.

Each trench is equipped with a minimum of three 0.15 m (6 in.) diameter polyvinyl chloride (pvc) standpipes located within the French drain and standing along the sidewalls of the trench. The bottom 0.9 m (3 feet) of each standpipe is fitted with either a slotted plastic pipe screen or a wound mesh plastic screen. Two of the three standpipes are located at each end of the excavation. The third standpipe is usually located at the trench midpoint (also standing in the French drain). These plastic standpipes function as observation wells or sumps.

Support Facilities and Structures

The support facilities include: (1) an administration building, (2) a health physics/security building, (3) a warehouse, (4) a garage, (5) a waste activities building, and (6) a storage shed. All structures at the site are one-story metallic structures on concrete pad foundations. The building areas for these five major structures are listed in Table 18.1

TYPICAL TRENCH CROSS-SECTION

PLAN VIEW OF TYPICAL TRENCH

FIGURE 18.6. Typical Trench Details for the Shallow Land Waste Disposal Facility

TABLE 18.1. Structures of the Shallow Land Waste Disposal Facility

Building or Facility	Area	
	m ²	ft ²
Administration	625	6,725
Health Physics/Security	800	8,610
Warehouse	470	5,060
Garage Mechanics	420	4,520
Waste Activities	560	6,025
Storage Shed	80	860

The functions of each of the support facilities are described below.

The administration building contains office space for site management and other administrative personnel working at the site. The activities performed within this building include coordination of waste shipments to the site, billing customers, and other routine file work. Site records are also stored within this building.

The health physics/security building serves as the focal point for the majority of disposal activities at the site. This building houses a security section, a counting room, health physics offices, a change room/locker room, a lunch area, and a supply room. The health physics and security personnel are housed in the same facility because many of the functions performed by these personnel are complementary. Security personnel check both site personnel and visitors into and out of the site through a centrally-located checkpoint. The health physics personnel have the prime responsibility for checking vehicles into and out of the disposal area. All persons leaving the site must pass through a frisker station to check for contamination which may have been picked up onsite. A safety decontamination shower is located adjacent to the frisker location. Emergency equipment such as safety ladders, respiratory equipment, and anti-contamination suits are stored in the vicinity of the frisker station. The employee change/locker room down the hall from the health physics offices includes both a street clothes ("clean") and work clothes area. Showers are also located in this section of the building.

The waste activities building houses several functional areas, including: a large item decontamination bay, a control room for the decontamination bay, a liquid treatment system, a waste solidification, packaging, and overpacking area, a supply room, and a small waste storage area. The decontamination bay is used for washing down (decontaminating) large pieces of equipment (including trucks if necessary) through the use of a high-pressure recirculating water supply system. Contamination levels in these decontamination liquids are generally quite low; however, water treatment consisting of filtration is applied to recirculating fluids. This can then be directed to the waste treatment system shown in Figure 18.2, if necessary. Small-scale

decontamination of tools and other small items may be accomplished within the solidification staging area. The solidification area includes batch concrete-mixing equipment for solidification of small quantities of low-activity liquids. A small storage area is provided for occasional temporary storage of shipments received from common carriers. A loading dock is located along the southern corner of this building.

The warehouse is used to store supplies used onsite. This facility is located within the administration area so that delivery trucks need not enter the disposal area. Among the stored items in this warehouse are cables, hooks, drums, bags, and other miscellaneous hardware. Casks and site vehicles are stored in the operational area.

Only vehicles and equipment that have been surveyed and decontaminated to within specified limits are allowed to use the garage. The garage is large enough to hold two vehicles at a time for maintenance. Mechanic's tools, spare parts, oil, and fuel (adjacent to the building in underground tanks) are also stored in the garage.

A storage shed is used to store supplies and miscellaneous tools used at the disposal trenches. This shed is portable and is usually located close to the active disposal trenches.

18.5 SAFETY-RELATED INFORMATION

In this section, safety-related information is provided. This includes a description of operating staff, input and output of hazardous materials and their location, staff contact with hazardous material, etc.

18.5.1 Operating and Maintenance Requirements

This representative facility is assumed to operate only during daylight working hours, 8 hours per day. It is closed to waste receipt on weekends and holidays, with only a caretaker staff on hand. The facility is open 250 days per year.

The operating staff at the representative burial facility can be divided into the following categories: site management, site operations, health physics, security, administration, and quality assurance. Of these functions, only personnel in site operations and, to some extent, health physics, come into proximity with the waste handling process described earlier. A summary of the staff expected for the facility is given in Table 18.2.

All routine maintenance is done onsite. Operations and maintenance are generally performed by staff in the immediate process area, with significant hands-on operations. Protective clothing is worn by staff working in potentially hazardous locations.

TABLE 18.2 Staff for the Shallow Land Waste Disposal Facility

<u>Number of Staff</u>	<u>Job Description</u>
<u>Main Office</u>	
7	Professional & Management
5	Clerical & Support
<u>Support</u>	
4	Waste Shipment Schedulers
4	Security Personnel
6	Secretarial & Accounting
<u>Operation</u>	
11	Technicians
8	Heavy Equipment Operators
25	Skilled & Unskilled Laborers
<hr/>	
70	Total Staff

18.5.2 Input/Output of Hazardous Materials

No output of materials is expected from this facility. The exception to this would be shipments refused on inspection for disposal at the site, failed equipment scrap, and routine garbage and sanitary sewage service. The facility is assumed to accept 50,00 m³ of waste per year.

Waste Quantities

The principal waste streams sent to this facility are from the nuclear fuel cycle (reactors and support facilities), industry, and institutional sources. The summary of principal waste streams assumed for the nuclear fuel cycle is given in Table 18.3. This is based on operational experience to date in light water reactors (NRC 1981, p.D-22).

Table 18.3 includes estimates of wastes from UF₆ conversion and fuel fabrication facilities. Projections for fuel fabrication wastes were obtained from ONWI-20 (Phillips et al. 1979).

Process wastes consist of ion exchange resins, concentrated liquids, filters, etc. Ion exchange resins typically consist of organic resins (cation or anion), or an inorganic zeolite media can be used. The old bed is typically flushed out as a slurry into a shipping container, and free water removed from the exchange media. It can contain 42 to 55% water by weight.

Concentrated liquids produced by evaporation of a wide variety of liquid streams in LWRs are solidified on various matrix materials including bitumen, cement, and urea-formaldehyde.

TABLE 18.3. Low-Level Wastes from the Nuclear Fuel Cycle^(a)

<u>Waste Type</u>	<u>Volume</u>
Process wastes from	(M ³ /MW _e .yr)
BWR	0.483
PWR	0.218
Other fuel cycle facilities	0.018
Trash from	
BWR	0.326
PWR	0.326
Other fuel cycle facilities	0.104

(a) From NRC 1981 and Phillips et al. 1979.

Cartridge filters consisting of woven or wound fabric or paper supported by a stainless steel mesh are disposed of in 210-liter (55-gallon) drums.

Trash from LWR operation consists of a wide variety of compactable and noncompactable wastes, including: clothing, cloth items (rags, gloves, etc.), dirt, tools and equipment, filters, glass, metals (wires, channels, conduit, cans, fittings, valves, etc.), concrete, wood, plastic (bags, shoes, bottles, etc.), and rubber. Current practice is to generally ship these items mixed in the same containers on a relatively frequent basis.

Process wastes from nuclear fuel cycle facilities include calcium fluoride generated from gas scrubbers, bed material from fluidized bed reactors, and lime from treatment of liquid effluents in UF₆ conversion facilities. Fuel fabrication plants produce waste streams consisting of limestone, calcium fluoride, oxides from calciners, filter sludges, and small amounts of oil. Trash includes paper, plastic, equipment, and miscellaneous combustible materials.

Institutional waste generators include colleges and universities, medical schools, research facilities, and hospitals. In general, four broad classes of wastes are produced from these sources (Anderson et al. 1978, Beck et al. 1979): liquid scintillation vials and fluids (toluene, benzene, xylene, etc.) shipped with absorbent materials, other liquids (solidified or shipped with absorbent materials), biological wastes (shipped with absorbent material and lime), and general trash.

Wastes from industrial facilities may be grouped into five streams that are relatively small in volume but high in radioactivity: medical isotope production wastes, highly radioactive wastes, tritium manufacturing wastes, sealed sources, and accelerator targets. In addition, there are two groups of industrial facilities that generate four volumetrically significant waste

streams containing relatively low levels of radioactivity: 1) facilities using source and special nuclear materials (generating trash and other miscellaneous wastes), and 2) other facilities that use radioactive material and generate low specific activity wastes containing less than 3.5 Ci/m^3 (0.1 Ci/ft^3). Waste from these facilities can also be divided into trash and miscellaneous other wastes. The institutional and industrial wastes are summarized in Table 18.4. Current volume production levels are given, along with an added factor per year assuming expansion of the waste stream. These levels are not tied directly to LWR production rates, and cannot be given on a per $\text{GW}_e\text{-yr}$ basis.

TABLE 18.4. Estimated Current and Projected Future Volumes and Specific Activities of Low-Level Wastes from Non-Fuel Cycle Activities^(a)

<u>Waste Streams</u>	<u>Volumes (m^3)</u>		<u>Approximate Gross Specific Activity (Ci/m^3)</u>
	<u>Current</u>	<u>Added Per Year</u>	
Medical isotope production waste	193	14	573
Industrial high-activity waste			
• Sealed sources	5	0.4	5700
• Other high activity waste	74	5	210
Industrial low-activity waste ($\leq 3.5 \text{ Ci/m}^3$)			
• Source and special nuclear material	12,050	807	0.03
• Other low activity waste	4,608	309	0.03
Industrial tritium waste	99	7	2330

(a) Taken from Anderson et al. (1978) and Beck et al. (1979).

Radionuclide Concentrations in the Wastes

Most of the radionuclides contained in the low-level and intermediate-level wastes are short lived. For these radionuclides, relatively short-term isolation from man's environment (i.e., up to about 30 years) is needed from the burial facility. Isotopes with half-lives up to a few years may reach the disposal site, but they decay to insignificant levels after relatively short times when compared to the undisturbed life of the burial grounds. The radionuclides that are considered to be of long-term importance, have half-lives of more than a few years (5 years was used as per NRC 1981, p. D-28) and are present in comparatively significant quantities. Table 18.5 lists the

TABLE 18.5. Radionuclides of Long-Term Concern to Low-Level Waste Disposal

Isotopes	Half-Life (Years)	Principal Means of Production
^3H	12.3	Fission ^6Li (n, α)
^{14}C	5730	^{14}N (n, p)
^{55}Fe	2.60	^{54}Fe (n, γ)
^{60}Co	5.26	^{59}Co (n, γ)
^{59}Ni	80,000	^{58}Ni (n, γ)
^{63}Ni	92	^{62}Ni (n, γ)
^{90}Sr	28.1	Fission
^{94}Nb	20,000	^{93}Nb (n, γ)
^{99}Tc	2.12×10^5	Fission; ^{98}Mo (n, γ), ^{99}Mo (β^-)
^{129}I	1.17×10^7	Fission
^{135}Cs	3.0×10^6	Fission; daughter ^{135}Xe
^{137}Cs	30.0	Fission
^{235}U	7.1×10^8	Natural
^{238}U	4.51×10^9	Natural
^{237}Np	2.14×10^6	^{238}U (n, 2n), ^{237}U (β^-)
^{238}Pu	86.4	^{237}Np (n, γ), ^{238}Np (β^-); daughter ^{242}Cm
^{239}Pu	24,400	^{238}U (n, γ), ^{238}U (β^-), ^{239}Np (β^-)
^{240}Pu	6,580	Multiple n-capture
^{241}Pu	13.2	Multiple n-capture
^{242}Pu	2.79×10^5	Multiple n-capture; daughter ^{242}Am
^{241}Am	458	Daughter ^{241}Pu
^{243}Am	7950	Multiple n-capture
^{243}Cm	32	Multiple n-capture
^{244}Wm	17.6	Multiple n-capture

radionuclides from all sources considered to be of importance from a quantity and biological toxicity viewpoint and thus are of long-term importance in shallow land burial. There are severe regulatory restrictions in the amounts of the transuranic nuclides that can be disposed of in shallow land burial (CFR 1982). Furthermore, relatively small amounts of most of the other radionuclides in Table 18.5 are expected to be received at the shallow land burial facility. The radionuclides considered as long-lived are identified by their typical concentration from the major waste sources in Table 18.6 (NRC 1981, p. D-28). Process wastes include ion exchange resins, concentrated liquids, filter sludges, cartridges, etc. Multiplying these values by volumes given in Table 18.3 will give Ci/MWe·yr for nuclear fuel cycle facilities. Total long-lived curies entering the facility per year is then $6.3\text{E}8$ Ci/yr from Table 18.6 ($50,000 \text{ m}^3/\text{yr} \times 1.26\text{E}+4 \text{ Ci/m}^3$). As received, the amount of short-lived curies would far exceed this value, but these decay rapidly.

The "special" column in Table 18.6 includes LWR decontamination which is difficult to estimate. NUREG-0782 (NRC 1981) assumes that reactors require primary system decontamination every 5 to 10 years, producing 90 m^3 and 47.5 m^3 of process wastes for BWRs and PWRs, respectively. The "other" category includes wastes from isotope production, sealed sources and accelerator targets.

It is estimated that the effluents from the packaging, decontamination processes will be negligible.

An estimate of the wastes produced in the waste activities building is not available. It is assumed that the wastes generated are small (i.e., less than 1%) compared to the material received from offsite, and is included in the $50,000 \text{ m}^3/\text{yr}$ capacity of the facility. Radionuclide will decay in permanent storage at the site.

18.5.3 Inventories of Major Hazardous Materials at Plant Locations Having Safety Related Importance

The major inventory of radioactive materials at the representative SLWBGH is located in the burial trenches themselves. Shipments are processed as they are received for placement in the trenches. Temporary storage is provided for only relatively small shipments and most shipments are kept on the shipping vehicle until accepted for disposal. Much lower inventories would be expected in liquid and solid waste processing.

For the purposes of this report, it is assumed that 10% of the daily receipt ($200 \text{ m}^3/\text{day}$) is located in the receiving and off-loading areas (20 m^3) at any one time, and that 1% (2 m^3) of the waste requires temporary storage (1 to 5 days). A similar quantity (1%) is estimated for use in repackaging functions at the waste activities building. No inventory estimates are available for liquids in the holding tanks for decontamination activities.

An estimate of inventories in the trenches is made assuming 58 trenches at the site, with a $1.72 \times 10^4 \text{ m}^3$ capacity per trench. Using the estimate of $1.26\text{E}4 \text{ Ci/m}^3$ of long-lived radionuclides given in Table 18.6 for the waste,

TABLE 18.6. Long-Lived Isotopic Content of Low- and Intermediate-Level Wastes (Ci/m³)(a)

Nuclide	Process Wastes			Trash			Institutional, Industrial		Special	
	PWR	BWR	Fuel Cycle	PWR	BWR	Fuel Cycle	Trash	Process	LWR Decon	Other
³ H	9.88-3	3.24-2	0	7.29-3	1.10-2	0	1.20 1	3.38-1	1.08-2	3.27+3
¹⁴ C	3.63-4	2.01-3	0	2.68-4	6.77-4	0	6.90-3	1.94-2	2.60-1	5.76+1
⁵⁵ Fe	8.86-1	2.46+0	0	1.43-1	9.75-1	0	0	0	2.27+3	1.15+2
⁵⁹ Ni	1.06-3	2.55-3	0	1.71-4	1.01-3	0	0	0	1.44+0	6.56-2
⁶⁰ Co	1.72+0	4.13+0	0	2.77-1	1.63+0	0	1.37-2	3.67-2	1.67+3	1.81+3
⁶³ Ni	3.24-1	5.57-2	0	5.27-2	2.20-2	0	0	0	2.13+2	2.41+2
⁹⁴ Nb	3.36-5	8.04-5	0	5.41-6	3.18-5	0	0	0	9.61-3	4.47-4
⁹⁰ Sr	7.20-4	7.19-3	0	5.33-4	2.06-3	0	1.90-3	1.83-2	4.28-2	1.16+3
⁹⁹ Tc	3.00-6	1.29-4	0	2.26-6	4.36-5	0	4.45-9	1.75-8	1.20-5	3.27-4
¹²⁹ I	9.35-6	3.44-4	0	6.69-6	1.16-4	0	0	0	3.34-5	2.72-6
¹³⁵ Cs	3.00-6	1.29-4	0	2.26-6	4.36-5	0	0	0	1.20-5	3.27-4
¹³⁷ Cs	8.35-2	3.44+0	0	6.03-2	1.16+0	0	5.98-3	2.35-2	2.18-1	1.16+3
²³⁵ U	6.20-7	4.20-7	3.95-5	1.90-7	1.98-7	2.31-6	2.36-6	4.60-5	6.4-5	1.02-5
²³⁸ U	4.84-6	3.30-6	4.49-4	1.49-6	1.56-6	8.60-6	8.80-6	1.71-4	5.40-4	3.81-5
²³⁷ Np	1.19-10	8.06-11	0	3.64-11	3.80-11	0	0	0	1.32-8	5.33-13
²³⁸ Pu	3.72-4	7.48-4	0	1.44-4	3.73-4	0	0	0	1.34+0	1.97-4
²³⁸ U/ ²⁴⁰ Pu	5.86-4	3.84-4	0	1.33-4	1.87-4	0	0	0	1.77+0	5.55-5
²⁴¹ Pu	2.56-2	1.87-2	0	5.79-3	9.14-3	0	0	0	3.55+1	7.10-3
²⁴² Pu	1.29-6	8.41-7	0	2.91-7	4.11-7	0	0	0	3.87-3	9.57-8
²⁴¹ Am	4.77-4	2.99-4	0	9.52-5	1.57-4	0	6.33-6	0	5.29-3	5.76+2
²⁴³ Am	3.21-5	2.02-5	0	6.42-6	1.06-5	0	0	0	3.59-4	1.25-6
²⁴³ Cm	5.27-7	5.83-7	0	6.57-8	3.14-7	0	0	0	3.46-4	1.65-4
²⁴⁴ Cm	3.20-4	4.47-4	0	6.26+5	2.42-4	0	0	0	3.27-3	2.88-7
Total	3.05+0	1.01+1	4.89-4	5.48-1	3.81+0	1.09-5	1.48-1	4.37-1	4.20+3	8.40+3
GRAND TOTAL: 1.26 x 10 ⁴ Ci/m ³										

(a) Taken from NRC 1981.

an estimate of $2.2\text{E}8$ Ci per trench is given. A similar calculation for the 20 m^3 assumed for the offloading and receiving area given $\sim 3\text{E}5$ Ci in solid packaged form.

Radionuclide inventories for the shipments were presented earlier. The license for the facility specifies a limit in curies for undisposed of waste, taken here as 60,000 Ci. Table 18.7 lists the inventory locations, quantities and chemical forms assumed.

TABLE 18.7. Inventories of Long-Lived Radioactive Materials in the Shallow Land Waste Disposal Facility

<u>Location/Material</u>	<u>Form</u>		<u>Quantity m^3</u>	<u>Activity Ci</u>
	<u>Chemical</u>	<u>Phase</u>		
Burial Trenches (58)	Various	S	1.72E4 per trench	2.2E8
Off-loading	Various	S	20	3E5
Receiving Area	Various	S	20	3E5
<u>Waste Activities Bldg.</u>				
Temporary Storage	Various	S	2	30,000
Repackaging	Various	S	2	N A
Liquid Holding Tank	H ₂ O	L&S	N A	N A
Concentrate Holding Tank	Sludge	L&S	N A	N A

Key:

S = solid

L = liquid

G = gaseous

NA = not applicable

18.5.4 Unit Operations Involved

This section identifies the unit operations involved for each step in the process flow scheme for the representative SLWBGF (See Figures 18.1 through 18.3). Table 18.8 lists these operations.

18.5.5 Accident Driving Forces

This section identifies the driving forces that could potentially cause accidents at the facility. Table 18.9 lists the driving forces and their locations.

TABLE 18.8 Unit Operations Involved in the Shallow Land Waste Burial Ground Facility

<u>Process Step</u>	<u>Circuit Operations</u>
<u>Waste Handling and Disposal</u>	
Receiving and Inspection	Radiation survey
Transport to burial site	Conveying by truck
Off-loading hook-up to crane	Lifting, conveying
Vehicle inspection	Radiation survey
Placement in trenches	Lifting by crane, conveying
Backfill trenches	Filling, grading
<u>Facility Waste & Effluent Management</u>	
Decontamination	Wiping, washing, scrubbing
Repackaging	Packaging, conveyance
Radionuclide concentration	Filtration, ion exchange, evaporation, condensation
Solid-liquid solidification	Cementing
Solids packaging	Compression, shredding
	Packaging

TABLE 18.9. Accident Driving Forces at the Shallow Land Waste Burial Ground Facility

<u>Accident Driving Force</u>	<u>Location</u>	<u>Characteristics</u>
Temperatures	Evaporator	50°C - 150°C
Pressures	Compactor	1E6 Pa (10 atm)
	Water Spray	7E5 Pa (7 atm)
	Decontamination	
Chemical Reactivity	Resin Bed	Moderate Reactivity
Potential Energy	Waste Off-Loading	Package Drop (2m)
Nuclear Reactivity	Storage Area	500g
	Burial Trenches	500g
Thermal Energy	Evaporator Power	NA ^(a)
	Source	

(a) NA = not applicable.

8.5.6 Containment and Shielding Systems

Shielding at the representative SLWBGF is provided on an as-needed basis to maintain personnel exposure levels in compliance with 10CRF20, and to keep levels as low as reasonably achievable. Normal external package surface radiation levels are limited to 200 mR/h or 10 mR/h at three feet. However, packages transported by "sole use" vehicles may have radiation levels up to 1 R/h at three feet. Portable shielding on package handling equipment could be employed in such cases. Shielding may be provided on waste concentrate storage tanks. Other than this, shielding provided by the normal structures of the waste handling building are sufficient.

18.5.7 Degree of Hands-On-Operation

This subject was discussed in Section 18.5.1.

18.6 ALTERNATE PROCESS SCHEMES

Alternatives for the representative SLWBGF consist of two basic areas:

- changes in the amount and type of onsite processing
- changes in the method of site construction and disposal techniques required.

These are discussed below, with unique safety-related information called out.

18.6.1 Onsite Processing

In the representative facility description, onsite processing of wastes was held to repackaging of improperly packed shipments and decontamination of equipment and personnel. In general, shipments were disposed of as they were received. Any processing for volume reduction or solidification was assumed to occur at the site where the waste originated.

³ It is assumed that facility fees for waste receipt are charged on a per m³ basis plus extra fees for any required treatment, decontamination or repackaging, so the incentive will remain for processing at the point of origin. However, the option does exist for processing at the SLWBGF. The processing options include: 1) volume reduction by compaction, evaporation, or incineration, and 2) volume increase by solidification in various media or soaking of liquids in absorbent media.

Volume Reduction

There are three basic processes that can be applied to waste streams that result in overall waste volume reduction: 1) physical processes such as compaction, 2) thermal processes such as evaporation (which in this study includes belumenization), and 3) incineration and other related combustion processes. Each of these processes produces a concentrate stream and an effluent stream. The respective concentrate streams are compressed wastes, concentrated liquids or crystals, and ash. The respective effluents displaced are the air, vapor, and gas. The radioactivity per unit volume of the

concentrate stream is usually higher than that of the untreated waste with the possible exception of volatile nuclides such as tritium, carbon, and iodine that may be removed as vapor and/or combustion products in the effluent steams.

Compaction. Compaction is an often used method--particularly at nuclear fuel cycle facilities--of reducing the volume of waste streams containing compressible material such as paper, plastic, glass, wood, and light-gauge metal. Most of the volume reduction is attained by compressing the waste to reduce its void volume. The term compactor is usually applied to hydraulic or mechanical rams that compress wastes into 210-liter (55-gallon) steel drums. The drums are then used as disposal containers. Typical hydraulic rams generate 1 to 1.3×10^5 N (20,000 to 30,000 pounds) of force, and are fitted with shrouds and simple air filtration systems to minimize release of airborne radioactivity. Safety hazards would be primarily industrial in nature, due to the high operating pressures.

Most compactors now in use can achieve average volume reduction factors of about two. Newer compactors, which place a metal inner sleeve inside the drum during compaction, are capable of a volume reduction factor of about four. Industrial hydraulic presses similar to those used to crush automobiles may be useful for compacting heavier gauge metal items such as pipes, tools, cans, drums, and scaffolding.

Evaporation. Evaporators concentrate liquid wastes by heating them to vaporize the volatile components. The vaporized water generally contains greatly reduced quantities (typically 3 to 5 orders of magnitude) of dissolved solids, suspended solids, and radioactivity relative to those found in the input waste stream. In the nuclear industry the vaporized water is normally condensed and collected, and then either discharged or recycled after testing to determine whether the condensate requires additional treatment. The concentrated solution (bottoms) left in the evaporator retains virtually all of the solids and radioactivity and is then solidified in another process and shipped to a disposal facility. This is done on a small scale for treatment of decontamination process water in the representative facility. However, processing of the original waste streams could require a much more elaborate system.

Evaporators can be categorized according to their methods of heat transfer. Natural circulation evaporators use convection as the means of heat transfer. Forced circulation evaporators use pumps to improve the flow of liquid over the heating surfaces. Fluidized-bed dryers produce dry salts by injecting atomized waste liquids onto a hot bed of inert granules that is suspended (fluidized) in a stream of hot air. The liquids flash-evaporate on contact with the hot bed, leaving behind a residue of dry solids. The inert carrier process uses a hot bath of inert fluid recirculating at high velocities as the heat exchanger. Solidification in bitumen can also be considered as a form of evaporation. The ideal evaporator produces a condensate that is free of radioactivity while attaining the maximum volume reduction of the input waste liquid. Industrial safety hazards result from the operation of high pressure and temperature processes. The presence of concentrated radioactive waste streams may make shielding and special maintenance procedures necessary.

Incineration. Incinerators and related devices decompose combustible waste materials by thermal oxidation. Combustion or incineration involves complete oxidation of wastes by burning in an excess of oxygen (air). Oxidation can also be accomplished by introducing combustible wastes and air into a bath of molten salt. Alternatively, acid digesters oxidize wastes in a hot mixture of concentrated nitric and sulfuric acids. Pyrolysis involves thermal decomposition in an oxygen-deficient atmosphere.

The various types of incinerators, pyrolyzers, and other such devices currently used or being developed for volume reduction of radioactive waste are too numerous to be considered here individually. Two types of representative incinerators have been selected for discussion in this section: incinerators for pathological wastes and fluidized-bed incinerators. The pathological incinerator is considered for optional use by large institutional waste generators such as hospitals and biomedical research facilities. The fluidized-bed incinerator is considered for optional use by fuel cycle waste generators or by operators of a potential regional waste processing facility incinerating wastes from small waste generators.

Pathological incinerators are typically multiple-chamber, hot refractory hearth incinerators and are normally operated with less sophisticated off-gas treatment systems. Airborne releases are principally controlled through control of the rate of input feed. They are designed primarily for the incineration of animal carcasses and operate at approximately 900 to 1000°C. Pathological incinerators may also be used by institutional waste generators for volume reduction of other biowastes, scintillation fluids, organic liquids, and trash. Aqueous liquids can also be evaporated on the refractory hearth.

Fluidized-bed incinerators operate by injecting combustible wastes into a hot bed of inert granules fluidized by a stream of hot gas. They operate on the same principle as fluidized-bed dryers or calciners which have been used for many years in nonnuclear industries to produce dry solids from liquid wastes by complete evaporation of water. Typical fluidized-bed incinerators can burn trash, organic solvents, and ion exchange resins. Wastes are normally screened to remove metal objects and are shredded before entering the process vessel. The process vessel is maintained at 800 to 1000°C. Residual ash from the combustion process is collected for solidification. Ash carried out of the process vessel with the hot effluent gas stream is separated from the effluent gas by an off-gas treatment system, and also collected for subsequent solidification. Safety hazards center primarily around the high temperatures involved in incineration. This includes high temperature gases and combustion products. Routine hazards associated with the heat source (likely natural gas) must also be considered. Reduction hazards come primarily from the increased handling of materials onsite.

Volume Increase

There are three basic processes that can be applied to waste streams that can result in an overall waste volume increase: solidification in a medium which chemically or physically binds the material, or addition of absorbent materials, and packaging. The radioactivity of the product stream is generally lower than that of the input waste.

Solidification in Media. This section considers a number of solidification processes that can be applied to waste streams such as LWR process wastes (concentrated liquids, resins, filter sludges, and cartridge filters), or dry salts and ashes produced by calciners and incinerators. Cartridge filters are assumed to be solidified by pouring the solidification agent into the spaces between the currently utilized waste canister and cartridges. This results in no change to the currently-shipped volume of the waste stream.

The solidification agents or techniques considered here are selected from those that are currently in use or being actively marketed. These include cement, urea-formaldehyde and other synthetic polymer systems.

Absorbents such as vermiculite and diatomaceous earth are not considered to be solidification agents since they do not chemically or physically bind the wastes. Both cement and urea-formaldehyde solidification systems are currently used by LWRs. Bitumen (another agent) and vinyl ester-styrene (a synthetic polymer) are being actively marketed. Several bitumen solidification systems (which are widely used in Europe) have been sold but are not yet operational in this country. Synthetic polymer systems are being currently used in LWRs, including the Dresden Unit 1 nuclear power plant, where decontamination solutions are to be solidified. Polyester (another synthetic polymer) has been evaluated in laboratory and pilot plant studies using simulated LWR liquid wastes and may be routinely used in the future.

The solidifying agents mentioned are used routinely in industry for non-nuclear purposes. The hazards generated would be primarily chemical in nature, including heat, vapors, and fumes, given off during mixing, pouring and curing of the agent. Routine industrial safety measures would suffice.

Absorbent Materials. Absorbent materials such as diatomaceous earth or vermiculite are currently added to several institutional waste streams to minimize potential transportation impacts. These streams include liquid scintillation vial waste, absorbed liquid waste, and biowaste. Existing commercial disposal facility operators require that these wastes be packaged with specified proportions of waste to absorbent material before they are accepted for disposal. For example, liquid scintillation vial waste is required to be packaged using sufficient absorbent material to absorb twice the total volume of the liquid in the package. Lime is frequently added to the biowaste stream. Double packaging of these waste streams is also used for additional safety. For the liquid scintillation vial and the absorbed liquid waste streams, a volume increase factor of 3.0 is assumed. For the biowaste stream, a volume increase factor of 1.9 is assumed. Lime is chemically active and requires caution in its use. Other media such as vermiculite or molecular sieves tend to be inert and safe to handle.

The practice of packaging wastes with absorbent material increases the difficulty of further processing of these wastes with currently available methods, if delivered to a centralized processing facility. This is because many of the common absorbent materials, in integral part of the waste stream when the package leaves the waste generator, are not incinerable and pose a

sludge-handling difficulty. Absorbents that are incinerable are either not cost-effective or not compatible with waste treatment systems. Because of this, it is very unlikely that absorbent materials would be used at a facility if more efficient incineration processes are available.

18.6.2 Disposal Options

A number of variations in near-surface disposal techniques are possible as shown in Table 18.10. These are the options considered for the SLWBGF developed in NUREG-0782 (NRC 1981). This table presents the relative number of additional personnel and construction equipment required for construction of the option compared to the representative SLWBGF. Other land disposal alternatives include intermediate-depth disposal and mined cavity disposal.

Intermediate-Depth Disposal

Intermediate-depth disposal is assumed here to mean disposal of wastes at depths greater than 15 m. Deep burial at some localities (especially many areas in the humid eastern United States) may be difficult as a result of relatively shallow depth to groundwater and heterogeneous subsurface media (e.g., fractured rock). However, it is believed that intermediate depth disposal could easily be practiced in a large number of areas in the western and southwestern U.S.

The use of deep trenches appears to have limited applicability. As previously stated, in the humid eastern U.S., the number of locations amenable to a deep excavation (35 to 50 m) without encountering either groundwater or fractured media is probably quite small. Additionally, the practicality of constructing a deep trench may be questionable from both a side slope requirement and an operational requirement. The large depth excavation would require (if excavation as a trench were feasible) either substantial shoring to keep the excavation open or would require terracing or gentle slopes. Once the slopes are terraced or gently sloped, the excavation then begins to resemble an open-pit mine or strip-mine geometry. Another drawback to using deep trench excavations is the potential difficulty in waste emplacement. Existing conventional lifting equipment would not be adequate. Either modified mine hoists or dumping over the side or placement with equipment inside the trench would be required. Use of hoists could significantly increase the labor requirements in elevated radiation fields while dumping wastes into deep excavations would probably rupture many waste containers.

Application of strip-mine or open-pit mine technologies appear to be more viable options. Surface mining technologies can be applied to either existing mines that have not been fully reclaimed or to new sites where geological conditions would permit such large excavations. (It should be noted that the principal goal of the application of intermediate-depth disposal is to provide added protection for the inadvertent intruder and such extensive protection may not be required). For purposes of analyses, an average annual waste input of $50,000 \text{ m}^3$ ($1.77 \text{ million ft}^3$) over a 20-year duration is assumed.

The intermediate-depth disposal facility can be accommodated on a leased tract of land equivalent to that described for the representative disposal

TABLE 18.10. Options for Near-Surface Disposal of Low-Level Wastes

Disposal Options	Relative No. of Additional Personnel	Relative No. of Additional Equipment
<u>Design Variation</u>		
Deeper trench (no shoring)	low	low
Deeper trench (shoring)	mod	low
Increased distance	none	none
Thicker trench cover	low	low
Layered waste disposal	low	low
Slit trench (10% of vol.)	low	low
Caisson disposal (10% of vol.)	mod	mod
Concrete walled trench (10% of vol.)	mod	mod
Concrete walled trench (100% of vol.)	high	high
Grouting	low	mod
Engineered intruder barrier	mod	high
<u>Control of Potential Long-Term Release</u>		
Improved monitoring	low	low
Improved thicker trench covers	low	low
Moisture barriers	low	low
Sand backfill	low	low
Surface water drainage	low	low
Weather shielding	low	low
Stacked emplacement	mod	mod
Waste segregation	low	low
Improved compaction	low	low
Decontainerized disposal	high	mod
Dynamic compaction	low	low

facility (approximately 0.6 km^2 (148 acres)), including the excavation itself, the administrative area, overburden storage areas, waste activities buildings and staging areas.

Both the design and operation of the intermediate-depth disposal facility differ significantly from the representative disposal facility described in this section. The excavation is assumed to be a circular open-pit "mine" with a spiral access road leading down into the excavation. The excavation is roughly circular with a 410 m diameter at the base of the excavation and a 480 m diameter at ground surface. The maximum depth of the excavation is 34 m. The disposal cell consists of three layers of waste each 5 m thick. Each layer of waste is emplaced by forklifts and boom cranes within the excavation. Random emplacement is assumed throughout the operations with a volume utilization of 50% assumed.

After the final waste layer is emplaced, a final disposal cap of 15 meters of overburden is added, which results in a minimum thickness of 15 m between the waste and the final grade. The equipment needs for the intermediate-depth disposal are similar to those for the representative disposal facility, with the addition of 3 forklifts for waste emplacement activities, 2 pan scrapers to handle the extra overburden volumes, 4 dump trucks to supplement earth-moving activities, and 2 extra cranes for waste emplacement. The building and structure requirements are assumed to be similar to those required for the representative near-surface disposal facility described earlier in this section.

Mined or Natural Cavity Disposal

The two basic options available for mined cavity disposal are: 1) creation of a new-mined cavity for disposal, and 2) rehabilitation of an existing mine for disposal. Construction of a new-mined cavity can be accomplished either in salt media or in hard rock media (e.g., granite or basalt). One significant variation from a near-surface disposal facility is the land requirement. To control access to the mine and prevent intrusion (especially in the form of well drilling), between 4.1 and 5.7 km^2 (1,000 and 1,400 acres) of surface property are assumed to be required for a mined cavity capable of handling $1,000,000 \text{ m}^3$ of waste.

The initial capital outlay for a mined cavity disposal is considerably higher than that for a near-surface disposal facility (\$40 to 60 million as opposed to \$7.5 million) (NRC 1981, pg. F-46). The significant components of capital investment include underground equipment (e.g., continuous mining machines and trucks), surface equipment (e.g., waste handling equipment), surface facilities, and the construction of four shafts (i.e., waste, worker, air intake, and air exhaust).

Compared to a near-surface disposal facility, a number of additional surface buildings and facilities are required for a mined cavity disposal facility. The additional surface facilities include: a waste receiving building, cap and powder storage magazines (for hard-rock mining only), a hoist building, and an electrical substation for underground power requirements. A significant area (0.1 km^2 or 25 acres) would have to be set aside at each mined cavity disposal site for mined material storage (new mines only).

Disposal operations require relatively close contact with the waste packages. Waste is handled, in effect, twice: once at the receiving building, and once within the mined cavity. As a result of the double handling and the confined working areas, it is estimated that the occupational exposures for mined cavity operation for 1,000,000 m³ of waste are about four times higher than those experienced at a near-surface disposal facility. A large mined cavity disposal facility will require between 150 and 175 persons on the payroll (over twice the requirement for a near-surface disposal facility). While a near-surface disposal facility will require approximately 40 people to work directly with the waste (intermittent or continuous work), a mined cavity facility will require between 80 and 100 persons to work directly with the waste.

Safety-Related Information for Disposal Options

Safety issues for disposal options can be divided into two broad categories: construction and operational safety. During construction, engineered safety features generally become more critical as depths increase. The need for shoring or caissons becomes important as trenches are deepened, and becomes mandatory for still deeper excavations or mining operations to prevent cave-ins. In the latter, control of atmosphere, lighting, and water also becomes safety issues. Designs which call for engineered trench covers or layered covers should only increase slightly the normal industrial hazards associated with earth-moving equipment and operations. Mining or deep excavations introduce new hazards associated with falls or falling equipment, in addition to the hazards mentioned above. Deep excavation may also require blasting and debris removal using specialized equipment.

During operation, options using shallow land burial with engineered cover layers or thicker soil caps are expected to be quite similar to the representative facility in terms of industrial safety hazards. Increased activities would, of course, be required for earth moving, etc. Radiation safety problems associated with handling and placement of wastes in the trench would be quite similar.

For deeper excavations or mined disposal, radiation exposure for handling packages in tight quarters should be examined. For example, doses in rock tunnels would likely be higher due to reflection compared to surface handling of similar packages. Radiation safety aspects of equipment failure and maintenance on special package conveying equipment for mining disposal may also make more demands than surface equipment.

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19.0 TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE NUCLEAR FUEL CYCLE

Currently operating and planned facilities associated with the nuclear fuel cycle are situated in diverse geographic locations. Radioactive materials in a wide variety of physical and chemical forms must be transported between these facilities. Materials such as uranium ore concentrate, uranium hexafluoride, fresh fuel, spent fuel, plutonium and many forms of radioactive waste have been shipped routinely in the U.S. Other radioactive materials requiring transportation are expected to be produced in the future (e.g., solidified high-level waste, fuel residues, plutonium with high heat generation rates and transuranic wastes).

This section presents a description of the various transportation systems in use or expected to be developed in the future for nuclear fuel cycle materials. Standards and criteria for packaging and transportation are contained in Title 10 and Title 49 of the Code of Federal Regulations. Federal regulations prescribe shipping container requirements, limitations on package contents, radiation dose limits, and packaging and handling procedures. State agencies regulate vehicle sizes and weights and in some cases transportation routes and times of travel. The purpose of the regulations is to ensure that radioactive material shipments pose minimal risks to the public and to transportation workers.

Transportation methods and safety-related information are described in this section for the following radioactive materials:

- uranium ore
- uranium ore concentrates (yellowcake)
- natural uranium hexafluoride (UF_6)
- enriched uranium hexafluoride (UF_6)
- plutonium dioxide (PuO_2)
- uranium dioxide (UO_2) and mixed UO_2 - PuO_2 fuel assemblies
- irradiated fuel
- high-level wastes
- transuranic wastes (contact-handled and remote-handled)
- low and intermediate level non-transuranic wastes.

The majority of radioactive material packages are presently carried on trucks, which 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 Mg (24 tons). Shipments exceeding this weight must be made in accordance with overweight permits issued by state agencies. 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. Payloads of 100 Mg (110 tons) or more can be accommodated in rail shipments.

19.1 SUMMARY

Transportation forms a vital link in the commercial nuclear fuel cycle. The transportation of radioactive materials is regulated by the U.S. Department of Transportation (DOT) and the U.S. Nuclear Regulatory Commission (NRC). Their regulations delineate packaging requirements, allowable radiation dose rates and handling procedures to insure that shipments of radioactive materials are made in a manner that poses minimal 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 19.1 summarizes the packaging and shipping information on radioactive materials in the nuclear fuel cycle. Transportation requirements for the current commercial "once-through" nuclear fuel cycle as well as for a postulated fuel reprocessing and plutonium recycle scenario are discussed. The status of the technology for transporting these materials is also shown, i.e., whether the shipping systems indicated are currently in use or are under development. As indicated in the table, most shipments of radioactive materials are made truck or by rail. This is expected to continue in the future.

The transportation systems for the materials listed in Table 19.1 are described in more detail in the following sections. Also included is safety-related information useful in determining the risks associated with transportation of radioactive materials. Alternative transportation systems are discussed where they exist.

19.2 TRANSPORTATION OF URANIUM ORE

Uranium ore is extracted from two types of mines; open-pit mines and underground mines. The ore is shipped as loose material to a uranium mill, which is usually assumed to be located relatively near the mine. Private haulage roads are most often used to transport this material. Uranium ore is primarily shipped in conventional dump trucks, although conventional rail boxcars are also utilized.

19.2.1 Physical Description of the Uranium Ore Shipping System

Uranium ore is shipped to the ore stockpiles adjacent to the mill in 23 Mg truckloads. Standard dump trucks are utilized for ore transportation, although the loose material is usually covered to minimize loss of uranium ore particles during transport. The average shipping distance from the mine facility to the milling facility is approximately 50 km. The transport of raw uranium ore takes place primarily on private haulage roads, rather than public highways (NRC 1979).

Uranium ore is designated as "low specific activity" (LSA) material in DOT regulations. It requires no special packaging when transported in an exclusive-use vehicle (one which carries no other freight) (Rhoads 1977).

TABLE 19.1. Summary of the Radioactive Materials Transportation Systems Described in this Study

Radioactive Material	Origin	Destination	Form of Material	Packaging	Quantity per Shipment	Curies per Shipment ^(a)	Shielding Required	Level of Package Containment ^(b)	Status of Technology
Uranium ore	U mine	U mill (largely on private roads)	U ore	Loose ore in dump truck	23 Mg	2.8E-2	No	LSA	In use
Yellowcake	U mill	UF ₆ conversion facility	U ₃ O ₈	U ₃ O ₈ in 210-l drums in enclosed truck trailer	50 drums or 18 Mg	1.2E1	No	LSA	In use
Natural UF ₆	UF ₆ conversion facility	U enrichment plant	Solid UF ₆	9.0 or 12.7 Mg steel cylinders	Two 9.6 Mg or one 12.7 Mg cylinder per truck; four each per railcar	NA	No	LSA	In use
Enriched UF ₆	U enrichment plant	UO ₂ fuel fabrication facility	Solid UF ₆	2.3 or 9 Mg steel cylinder in Type B overpack on trucks	Five 2.3 Mg or one 9 Mg cylinder per truck	3.8E1	No	Type B	In use
Plutonium	Fuel reprocessing plant	Mixed oxide fuel fabrication plant	PuO ₂ powder	Primary pressure vessel inside Type B pressure vessel	Seven 32 kg capacity containers per truck	5.8E5	Yes	Type B	Under development
Unirradiated UO ₂ and mixed UO ₂ -PuO ₂ fuel assemblies	UO ₂ fuel fabrication plant or mixed oxide fuel fabrication plant	Reactor	Assembled UO ₂ and UO ₂ -PuO ₂ fuel rods	BWR fuel: inner steel container within wooden rectangular box. PWR fuel: inner steel cradle assembly within steel outer cylinder	Each container transports two assemblies. For BWR, sixteen max. per truck; PWR, 8 max. per truck	NA	Yes (for MOX)	Type A	In use for UO ₂ fuel
Spent fuel	Reactor	Fuel reprocessing plant, interim storage facility, or final isolation facility	Bare fuel assemblies	Truck cask	1 PWR or 2 BWR assemblies	1.6E6	Yes	Type B	In use
				Rail cask	7 PWR or 18 BWR assemblies	1.2E7	Yes	Type B	In use
Solidified high-level wastes	Fuel reprocessing plant	Interim storage facility or final isolation facility	Borosilicate glass	Truck cask	One canister of SHLW	6.9E5	Yes	Type B	Under development
				Rail cask	Nine canisters of SHLW	5.6E7			Under development
Contact-handled TRU wastes	Fuel reprocessing plant or mixed-oxide fuel fabrication plant	Interim TRU waste storage facility or final isolation facility	Many	Type A canister (drums or boxes) inside a Type B overpack	Truck; (1 TRUPACT) Rail; (2 TRUPACT)	8.7E3	No	Type B	Under development Under development
Remote-handled TRU wastes	Fuel reprocessing plant or mixed oxide fuel fabrication plant	Interim TRU waste storage facility or final isolation facility	Fuel residues	60 cm OD x 3.1 m long canisters	Truck cask; one canister and one cask per truck.	2.6E4	Yes	Type B	Under development
					Rail cask; three canisters and one cask per railcar	7.8E4			Under development
			Many	210-l drums	Three drum cask; one cask per truck and three per railcar	2.3E3	Yes	Type B	In use
Low and intermediate level non-TRU wastes	All fuel cycle facilities	Final isolation facility	Many	Metal drums, bins, boxes inside shielded and unshielded overpacks	Many types	NA	Yes	LSA Type A	In use

(a) Where a variety of package alternatives exist, the container yielding the largest level of radioactivity per shipment has been used for the purpose of this table.

(b) For definition of these categories, see 10CFR71 and 49CFR173.

19.2.2 Safety-Related Information

The operating staff required for transport of uranium ore from the mine to the mill consists of one driver per truck.

The primary input requirements to uranium ore transportation involve the ore product (23 Mg) and the diesel fuel required to power the trucks. The major effluents from the transportation step are the airborne pollutants associated with combustion of diesel fuel (SO_x , NO_x , particulates, etc.). The primary hazard from uranium ore stems from its radioactivity. Uranium ore has a specific activity of $1.2\text{E-}9$ Ci/g, or $2.76\text{E-}2$ Ci/truckload. The material is in solid form. The ore contains a significant fraction of moisture and has a lower percentage of fines than ore that has been crushed. Thus, in the event of an accident, most of the material, although readily released, is not respirable.

The primary accident driving forces for transportation of uranium ore are those mechanical energy forces involved in any potential transportation accident. An impact accident, or collision, appears to present the only major potential for accidental release since uranium ore is not flammable and does not react energetically with water. The material is contained within the dump truck, and protected by a plastic sheeting on top of the truck hold.

19.3 TRANSPORTATION OF URANIUM ORE CONCENTRATE

Uranium ore concentrate (U_3O_8 or $(\text{NH}_4)_2\text{U}_2\text{O}_7$), also called yellowcake, is shipped from the uranium milling facility to the UF_6 conversion facility in 210-liter steel drums. Each drum contains about 360 kg of concentrate. Exclusive-use vehicles are generally employed for these shipments. Uranium ore concentrates qualify as LSA material under DOT regulations, and thus require only strong, tight, industrial packaging when shipped in exclusive-use vehicles (Rhoads 1977).

Most yellowcake is transported by truck in an enclosed trailer. In general, 50 drums may be packed in one truck shipment without exceeding current weight limitations for highway transport. This results in a net cargo weight of about 18 Mg per shipment. The average shipping distance for uranium ore concentrates is 2660 km (Geffen 1981).

19.3.1 Physical Description of the Uranium Ore Concentrate Shipping System

The uranium ore concentrate shipping system consists of fifty Department of Transportation specification 17H, 210-l steel drums carried in a van-type trailer. Each drum has a removable head consisting of a lid and bolt locking ring closure (see Figure 19.1). The sides and bottom of the drum are made of 1.3-mm (18-gauge) low carbon sheet steel and the lid is made of 14-gauge low carbon sheet steel. The bolted ring is made of 27-mm (12-gauge) low carbon steel with a 1.6 cm bolt and nut. Each drum has three rolling hoops rolled into the body, one of which is near the top curl. Inside dimensions are 0.572-m ID x 0.845-m height and external dimensions are 0.610-m OD x 0.889-m height. The gasket material for the top closure must have a minimum operating range of -40°C to 57°C . Tare weight of each drum is about 18 kg.

FIGURE 19.1. Department of Transportation Specification 17H Steel Drum

Each drum contains about 360 kg of yellowcake (uranium ore concentrate). The fifty drums are loaded, standing on end, into a conventional general commodity, van-type trailer. Drums are arranged in the trailer in two groups of twenty-five each to concentrate the weight over the trailer axles, as shown in Figure 19.2. Wood supports are nailed to the floor of the trailer as shown to prevent shifting of the drums during normal transport. Distance between the two groups of drums is about 6 m (Geffen 1981).

19.3.2 Safety-Related Information

The operating staff required for transport of uranium ore concentrate from the mine to the mill consists of one or two drivers per truck. In general, no regular maintenance operations (other than refueling) are required while the shipment is in transit.

FIGURE 19.2. Configuration of Yellowcake Drums in Semi-Trailer

The primary input requirements to uranium ore transportation involve the refined product (18 Mg), the containers for the material and the diesel fuel required to power the trucks. The major effluents from the transportation step are the airborne pollutants associated with combustion of diesel fuel (SO_x , NO_x , particulates, etc.).

Uranium ore concentrates present both a chemical and a radiological health hazard if inhaled. The chemical effects of the material are due to the toxicity of the heavy metal, uranium. The radiological effects stem from exposure to the radioactive isotopes of natural uranium. The extent of the health effects experienced with an inhalation of yellowcake in the case of a release of material is dependent on the solubility of the material. As an insoluble compound, uranium deposited in the lungs is slow to be eliminated from the body; its major effect is associated with the radiological hazard caused by continued exposure. As a soluble compound, the chemical toxicity of uranium is more evident and can result in damage to the exchange organs, especially the kidneys. Uranium ore concentrates have a specific activity of about $6.47\text{E-}4$ Ci/kg, or about 11.65 Ci/truckload. The material is a solid, in powder form.

19.4 TRANSPORTATION OF URANIUM HEXAFLUORIDE

Natural uranium hexafluoride (UF_6) is produced from uranium ore concentrates at a UF_6 purification and conversion facility. This product is then shipped to the enrichment plant where it is enriched to about 3% in the isotope ^{235}U . The two products from the gaseous diffusion enrichment plants are enriched UF_6 and depleted UF_6 . Enriched UF_6 is shipped to a UO_2 conversion plant for use in fabricating reactor fuels. Depleted UF_6 is currently stored onsite at the enrichment plant, although it may in the future be used in the fabrication of mixed oxide fuels (Rhoads 1977). The shipping systems for natural and enriched UF_6 are discussed below. Depleted UF_6 , if required to be shipped, will utilize the same shipping systems as natural UF_6 .

19.4.1 Physical Description of the Shipping System for Natural Uranium Hexafluoride

Natural uranium hexafluoride qualifies as LSA material. Packaging standards are thus set by the chemical hazard of the material rather than by its radioactive properties. Natural UF_6 is shipped in 9-Mg (10-ton) UF_6 capacity or 12.7-Mg (14-ton) UF_6 capacity steel cylinders. The UF_6 is placed in the cylinders as a liquid at about 90°C (200°F). It is cooled to room temperature to form a volatile solid before the cylinders are shipped. The cylinders may be transported by truck or rail. However, most shipments are by truck. Two 9-Mg or one 12.7-Mg cylinder can be transported on a truck trailer of standard construction. Special cradles and tie-downs are used to secure the cylinders to the transport vehicle. The transport vehicle is usually modified by permanent installation of the cradles and tie-down equipment. A protective device is placed over the cylinder filling valve to prevent damage to the valve during handling or transportation operations (Rhoads 1977). General data on the cylinders used to transport natural UF_6 are presented in Table 19.2.

The average shipping distance for natural UF_6 from the production plants to the enrichment plants is about 720 km (excluding import shipments) (Geffen et al. 1978).

TABLE 19.2. General Data for Cylinders of Natural UF_6

	<u>Model 48 x (9Mg)</u>	<u>Model 48 x (12.7Mg)</u>
Nominal Diameter	122 cm (48 in.)	122 cm (48 in.)
Nominal Length	307 cm (121 in.)	381 cm (150 in.)
Wall Thickness	1.59 cm (5/8 in.)	1.59 cm (5/8 in.)
Nominal Tare Weight	2041 kg (4,500 lb)	2359 kg (5,200 lb)
Maximum Net Weight	9539 kg (21,030 lb)	12,501 kg (27,560 lb)
Nominal Gross Weight	11580 kg (25,530 lb)	14,860 kg (32,760 lb)
Minimum Volume	3.08 m^3 (108.9 cu ft)	4.04 m^3 (142.7 cft)
Basic Material of Construction	Steel	Steel
Service Pressure	1380 kPa (200 psig)	1380 kPa (200 psi)
Hydrostatic Test Pressure	2760 kPa (400 psig)	2760 kPa (400 psi)
Isotopic Content Limit	4.5% ^{235}U Max with Moderation Control	4.5% ^{235}U Max with Moderation Control

Source: Geffen et al. 1978.

19.4.2 Physical Description of the Shipping System for Enriched Uranium Hexafluoride

Enriched UF_6 , unlike natural UF_6 , is not exempt from classification as a fissile material. Therefore it cannot be shipped as LSA material. Usually transported quantities constitute a Type B quantity of radioactive material under NRC regulations. Type B packaging in the form of protective overpacks for 2.3-Mg capacity and 9-Mg UF_6 cylinders are used to satisfy the transportation accident test criteria of 10CFR71. Provisions to adequately separate UF_6 containers, protect against accident conditions, and assure criticality prevention have been incorporated in the designs of the protective outer packagings. Protective outer packagings which have been designed and tested in accordance with applicable DOT regulations for UF_6 cylinder models 30A and 30B are shown in Figure 19.3. Enriched UF_6 is generally shipped by truck in 2.3-Mg cylinders with overpack. Five 2.3-Mg cylinders with overpack (or one 9-Mg cylinder with overpack) can be shipped on one truck trailer. A typical shipment is shown in Figure 19.4 (Rhoads 1977). General design data on the cylinders used to transport enriched UF_6 are presented in Table 19.3. The average shipping distance for enriched UF_6 from the gaseous diffusion plants to the fuel fabrication plants is 850 km (excluding shipments destined for export) (Geffen et al. 1978).

AVERAGE WEIGHTS		
SHIPPING CONTAINER WEIGHT ----	1750 LBS.	1 INCH = 2.54 cm
FILLED 30" CYLINDER	6400 LBS.	1 POUND = 454 g
TOTAL WEIGHT 8150 LBS.		

FIGURE 19.3. Horizontally-Loaded Protective Outer Package for UF_6 Cylinder Models 30A and 30B

FIGURE 19.4. Five 2.3 Mg UF₆ Cylinders in Protective Outer Packages

TABLE 19.3. General Design Data for Cylinders for Enriched UF₆

	Cylinders	
	Model 30A (2.3 MT)	Model 30B (2.3 MT)
Nominal Diameter	76.2 cm (30 in.)	76.2 cm (30 in.)
Nominal Length	205.74 cm (81 in.)	205.74 cm (81 in.)
Wall Thickness	1.03 cm (13/32 in.)	1.27 cm (1/2 in.)
Nominal Tare Weight	635 kg (1,400 lb)	635 kg (1,400 lb)
Maximum Net Weight	2245 kg (4,950 lb)	2277 kg (5,020 lb)
Nominal Gross Weight	280 kg (6,350 lb)	2912 kg (6,420 lb)
Minimum Volume	0.73 m ³ (25.65 cu ft)	0.74 m ³ (26 cu ft)
Basic Material of Construction	Steel	Steel
Service Pressure	1.32 x 10 ⁶ Pa (192 psig)	1.38 x 10 ⁶ Pa (200 psig)
Hydrostatic Test Pressure	3.45 x 10 ⁶ Pa (500 psig)	2.76 x 10 ⁶ Pa (400 psig)
Isotopic Content Limit	5.0% ²³⁵ U Max with Moderation Control	5.0% ²³⁵ U Max with Moderation Control

Source: Geffen et al. 1978.

19.4.3 Safety-Related Information

The operating staff required for transport of natural or enriched uranium hexafluoride basically consists of two drivers per truck. Loading/unloading operations are performed as part of the origin/destination facility operations. However, drivers will probably need to inspect the containers and tie-downs at refueling/rest stops.

The primary input requirements to the transportation of UF_6 involve the material itself, the containers for the material and tie-down equipment, and the diesel fuel required to power the trucks. The major effluents from the transportation activity are the airborne pollutants associated with combustion of diesel fuel (SO_x , NO_x , particulates, etc.).

Uranium hexafluoride is a highly reactive material which reacts chemically with water, ether, and alcohol to form soluble reaction products. It reacts with most organic compounds and with many metals; however, its reactivity with most saturated fluorocarbons is low. UF_6 does not react with oxygen, nitrogen, or dry air, and it is sufficiently inert to aluminum, copper, Monel nickel, and aluminum-bronze that they can be exposed to UF_6 without excessive corrosion. At room temperature, UF_6 is a white, volatile solid. At a temperature of $64^\circ C$ and a pressure of 1.52×10^5 N/Pa, UF_6 melts to form a colorless liquid of high density (Geffen et al. 1978).

When released to the atmosphere, uranium hexafluoride combines chemically with the moisture present in the air to form hydrogen fluoride (HF) gas and uranyl fluoride (UO_2F_2), a particulate. All products of a UF_6 release represent a health hazard when inhaled. Hydrogen fluoride gas presents a danger which is chemical in nature only, while the effects of inhaling UO_2F_2 are both chemical and radiological. However, for UF_6 releases involving material with an enrichment of less than 10% ^{235}U , the chemical toxicity is generally considered a far greater threat than the radiotoxicity.

Hydrogen fluoride gas causes severe irritation of the eyes and respiratory system. If sufficient quantities of the gas are inhaled, death from pulmonary edema can occur. HF damage is a function of both the concentration of gas inhaled and the duration of exposure. Uranyl fluoride presents both a chemical and radiological hazard. The chemical effects of UO_2F_2 involve the toxicity of both fluoride and uranium. As with HF gas, the greatest damage is caused by the effects of the fluoride compound on the respiratory system. Radiological effects result from exposure to the radioactive elements contained in the UF_6 , most notably the isotopes of uranium (Geffen et al. 1978).

Natural UF_6 contains 0.71% ^{235}U ; it also contains the isotopes ^{234}U and ^{238}U . Enriched UF_6 is usually enriched to 3.2% ^{235}U . The isotope inventory for enriched UF_6 is shown in Table 19.4.

The primary driving forces for accidental release of the UF_6 cargo are the mechanical energy forces involved in any transportation accident and temperature/pressure forces. Since UF_6 reacts with water, any leakage of material could result in a chemical reaction. However, the UF_6 containers would need to suffer structural failure or leakage of fittings before material could escape.

TABLE 19.4. Isotopic Composition of 3.2% ^{235}U Enriched UF_6

<u>Isotope</u>	<u>Ci/kgU</u>
^{232}U	$4.67\text{E-}5^{(a)}$
^{233}U	$2.36\text{E-}7$
^{234}U	$3.57\text{E-}3$
^{235}U	$6.92\text{E-}5$
^{236}U	$8.84\text{E-}4$
^{238}U	$3.21\text{E-}4$
^{239}Pu	$2.43\text{E-}7$
^{237}Np	$4.32\text{E-}7$
^{106}Ru	$4.22\text{E-}7$
$^{95}\text{Zr}, ^{95}\text{Nb}$	$9.3\text{E-}7$
^{137}Cs	$6.9\text{E-}8$
^{144}Ce	$6.9\text{E-}8$
^{90}Tc	$3.16\text{E-}5$
^{90}Sr	$1.72\text{E-}8$

(a) $4.67\text{E-}5 = 4.67 \times 10^{-5}$

Source: Geffen et al. 1978.

19.5 TRANSPORTATION OF PLUTONIUM OXIDE POWDER

Plutonium is separated from irradiated nuclear fuel at a reprocessing plant. Federal regulations (10CFR71) require that plutonium be shipped as a solid (usually plutonium oxide for fuel cycle use). All material to be shipped must be packaged in a separate container placed within outer packaging that meets the Type B packaging standards set forth in 10CFR71. The separate inner container must not release plutonium when the entire package is subjected to normal and accident conditions defined in 10CFR71 Appendices A and B. Shipments of plutonium in excess of 2 kg (4.4 lb) may be made only in accordance with Nuclear Regulatory Commission approved transportation plans that provide for the physical protection of special nuclear material (SNM) in transit.

Currently available packages for the shipment of plutonium have limited cargo capacities, no shielding, and limited decay heat removal capability. They are adequate for transporting first recycle plutonium (Pu separated from LWR UO_2 fuel) in quantities up to five kg (11 lb). This plutonium has a relatively low isotopic content in ^{238}Pu , which is responsible for more than 80% of the decay heat and a large fraction of the fast neutron source. However, as the number of plutonium recycles

increase, the isotopic content of ^{238}Pu in the material also increases and the currently available Pu transport package designs become inadequate to remove the decay heat and reduce surface radiation dose rates to federal guidelines. For the purposes of this study, a conceptual plutonium transport package designed for high heat generation rates and provided with gamma and neutron shielding is used. The payload of the conceptual package is significantly larger than the payloads of currently available packages.

19.5.1 Physical Description of the PuO_2 Shipping System

A packaging for the shipment of up to 32 kg (70 lb) of plutonium oxide was developed by Allied-General Nuclear Services and Battelle-Columbus Laboratories (ERDA 1976). The packaging is illustrated in Figure 19.5. The plutonium products package consists of four PuO_2 powder canisters, a primary pressure vessel for packaging and/or storage, and a Type B overpack.

Eight kilograms (17.6 lb) of plutonium oxide powder can be contained in each of the four inner canisters. Canisters are 15.2 cm (6 in.) in diameter by 30.5 cm (12 in.) long and are fabricated from thin-walled stainless steel. Canisters are provided with a filtered vent to permit the escape of water vapor and radiolytically generated gases while retaining the plutonium oxide powder. The canisters have an annular powder volume to reduce centerline temperatures.

The primary pressure vessel contains four canisters. This vessel is also provided with a vent 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.

The Type B overpack is a pressure vessel designed to the standards of Section VIII of the ASME Boiler and Pressure Vessel Code. The overpack is designed to provide containment of the plutonium oxide if the primary pressure vessel or inner canister fails for any reason. Radiation shielding is provided by the thick walls of the primary pressure vessel and overpack and by a 20-cm (8-in.) thick layer of hydrogenous neutron shielding material in the annular space between the primary vessel and overpack. The neutron shield also provides some impact protection. A fire protection shell is installed over the outer neutron shield containment wall. The fire protection shell expands away from the neutron shield containment shell when subjected to a fire environment, thus providing a significant containment barrier.

The outer dimensions of the overpack are approximately 76 cm in diameter by 183 cm high. The total package weighs about 1360 kg. Because of weight limitations, only about seven of these containers could be transported in a safeguards-equipped truck. Only truck shipments are considered in this study.

FIGURE 19.5. Package for Plutonium Dioxide

19.5.2 Safety-Related Information

This section contains safety-related information pertinent to transportation of plutonium oxide powder. This information is useful in identifying the risks associated with PuO_2 transportation.

Operating and Maintenance Requirements

Plutonium oxide is an attractive target for theft and is more accessible during transport than at other stages of the nuclear fuel cycle. Plutonium shipments are safeguarded according to the requirements of 10CFR73. These regulations specify that any shipment of strategic quantities of special nuclear materials, in this case greater than 2 kg (4.4 lb) of plutonium, are protected from sabotage and diversion by several procedures and equipment features including:

- use of only pre-planned routes and schedules
- signed acceptance of responsibility at origin, destination, and transfer routes
- tamper-indicating seals and locks on containers if the cargo compartment is not locked
- use of a specially designed truck with immobilizing features or armed drivers and a penetration-resistant cargo compartment or, alternatively, an accompanying escort car with armed drivers
- continuous radiotelephone communication capability
- call-in requirements at specified maximum intervals
- contingency plans for response actions to hijackers or saboteurs.

A conservative shipping distance of 4000 km (2500 miles) is chosen for PuO_2 shipments. This is the approximate distance from the representative fuel reprocessing facility in Barnwell, South Carolina to a potential mixed oxide fuel fabrication plant in Richland, Washington.

Inventories of Hazardous Materials

The expected properties of plutonium separated from LWR plutonium recycle fuel (second generation recycle) are presented in Table 19.5. The key isotope on the basis of heat generation and neutron generation, as mentioned previously, is ^{238}Pu . Table 19.6 shows the expected masses and curie contents for a typical truck shipment of the plutonium characterized in Table 19.5.

Containment and Shielding Systems

The plutonium products package described in this section provides the regulatory requirement for double containment of plutonium shipments in excess

TABLE 19.5. Expected Properties of Pu from LWR Pu Recycle Fuel (ERDA 1976)

Isotopic Composition	^{238}Pu - 2.44% ^{239}Pu - 47.02% ^{240}Pu - 15.51% ^{241}Pu - 8.99%
Heat Generation Rate	17.5 watts/kg Pu
Neutron Generation Rate	4.9×10^5 n/sec-kg Pu

TABLE 19.6. Plutonium Isotopic Content of a Typical PuO_2 Truck Shipment

Isotope	Mass per Shipment (kg)	Curies per Shipment ^(a)
^{238}Pu	0.78	$1.3\text{E}4^{(b)}$
^{239}Pu	15.1	$9.2\text{E}2$
^{240}Pu	8.3	$1.9\text{E}3$
^{241}Pu	5.0	$5.6\text{E}5$
^{242}Pu	<u>2.9</u>	<u>$1.1\text{E}1$</u>
Total	32	$5.8\text{E}5$
Total Fissile Content		20.1 kg
Heat Generation Rate		560 W

(a) Values listed are for gross alpha, beta, and gamma activity except for ^{241}Pu which decays essentially by beta emission only.

(b) $1.3\text{E}4 = 1.3 \times 10^4$.

of 20 Ci. Inner and outer steel pressure vessels are provided to meet this requirement. Neutron shielding is required for this package, mostly because it is used to transport plutonium with a relatively high ^{238}Pu content.

19.5.3 Alternatives to the Representative PuO_2 Shipping Container

The representative plutonium transport package described in this study is designed to haul plutonium with high heat- and neutron-generating rates. The isotopic composition of this plutonium is typical of Pu that would be shipped in the U and Pu recycle fuel cycle scenario. Plutonium currently produced is primarily first generation recycle plutonium that contains relatively small amounts of ^{238}Pu and has low heat generation and neutron production rates. Containers designed to transport this material are not adequate to transport plutonium separated from irradiated mixed oxide fuel.

Two shipping containers primarily have been used to date for transport of PuO_2 powder. These are the DOT specification 6M container and the LLD-1 container.

The specification 6M container consists of inner and outer containers separated by solid disks or rings made of industrial cane fiberboard or hardwood or plywood as shown in Figure 19.6. The outer container can be DOT 6C or 17C 40-l (10-gal) to 430-l (110-gal) steel drums with at least four covered vent holes drilled in the sides, near the top. The inner containment vessel is a carbon steel or stainless steel canister up to 13.3 cm (5.25 in.) inside diameter and no more than 15 cm (6 in.) in height. The plutonium must be packaged within one or more sealed and leaktight metal cans or polyethylene bottles within the inner container.

Up to 4.5 kg (9.9 lb) of plutonium metal, alloy, or compound or up to 13.5 kg (29.7 lb) of ^{239}Pu metal or alloy can be shipped in the 6M container. The material must be in the form of a solid that will not decompose at temperatures up to 121°C (250°F). No design features are provided for heat removal or for radiation shielding. The normal licensed heat generation limit is 10 W (34 Btu/hr).

FIGURE 19.6. DOT Specification 6M Plutonium Shipping Container

The LLD-1 shipping container is shown in Figure 19.7. This container consists of inner and outer containment vessels. The cylindrical outer vessel is supported by a cage of 1.9-cm (0.75-in.) OD steel tubing. The inner containment vessel is fabricated of 12.7-cm (5-in. schedule 80) steel pipe with 0.95-cm wall and with a threaded plug. Pu-oxide powder is contained in two No.8 steel cans that are placed inside the inner container. For transporting, the assembled cage/container is placed inside a sheet metal box 41.9 cm x 41.9 cm x 66.0 cm (16.5 in. x 16.5 in. x 26 in.).

The capacity of the LLD-1 container is limited to not more than 4.5 kg (9.9 lb) of plutonium oxide or up to 7 kg (15 lb) of Pu metal or Pu-U alloy containing not more than 7 kg (15 lb) of fissile material, any uranium enrichment. Heat removal is by convection only with a maximum heat removal capacity of about 20W (68 Btu/hr). No radiation shielding is provided.

FIGURE 19.7. LLD-1 Plutonium Shipping Container

19.6 TRANSPORTATION OF UNIRRADIATED URANIUM AND MIXED OXIDE FUEL ASSEMBLIES

Unirradiated mixed oxide and uranium-only nuclear fuel is dried, compressed into pellets, and sintered. The fuel pellets are stacked inside hermetically sealed, thin-walled tubing and banded together to form fuel elements. These elements consist of 17 x 17 arrays (for PWRs) and 8 x 8 arrays (for BWRs) of rods. The representative PWR fuel assembly shipping container is the Westinghouse RCC-1. The BWR fuel assembly shipping container used for this study is the RA-1 manufactured by the General Electric Co. The RCC-1 PWR fuel shipping container consists of an outer shell and an inner cradle assembly that clamps and positions the fuel assemblies. Two PWR assemblies can be transported inside each RCC-1 container. The RA-1 shipping container consists of a metal inner container within a rectangular wooden box. A cushioning material is provided between the inner and outer containers. The capacity of each RA-1 container is two BWR fuel assemblies. Six PWR containers or 16 BWR containers can be transported on a truck trailer. Rail shipments are not normally used for fresh fuel.

19.6.1 Physical Descriptions of the Unirradiated Uranium and Mixed Oxide Fuel Assembly Shipping Systems

This section contains descriptions of the unirradiated fuel assembly shipping systems. Separate shipping containers are described for PWR and BWR fuel bundles. It was assumed for the purposes of this study that mixed oxide fuel assemblies are shipped in the same types of shipping containers as the UO_2 fuel assemblies. It is also assumed that fresh fuel elements are shipped to the reactors by truck only.

Unirradiated PWR Fuel Shipping System

Fresh (unirradiated) PWR fuel elements are transported to the reactor in protective outer packages designed to prevent damage to the fuel elements in transit. Two PWR fuel bundles are placed inside a protective overpack designed to protect the valuable fuel bundles from damage during transport. Six overpack containers of PWR fuel can be shipped per truck trailer. Conventional tie-down techniques are used. A fresh PWR fuel shipment is pictured in Figure 19.8.

The representative fresh PWR fuel shipping container is the Westinghouse Electric Corp. RCC-1. The fuel elements are held in an inner cradle assembly that provides support for the full length of the fuel bundles during transport. The cradle assembly is mounted in a 3.2-mm (14-gauge) steel outer container. Neutron absorber plates are required for criticality control of the contents. Neutron absorber plates consist of either 0.5 cm (0.19 in.) thick, full length stainless steel containing 1.3% boron or 0.5 cm (0.19 in.) thick copper plates placed between fuel elements. This container, which has a gross weight of 3.3 Mg (7200 lb), is certified to transport Type A quantities of Fissile Class I and II materials.

The RCC-1 shipping container is not currently certified to transport mixed PuO_2 - UO_2 fuel assemblies. This packaging may have to be modified for mixed oxide fuel to include some neutron shielding in order to meet DOT requirements

FIGURE 19.8. Truckload of PWR Fuel Shipping Containers for Unirradiated Fuel

pertaining to external radiation dose levels for normal conditions for transport. Shipments of mixed oxide fuel may also require special transport equipment or procedures to guard against theft of the fuel.

Unirradiated BWR Fuel Shipping System

The representative unirradiated BWR fuel shipping container is the General Electric Co. RA-1. The shipping container consists of a metal inner container within a right rectangular wooden box. The inner container and outer box are separated by cushioning material. Two BWR fuel assemblies can be transported within the inner container. The RA-1 container is pictured in Figure 19.9.

The dimensions of the metal inner container are 29.2 cm (11.5 in.) by 45.7 cm (18 in.) by 455 cm (179 in.) long. The inner container is positioned within an all-wood or fiberglass-coated wood on aluminum outer container. The dimensions of the outer container are approximately 81 cm (32 in.) by 84 cm (33 in.) by 526 cm (207 in.) long. Cushioning is provided between the inner and outer containers by phenolic impregnated honeycomb and rigid foam. Closure is accomplished by bolts, latches, or the equivalent. A pressure relief valve is provided on the inner container and is set for 3.5 kPa (0.5 psi) differential pressure. The gross weight of the container and its contents is 1.3 Mg (2800 lb).

The RA-1 shipping container has been certified to transport mixed PuO_2 - UO_2 fuel elements, as well as UO_2 -only fuel elements. It has been certified to transport Fissile Class I and II shipments. Special safeguards and physical protection of mixed oxide fuel may be required during transport.

19.6.2 Safety-Related Information

This section contains information that will be useful in identifying the risks associated with the transportation of unirradiated nuclear reactor fuel.

Operating and Maintenance Requirements

No information pertaining to operating and maintenance requirements of the RCC-1 and RA-1 shipping containers was available. It is expected that unloading operations at a reactor are essentially the reverse of loading operations at the fuel fabrication plant. The average shipping distance for fresh fuel shipments is 2000 km (1250 miles) or the distance from the midwest to either east or west coasts.

Inventories of Hazardous Materials

The UO_2 fuel assemblies utilized for the purposes of this study are representative of the newer generation of PWR and BWR nuclear reactors. The PWR fuel assembly consists of a 17 x 17 array of rods (264 are fuel rods, 25 are non-fuel rods). A total of 461 kg (1015 lb) of uranium, of which 17 kg (37.4 lb) are uranium-235, are contained in each PWR fuel assembly.^(a) The

(a) Maximum ^{235}U enrichment is 3.65 weight percent.

FIGURE 19.9. BWR Fuel Element Shipping Container

BWR fuel assembly consists of a 8 x 8 array of rods (63 are fuel rods and one is a non-fuel rod). BWR assemblies contain about 189 kg (416 lb) of uranium, of which 9.5 kg (21 lb) are ^{235}U ^(a). Using this data, a PWR fresh fuel shipment transporting six RCC-1 packages contains about 550 kg (12000 lb) of uranium, of which 200 kg (450 lb) is ^{235}U . A BWR fresh UO_2 fuel shipment transporting sixteen RA-1 packages contains approximately 6000 kg (13200 lb) of uranium, including about 300 kg (660 lb) of ^{235}U .

Mixed oxide (MOX) fuel assemblies are expected to resemble uranium fuel assemblies. Each MOX assembly is assumed to contain up to eight percent PuO_2 by weight in natural or depleted UO_2 . Therefore, a typical MOX PWR fuel assembly contains up to 37 kg (81 lb) of PuO_2 and 424 kg (930 lb) of UO_2 . A typical BWR fuel assembly contains 15 kg (33 lb) of PuO_2 and 174 kg (380 lb) of uranium. Typical ^{239}Pu (fissile) contents are 25 kg (55 lb) and 10 kg (22 lb) for PWR and BWR MOX fuel assemblies respectively. ^(b) Based on this information, a typical shipment of fresh MOX fuel for a PWR contains about 450 kg (1000 lb) of PuO_2 , of which 300 kg (660 lb) is fissile ^{239}Pu . Typical BWR MOX fuel shipments contain about 320 kg (700 lb) of PuO_2 including about 200 kg (440 lb) of ^{239}Pu . The number of curies of radioactivity per shipment was unavailable.

Containment and Shielding Systems

Little shielding is required for the transport of fresh uranium fuel. Shipments of MOX fuel assemblies may require modest neutron shielding to reduce the surface dose rate to acceptable levels. The amount of shielding that may be required was unavailable. The PWR package (RCC-1 container) contains provisions for criticality control between the two assemblies transported in each container. Neutron absorber plates are provided for this purposes. These plates are either stainless steel with 1.3% boron added or are copper. The thickness of the plates is 0.5 cm (0.19 in.). Two plates are provided between the fuel elements. No provisions for neutron absorption are required in the RA-1 BWR fuel container. No gamma shielding is required for either shipping container.

19.7 TRANSPORTATION OF SPENT FUEL

Transportation of spent power reactor fuel occurs under different circumstances depending upon which nuclear fuel cycle scenario is being considered. For the "once-through" fuel cycle, spent fuel elements will be shipped from reactor storage basins to either an interim storage facility (if they are less than about six years after reactor discharge) or to a facility where they can be encapsulated for final disposal in a Federal repository (if they are older than about six years after reactor discharge). A spent fuel packaging facility can be associated with the reactor, the interim storage

(a) Maximum ^{235}U enrichment is 5.0 weight percent.

(b) Based on ratio of weight percent ^{239}Pu and PuO_2 of 67%.

facility, or the geologic repository. An interim storage facility for encapsulated spent fuel may also be required in case a repository is not available. At any rate, spent fuel assemblies (packaged or unpackaged) may be transported between any or all of these facilities in the once-through fuel cycle.

The other nuclear fuel cycle considered in this study is the "U and Pu recycle" scenario. For this fuel cycle, spent fuel is transported either from the reactor to an interim spent fuel storage facility, and then to a fuel reprocessing plant (FRP), or from the reactor directly to the FRP. A sub-alternative consists of shipping spent fuel to an interim storage facility that is co-located with the FRP. The interim storage facility could be designed to feed spent fuel to the head end of the FRP via underwater canals, thereby eliminating one potential spent fuel transportation step.

Irradiated nuclear fuel has been shipped in Type B containers in the United States for many years. Currently available casks are massive, heavily shielded and are designed to accept unpackaged fuel assemblies. If shipment of overpacked (encapsulated) fuel is required, existing cask designs would have to be modified to accept the additional length and diameter of encapsulated fuel. Alternatively, new casks could be designed to accept packaged spent fuel.

The primary sources of the information contained in this section are PNL-2588 (Elder 1978) and PNL-2682 (Elder 1981).

19.7.1 Physical Descriptions of the Spent Fuel Shipping System

Two spent fuel shipping systems are described in this section. The first is a legal-weight truck shipping system utilizing the NAC-1 (NFS-4) shipping cask. The other is a rail system that utilizes the IF-300 shipping cask.

Truck Shipping System

The representative truck spent fuel shipping system consists primarily of the NAC-1 (NFS-4) shipping cask, a truck trailer, and cask peripheral equipment. The NAC-1 shipping cask is water filled and air-cooled and is designed to transport either 1 PWR or 2 BWR spent fuel assemblies. The cask is designed to handle the large size and high burnup of present generation reactor fuel. The approximate loaded cask weight is 23.6 Mg (26 tons), which can be transported across country without restriction as to legal weight limits. The total shipment weight, including cask, spent fuel, trailer, and peripheral equipment is approximately 33.3 Mg (73,280 lb). Figure 19.10 shows a cut-away drawing of the NAC-1 shipping cask. Figure 19.11 shows a schematic drawing of the cask and trailer configuration.

The cask has an overall length of 5.44 m (214 in.) and a diameter of 0.96 m (38 in.). Dimensions of the internal cask cavity are 4.52 m (178 in.) in length and 0.34 m (13.5 in.) in diameter. Interchangeable fuel baskets support the spent fuel assemblies inside the inner cavity. Two types of fuel baskets can be used; one has the capacity for one PWR fuel assembly, the other is for two BWR fuel assemblies. The fuel assembly is supported by four removable 2.5-cm (1-in.) support plates. The support plates rest on support tabs

FIGURE 19.10. Cut-Away View of the NAC-1 Spent Fuel Shipping Cask

permanently attached to the cavity interior. Table 19.7 presents the important characteristics of the NAC-1 (NFS-4) spent fuel cask.

The primary or inner cask cavity consists of a nominal 0.8-cm (0.31-in.) thick lead gamma shield and a stainless steel penetration barrier 3.2-cm (1.25-in.) thick. A cross-section of the NAC-1 cask is shown in Figure 19.12. A compartmentalized tank, 11.4 cm (4.5 in.), thick surrounds the outer shell. This tank contains a borated water-antifreeze solution to provide neutron shielding. The neutron shield consists of an annular region, as shown in Figure 19.12, that is comprised of four 90° isolated compartments. The nominal operating pressure in the neutron shield is assumed to be 630 kPa (90 psig). Pressure relief valves are set at 690 kPa (100 psig) and are located midway along the length of the shield. An expansion chamber for the shield tank accommodates temperature changes of the borated water-antifreeze solution and assures uniform water shielding thickness at all times.

FIGURE 19.11. Schematic Drawing of the Cask/Trailer Configuration

TABLE 19.7. Characteristics of the NAC-1 (NFS-4)
Spent Fuel Shipping Cask

Cask

Cask Length	544 cm (214 in.)
Cavity Diameter	34.3 cm (13.5 in.)
Inner Shell Thickness	0.79 cm (0.312 in.)
Lead Shield Thickness	16.8 cm (6.625 in.)
Outer Shell Thickness	3.2 cm (1.25 in.)
Water Jacket Thickness	11.4 cm (4.5 in.)
Thickness of Outer Shell of Water Jacket	0.27 cm (0.105 in.)

Fuel Support and Shield Plates

Material	Stainless Steel
Thickness	2.54 cm (1.0 in.)
Approximate Distance from Cavity Wall	1.27 cm (0.5 in.)

FIGURE 19.12. Cross-Section of the NAC-1 Shipping Cask

The inner cavity of the cask is filled with water coolant. Water transfers decay heat from the spent fuel elements by convection to the inner cavity wall. Heat is conducted through the inner shell, lead gamma shield, and outer shell to the neutron shield where it is transferred by convection to the cask outer wall. Heat is then conducted through the outer wall and rejected to the atmosphere by convection and radiation. No active cooling is used. The maximum heat rejection capacity of the cask is 11.5 kW (about 39,000 Btu/hr). Maximum transport conditions for the inner cavity [i.e. 54°C (130°F) direct sunlight, maximum fuel burnup (40,000 MWd/MgU), minimum fuel cooling period (120 days after reactor discharge), and still air] are 174°C (345°F) and 1050 kPa (150 psig). The primary cavity is designed to withstand temperature and pressure conditions of 278°C (532°F) and 6900 kPa (984 psig) under the fire accident condition (i.e., 1/2 hr at a temperature of 800°C (1475°F)).

The cask has a single lid which seals and shields the cask cavity. The lid is stainless steel and is attached to the cask by six high-strength bolts. Two teflon® O-rings provide the head seal. They are arranged so that each may be pressure tested. Nominal cavity pressure is 1220 kPa (175 psig). A rupture disk is provided that vents the cavity to the atmosphere when cavity pressure exceeds 7720 kPa (1100 psig).

The bottom end of the cask consists of a 20-cm (7.9-in.) thick disc of stainless steel welded to the main 3.2-cm (1.25-in.) cask shell and the inner cavity shell. The top end of the cask consists of a stainless steel ring flange welded to the cask shells in a similar fashion. There are two 1.3-cm (0.5-in.) drain holes in the bottom end of the cask. The drain holes open to the end of the cask cavity and are drilled through the gamma shield to ball valves imbedded in the outer perimeter. Access tubes are provided through the lower impact limiter (discussed below) and are sealed with port covers and O-ring seals. A vent valve similar to the drain valves is also provided in a counterbore in the cavity flange. A 1.3-cm (0.5-in.) diameter rupture disc is buried in the cavity flange and connected to the cask cavity. The rupture disc assembly is mounted in a counterbore sealed by a cap and also has a safety relief valve for backup.

Impact limiters provide necessary crash protection. Both ends of the cask are provided with side-mounted, ring-type impact limiters permanently located at both ends of the cask. These provide crash protection in side-on or horizontal impact. They are constructed of balsa wood and encased in stainless steel. The bottom impact limiter consists of a 0.76-m ID x 1.27-m OD (30 in. ID x 50 in. OD) annular structure with eight 0.95-cm (0.38-in.) thick radial steel gussets as shown in Figure 19.13. The eight cavities are filled with balsa wood. The lower impact limiter is 40.6 cm (16 in.) long and also serves as an expanded base when the cask is set vertical.

The upper impact limiter is used to protect the cask head. The upper and lower impact limiters are of similar design except the upper one is only 30.5 cm (12 in.) long and is removable. This limiter is normally removed and stored

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FIGURE 19.13. Ring-type Impact Limiter Details

on the transport vehicle during normal unloading operations. Vent and drain valves, pressure test connections, and relief valves are buried within the impact limiter and special structure for protection against accident conditions.

Sacrificial impact limiters exist on both ends of the cask to absorb end-on impact energy. A balsa wood impact limiter is located on the top or lid end and is assumed to be clad in light gauge sheet metal. Four 3.2-cm (1.25-in.) bolts are used to attach this impact limiter to the cask lid.

A disc of balsa wood is also located in a cavity internal to the bottom ring-type impact limiter at the bottom of the cask. A 0.95-cm (0.38-in.) thick stainless steel shell, which serves as an internal surface of the bottom ring impact limiter, surrounds the balsa wood disk.

Two sets of trunnions are used for normal cask handling and transport tie-down purposes. The upper set, attached to the upper impact limiter, is used for lifting the cask in conjunction with a special "swing-arm" type yoke. This yoke is normally permanently locked to the lifting trunnions throughout the complete handling cycle at the reactor, reprocessing site, interim storage facility, or spent fuel repository. The lower trunnions are offset to provide a gravity pivot from the vertical loading and unloading position to the horizontal transport mode.

Rail Shipping System

The representative rail spent fuel shipping system consists primarily of the General Electric Co. IF-300 rail shipping cask, a railcar, and cask peripheral equipment. The cask is normally transported by rail on a 100-Mg

(110-ton) capacity, four-axle flatcar. The cask is mounted on a skid and supported on a saddle at the head end and a cradle at the bottom end. The IF-300 also may be transported short distances on a special nine-axle truck requiring special overweight permits. This special truck is used to move the cask from reactors without rail facilities to the nearest rail siding. A view of the shipping cask, skid, and rail flatcar is shown in Figure 19.14. The cask weight when loaded is between 61.8 Mg (68 tons) and 63.6 Mg (70 tons) depending upon the particular type of fuel being shipped. The cask cooling system is mounted on the skid and these two components together weigh approximately 20.5 Mg (22.5 tons).

The IF-300 shipping cask is designed to ship 18 BWR or 7 PWR fuel elements. The two types of fuel assemblies are accommodated through utilization of interchangeable fuel baskets that are inserted into the inner cask cavity and through two different length closure heads. One basket and one closure head accept BWR spent fuel and the others accept PWR fuel. The cask has an overall length of 5.33 m (210 in.) and a diameter of 1.62 m (64 in.). The internal cavity measures 4.58 m (180.25 in.) long and has a diameter of 0.95 m (37.5 in.). Figure 19.15 shows a cut-away drawing of the IF-300 cask. Table 19.8 summarizes the characteristics of the representative rail cask.

All external and internal surfaces of the cask are stainless steel. Gamma and neutron shielding are provided by water in the cask cavity, depleted uranium metal within the cask shell, and an exterior water-filled enclosure. The exterior shielding water enclosure is fabricated from thin-walled stainless steel and is corrugated to maximize the heat transfer area. These corrugations also significantly increase the strength of the outer jacket and its resistance to damage. This cylindrical containment is attached to the cask body and masks the active fuel zone. The cask cavity may be shipped dry (air coolant) under low fuel decay heat conditions.

Removable, slotted, stainless steel baskets are provided to contain the fuel assemblies within the internal cask cavity. Two alternative baskets are available; one designed to accommodate PWR assemblies and the other for BWR assemblies. Spacers are mounted on the inside of the cask closure head for axial restraint of the fuel elements. Nine disk spacers are mounted along the fuel basket using standard grapples to center the basket within the cask cavity.

The cask inner cavity is a stainless steel cylinder, 95.2 cm (37.5 in.) ID with 1.27 cm (1.5 in.) thick walls. The bottom is sealed with a 3.8 cm (1.5 in.) thick stainless steel plate. The upper end is welded to the closure flange.

Surrounding the inner cavity is the depleted uranium metal shielding material. This heavy metal assembly consists of annular castings, each with a 97.8-cm (382-in.) ID and a 10.2-cm (4-in.)-thick wall. Sections are interlocked, end-to-end, using an overlapping joint which holds the stack together and prevents radiation streaming. This assembly is shrink-fitted to the inner cavity to ensure good thermal contact for heat transfer purposes. The bottom end shield is a 9.5 cm (3.75 in.) thick uranium metal casting.

FIGURE 19.14. IF-300 Irradiated Fuel Shipping Cask

FIGURE 19.15. Cut-Away View of the IF-300 Cask

TABLE 19.8. Characteristics of the IF-300 Shipping Cask and Spent Fuel

	BWR	PWR
Fuels		
Exposure, GWd/MgU (average)	35.0	35.0
Operating Power, kW/kgU	30.0	40.0
Assembly Heat Power, W/kgU	19.70	24.03
Assemblies per Cask Load	18	7
Uranium, kgU/Assembly	197	475
Maximum Decay Heat Rate-Cask, kW	238,000	262,000
Cask		
Cavity Length	475.2 cm (180 in.)	429.9 cm (160.2 in.)
Cavity Diameter	95.2 cm (37.5 in.)	95.2 cm (37.5 in.)
Inner Shell Thickness	1.27 cm (0.5 in.)	1.27 cm (0.5 in.)
Shielding Thickness	10.2 cm (4.0 in.)	10.2 cm (4.0 in.)
Outer Shell Thickness	3.8 cm (1.5 in.)	3.8 cm (1.5 in.)
Cask Length, Excluding Fins	981.5 cm	482.9 cm
Cask Cavity Water	2.31 Mg (5,100 lb)	2.09 Mg (4,800 lb)
Shielding Water	2.06 Mg (4,540 lb)	2.06 Mg (4,540 lb)
Cavity Relief Pressure	2450 kPa (350 psig)	
Exterior Containing Relief Pressure	1400 kPa (200 psig)	
Cooling System		
Cooling Medium	Forced Air	Forced Air
Number of Full-Length Cooling Nozzles	4	4
Cooling Air Arrival Velocity	14.3 m/s	14.3 m/s
System Classification	Open	Open
Surface Film Coefficient	7.0	7.0
Minimum Air Flow	280 m ³ /min	280 m ³ /min

The outer body shell is also a stainless steel cylindrical vessel with a 118.1 cm (46.5 in.) ID and a 121.9 cm (48 in.) OD. This shell is also shrink-fitted to the depleted uranium shield to ensure good heat transfer characteristics.

The cylindrical portion of the cask is encircled by a thin-walled corrugated stainless steel water jacket. The water jacket extends axially from the upper valve box (see Figure 19.15) to a point slightly above the cask bottom, thus masking the active fuel zone. Neutron shielding is provided by water in this structure. Antifreeze may be added to the water when the cask is subject to cold ambient temperatures. The surface of the water jacket is corrugated for heat transfer purposes. Additionally, the use of continuous corrugation provides an exterior containing relief pressure rating of 1380 kPa (200 psig).

Four 17.8 cm (7 in.) high and 3.2 cm (1.25 in.) thick circumferential fins are welded to the outer shell and serve as lifting rings and impact limiters. They are also used to support the water jacket sections. The cask is lifted by a set of trunnions located just below the closure head flange. These lifting trunnions are pinned to the upper set of heavy rings and are designed to be removed for transit. The upper set of lifting rings is also the forward support/axial restraint when the cask is in the horizontal transport position. Thirty-two stainless steel impact fins are radially mounted at the lower end of the cask. These fins are 3.2 cm (1.25 in.) thick and about 20.3 cm (8 in.) high.

The cask can be equipped with either of two different heads to match the different cavity lengths required for the two types of fuel assemblies. The overall cavity length is 430.5 cm (169.5 in.) with the short head in place and is used for most PWR fuel. The long head increases the cavity length to 457.2 cm (180 in.) and is used for BWR fuel and some longer designs of PWR fuel.

Shielding in the cask head is provided by 7.6 cm (3 in.) of depleted uranium. The outer shell and flange is a single stainless steel machine casting. A circular stainless steel plate is welded in place to form the head cover. The closure head is sealed with a metallic gasket designed for a minimum burst pressure of 4200 kPa (600 psig) and a maximum operating temperature of 815°C (1500°F). The maximum normal operating pressure for the cask cavity is 1400 kPa (200 psig) and the design working pressure is 2800 kPa (400 psig) at a material temperature of 435°C (815°F). A pressure relief valve, set to open at 2600 kPa (375 psig), is provided for protection against overpressure. The valve is set for a maximum steam or gas blowdown of 5% and a liquid blowdown of 10%. The cask cavity is equipped with two nuclear service valves, one in each of two valve boxes (see Figure 19.15).

Each closure head has 32 radially mounted fins on the end, 16 of which protrude 24.1 cm (9.5 in.) from the surface. These fins are stainless steel, welded in place, and provide impact protection for the cask and contents.

The cask is supported in two locations while it is being transported in a horizontal configuration. A saddle supports the cask just below the closure flange and a pivot cradle supports the cask base. The saddle and cradle are welded directly to the skid frame. The U-shaped saddle supports the cask at its upper lifting rings. Hardened pins are inserted through the lifting rings to provide vertical and lateral restraint of the head end. Axial restraint for the total cask weight is provided by the ears of the saddle. The pivot cradle consists of two pedestals and a counterbalanced cradle that support the cask. The cradle pivots between the pedestals on two trunnions.

The equipment skid functions as a unitized pallet for the cask and cooling equipment. The skid frame uses 70 cm (24 in.) fabricated I-beams and incorporates fuel tanks for the cooling system diesel engines. The skid is 11.43 m (37.5 ft) long, 2.44 m (8 ft) wide and is all-steel construction. During transport, the skid is restrained by a specially designed securing system.

Dissipation of decay heat from the irradiated fuel to the cask cavity walls is by natural circulation of the contained water, by conduction through the cask walls to the outer neutron shield, and by convection through the neutron shield to the corrugated outer wall where heat is radiated and convected to the atmosphere. The cask cooling system blows air at a velocity of 14.3 m/s (47 fps) in a direction perpendicular to the cask surface. Air is blown through four ducts, running the length of the cask and 90 degrees apart, bisecting the four quadrants. Cooling air is supplied by two blowers driven by air-cooled diesel engines. The maximum heat removal capacity of the cask is 76 kW (260,000 Btu/hr). Under low fuel decay heat conditions, the internal cask cavity may be shipped dry (air coolant).

The fuel tanks located in the skid have a total storage capacity of 2.16 m³ (570 gal) of diesel fuel oil. This quantity permits continuous operation of both blower units for ten days. Either unit is capable of supplying sufficient air to cool the cask surface.

Both blowers discharge into a common air plenum feeding the four axial ducts that cool the cask. If one blower fails, a gravity damper prevents a back-flow of air from the plenum.

19.7.2 Safety-Related Information

This section contains important safety-related information that is useful to help identify the risks associated with the transportation of spent nuclear power reactor fuel.

Operating and Maintenance Requirements

Because spent fuel might be the target of theft or sabotage while it is being transported, NRC regulations require measures to assure the physical protection of the shipment during transit. Spent fuel has some physical characteristics that make it an undesirable target for theft. These include

high levels of radioactivity, size, weight (0.23 to 0.45 Mg per assembly), and relatively low plutonium content per assembly. In addition, the massive shipping casks provide substantial protection. Physical protection measures could reduce the frequency of accidents involving spent fuel shipments as well as enhancing the response to a potential accident. An average shipping distance for spent fuel is approximately 900 km (560 miles) (Elder 1981).

Design basis limitations are placed on the amount and type of spent fuel that each cask can transport. The NAC-1 fuel limitations include a maximum fuel burnup of 40,000 MWd/MgU, a maximum decay heat load the cask can dissipate of 11.5 kW, and a minimum fuel decay period of 120 days after discharge from the power reactor. The maximum enrichment for the fuel is 3.6 and 3.0 weight percent ^{235}U for PWR and BWR fuel, respectively. For the IF-300, the design bases for the spent fuel are an average exposure of 35,000 MWd/MgU and a maximum decay heat load (for PWR assemblies) of 76 kW (the licensed decay heat load is 62 kW). Maximum enrichments are assumed to be similar to those reported for the NAC-1 truck cask.

All shipping casks have stainless steel exterior surfaces that facilitate decontamination. Decontamination is accomplished by high pressure steam, deionized water, and decontamination solutions. Decontamination solutions could include special detergent solutions or acidic and alkaline solutions (e.g., 5% HNO_3 or 5% NaOH). Some scrubbing with brushes or rags may be required to loosen contaminants from the cask.

Inventories of Hazardous Materials

The representative spent fuel shipping casks have maximum capacities of 1 PWR or 2 BWR fuel assemblies (NAC-1 truck cask) and 7 PWR or 18 BWR fuel assemblies (IF-300 rail cask). Therefore, based on PWR fuel assemblies for the NAC-1 and BWR fuel assemblies, for the IF-300, the casks contain a maximum of 0.46 MgHM and 3.40 MgHM, respectively. The maximum radionuclide inventories of the spent fuel in the casks is shown in Table 19.9, based on these quantities. Included in this table is the maximum inventory of diesel fuel oil for the cooling system.

Containment and Shielding Systems

Spent nuclear fuel shipping casks require substantial shielding to reduce the dose rate at the surface of the cask to less than or equal to 200 mR/hr. The NAC-1 truck cask includes 17 cm (6.63 in.) of lead for gamma shielding, 11.4 cm (4.5 in.) of water for neutron shielding, and 4.3 cm (1.7 in.) of stainless steel. The IF-300 rail cask includes 10.2 cm (4 in.) of depleted uranium for gamma shielding, a 7.6 cm (3 in.) water neutron shield, and 8.9 cm (3.5 in.) of stainless steel.

Spent fuel contains quantities of plutonium in excess of 20 Ci and therefore requires double containment to the cargo in transit. However, spent fuel is exempt from the double containment rule because the fuel cladding qualifies as a level of containment.

TABLE 19.9. Inventories of Hazardous Materials for the Representative Spent Fuel Shipping Systems

Hazardous Material	Form		Inventory	
	Chemical	Phase	NAC-1	IF-300
Spent Fuel ^(a,b)	UO ₂	S	0.46 MgHM	3.40 MgHM
Activation Products			6.0E+3 Ci	4.4E+4 Ci
Noble Gases			4.1E+3 Ci	3.0E+4 Ci
Iodine			1.6E-2 Ci	1.2E-1 Ci
Tritium			2.1E+2 Ci	1.5E+3 Ci
Semi-volatile fission products			3.2E+5 Ci	2.4E+6 Ci
All other fission products			1.2E+6 Ci	8.8E+6 Ci
Uranium and lower actinides			3.0E+0 Ci	2.2E+1 Ci
Transuranics			1.1E+5 Ci	8.2E+5 Ci
Total			1.6E+6 Ci	1.2E+7 Ci
Diesel Fuel Oil	Hydrocarbons	L	0	2.16 m ³

- (a) This information was extracted from the radionuclide decay data tables found in DOE 1979 (Tables 3.3.6, 3.3.9 and 3.3.15).
- (b) Radionuclide inventories were calculated assuming spent fuel characteristics for the U and Pu recycle scenario, fuel that has an average burnup of 29,000 MWd/ MgU, and fuel that has decayed 180 days after reactor discharge.

For both casks, spent fuel assemblies are contained within an inner, mechanically sealed cavity. A second containment boundary is provided by the cask outer shell. The cask lids are bolted and sealed by O-rings. Nominal cavity pressures are 1220 kPa (175 psig) and 2450 kPa (350 psig) for the NAC-1 and IF-300, respectively.

Spent fuel shipping casks are required by regulations to be able to withstand severe hypothetical accident conditions without:

1. A reduction of shielding that would be enough to increase the radiation dose rate at 0.9 m (3 ft) from the external surface of the cask to more than 1000 mR/hr.
2. A radioactive material release.

19.7.3 Alternative Spent Fuel Shipping Systems

Several other light water reactor (LWR) spent fuel shipping casks have been licensed and fabricated for use in the United States. Some important characteristics of these casks are presented in Table 19.10.

**TABLE 19.10. Characteristics of Alternative Spent LWR
Fuel Shipping Casks**

Common Name	TN8	TN9	NLI-1/2	NLI-10/24
Supplier	Transnuclear Inc.	Transnuclear Inc.	NAC ^(a)	NAC ^(a)
Assemblies (PWR/BWR)	3/0	0/7	1/2	10/24
Shielding	Pb/ST	Pb/ST	Pb/U/ST	Pb/U/ST
Cooling Requirements (in transit)	Air (natural convection)	Air (natural convection)	Air (natural convection)	Air (natural convection)
Inner cavity coolant	Air	Air	Helium	Helium
Maximum heat removal (kW)	35.5	24.5	10.6	70
Loaded cask weight (Mg) ^(b)	36.4	36.4	21.8	91
Overall dimensions (cm) ^(c)	172 cm dia x 488 cm long w/o covers	172 cm dia x 513 cm long w/o covers	102 cm dia x 490 cm long	224 cm dia x 519 cm long
Transport mode	Overweight Permit (truck)	Overweight Permit (truck)	Truck	Rail

(a) Current supplier is Nuclear Assurance Corp. (NAC). These two casks were originally supplied by National Lead Industries, Inc. (NLI).

(b) 1 Mg = 1.1 tons

(c) 1 cm = 0.39 in.

Another alternative for shipment of spent LWR fuel may include shipping consolidated fuel. Consolidation consists of disassemblage of the fuel assemblies and placement of loose fuel rods into canisters. The consolidated fuel is then placed into casks for shipment. Currently, no spent fuel casks are licensed for transportation of consolidated fuel. However, one cask supplier (Nuclear Assurance Corp.) intends to submit an application to the NRC to license one or more of their casks for shipment of consolidated fuel.

19.8 TRANSPORTATION OF SOLIDIFIED HIGH-LEVEL WASTE

High-level liquid waste is the primary waste generated when spent fuel is reprocessed. It will be converted to a solid form (borosilicate glass) and shipped to a geologic repository or to an interim storage facility if the repository is not operational. Shipping containers designed specifically for solidified high-level waste (SHLW) have not been built. These containers are expected to resemble the current generation of spent fuel shipping casks. The cask for transporting SHLW will be required to satisfy the criteria for Type B packaging.

Current Federal regulations (10CFR50, Appendix F) stipulate that high-level liquid radioactive wastes be converted to a dry solid not later than five years following separation of the fission products from the irradiated fuel. The solidified waste must be shipped to a repository for disposal not later than ten years after the waste was separated from the fuel. It may be shipped to an interim storage facility if a repository is not available.

19.8.1 Physical Description of the SHLW Shipping System

Solidified high-level waste will be contained in a disposable, monolithic canister with welded closures. The canister is assumed to be constructed of stainless steel between 6 and 12 mm (0.24 and 0.48 in.) thick. The overall dimensions of the conceptual canister are a diameter of 30.5 cm (12 in.) and length of 3.1 m (10 ft). The canister is assumed to contain the quantity of SHLW generated by reprocessing 2.5 MgHM. The combined weight of the SHLW and canister is about 1.0 Mg (1.1 ton).

Conceptual high-level waste rail and truck casks are described in the following subsections. The conceptual railcask described is based on a design published by Battelle Pacific Northwest Laboratory (Peterson and Rhoads, 1978). A truck cask is also described based on the NAC-1 spent fuel truck cask design. Conservative assumptions are made as necessary where spent fuel truck cask design parameters are modified to fit SHLW truck cask requirements.

Rail Shipping System

The conceptual high-level waste rail cask is a lead-filled, double-walled, stainless steel cylinder weighing approximately 100 Mg (110 ton). The cask has a capacity of nine canisters of SHLW. The conceptual cask is cooled by natural convection from the cooling fins on the cask body. The maximum heat removal capability of the cask is 50 kW. The cask is shipped on a conventional 6-axle depressed-center flat railcar that has been modified by the permanent installation of the cask tie-down system. Figure 19.16 shows an artist's representation of the conceptual cask/railcar system. Important design parameters of the conceptual cask are shown in Table 19.11.

Details of the cask structure and internal design features are shown in Figure 19.17. The cask body is 4 m (13 ft) long, with an overall length, including cooling fins, of 4.4 m (14.3 ft). The cask cavity is 1.5 m (5 ft) in diameter and the outside diameter with the cooling fins is 2.5 m (8 ft). The inner cask cavity contains a solid aluminum insert that positions the waste canisters and transfers radioactive decay heat from the canisters to the main cask body. Each SHLW canister will be held in place by a three-part positioner that is pressed against the canister by leaf springs. An aluminum honeycomb impact absorber is placed at each end of the canister when it is loaded into the cask. The SHLW cask is designed to be shipped dry (i.e., water is not required in the cask cavity for heat transfer).

The cask body is constructed of a series of concentric shells. The inner shell is constructed of 1.9 cm (0.75 in.) thick stainless steel. A

FIGURE 19.16. Conceptual Solidified High-Level Waste Shipping Cask and Railcar

TABLE 19.11. Important Design Parameters for the SHLW Rail Shipping Cask^(a)

Overall cask length	4.4 m (14.3 ft)
Overall cask diameter	2.5 m (8 ft)
Length of cask cavity	3.3 m (10.7 ft)
Diameter of cask cavity	1.5 m (5 ft)
Capacity	9 canisters of SHLW
Heat removal capability	50 kW
Weight	
Loaded cask	100 Mg (121 ton)
Gross vehicle weight ^(b)	160 Mg (176 ton)
Inner shell	
Material	Stainless steel
Thickness	1.9 cm (0.75 in.)
Gamma shield	
Material	Lead
Thickness	20 cm (8 in.)
Neutron shield	
Material	Borated water
Thickness	10 cm (4 in.)
Outer shell	
Material	Stainless steel
Thickness	5 cm (2 in.)
Impact Absorbers	
Internal	Honeycomb
External	Steel fins

(a) PNL-2244 (Peterson and Rhoads 1978).

(b) Includes weight of loaded cask, railcar, and associated tie-downs and peripheral equipment.

20 cm (7.9 in.) lead gamma shield surrounds the inner shell. The gamma shield is contained by a 5-cm (2-in.) thick stainless steel structural wall. A 10-cm (3.0-in.) thick borated water neutron shield surrounds the outer structural wall. The neutron shield is a compartmentalized structure comprised of four 90° isolated compartments. Each compartment is connected to a 0.1-m³ (26.4-gal) expansion tank (not shown in the Figures). Each compartment is also provided with a pressure relief valve that prevents the neutron shield water pressure from exceeding 525 kPa (75 psig). Ethylene glycol is added to the shield water to prevent freezing when the cask is shipped empty under cold

FIGURE 19.17. Details of the Conceptual SHLW Cask Structure and Internal Design Features

ambient conditions. A 1.2-cm (0.5-in.) thick stainless steel outer wall and a series of radial cooling fins 1.2 cm (0.5 in.) thick and 10 cm (3.9 in.) high complete the cask body.

A two-piece stainless steel cask lid is provided. The inner lid of the conceptual SHLW rail cask contains a 12.7-cm (5.0-in.) thick depleted uranium gamma shield. Depleted uranium metal also provides additional strength in the lid if the cask is involved in an accident. The outer lid contains a 7.6-cm (3.0-in.) thick layer of solid hydrogenous material for neutron shielding. Stainless steel fins on the outer lid function as impact absorbers in an accident. Each lid is attached to the cask cavity flanges by twenty 5-cm (2-in.) diameter stainless steel bolts. A metallic O-ring gasket is used to seal the outer lid to the cask body. The cask bottom is not removable (i.e., it is welded to the cask body) but is otherwise similar to the lid. Four 18-cm (7-in.) diameter trunnions are mounted on the cask. The trunnions extend 14 cm (5.5 in.) from the cask body and are used to support the cask during handling and shipping.

The cask tie-down equipment is permanently mounted on an exclusive-use railcar. The tie-downs consist of four columns that support the cask in a horizontal position for shipping. The trunnions on the cask body rest on the support columns. The cask is clamped in place at each end by a steel, semicircular, 2.5-cm (1-in.) thick retaining ring that is bolted to the top of the support columns. Energy-absorbing impact fins are welded to the steel rings. An expanded metal cage covers the cask during shipment to keep debris off the cask and to prevent people from touching the heated outer surface of the cask. The cage is constructed in two pieces that slide back onto the car deck during cask loading and unloading operations.

Truck Shipping System

The SHLW truck shipping cask described in this section is based on the SHLW rail cask and the NAC-1 spent fuel truck cask. This cask is assumed to be a lead-filled, double-walled, stainless steel cylinder that can be shipped as a legal weight truck system. This cask will have a capacity of one canister of SHLW. Heat removal is accomplished by natural convection and radiation of radioactive decay heat away from the cask external surface.

The basic cask design is assumed to resemble the NAC-1 spent fuel truck cask. The cask will consist of concentric stainless steel shells. A layer of lead gamma shielding will be sandwiched between the inner and outer shells. A four-compartment borated-water neutron shield tank surrounds the outer cavity. Expansion tanks are provided for each compartment to accommodate temperature changes of the shielding water and assure a uniform shielding thickness. Table 19.12 shows the design parameters for the conceptual SHLW truck cask. Refer to the discussion on the NAC-1 spent fuel truck cask for additional information.

TABLE 19.12. Design Parameters for the Conceptual SHLW Truck Cask^(a)

Overall cask length	5.4 m (2.4 in.)
Overall cask diameter	0.66 m (26 in.)
Length of cask cavity	4.5 m (178 in.)
Diameter of cask cavity	0.34 m (13.5 in.)
Capacity	1 canister of SHLW
Inner shell	
Material	Stainless steel
Thickness	0.8 cm (0.3 in.)
Gamma shield	
Material	Lead
Thickness	16.8 cm (6.6 in.)
Outer shell	
Material	Stainless steel
Thickness	3.2 cm (2 in.)
Neutron shield	
Material	Borated water
Thickness	11.4 cm (4.5 in.)
Outer surface	
Material	Stainless Steel
Thickness	0.3 cm (0.1 in.)

(a) Design parameters for the conceptual SHLW truck cask are the same as those for the NAC-1 spent fuel truck cask.

19.8.2 Safety-Related Information

This section contains important safety-related information pertaining to transportation of SHLW. This information is useful to determine the risks of transporting this material.

Operating and Maintenance Requirements

Solidified high-level waste is not a likely target of theft or sabotage while it is being transported. The physical characteristics of SHLW, including strong radioactivity, low plutonium content and large size and weight, make it an undesirable target. In addition, the massive shipping casks offer substantial protection. Other information pertaining to the accessibility of large quantities of radioactive material during transport is contained in 10CFR50.

Inventories of Hazardous Materials

This section contains the inventories of radioactive materials in the truck and rail SHLW shipping casks. The rail cask contains a maximum of nine of the representative SHLW canisters described in Section 16.B. The SHLW truck cask capacity is one representative SHLW canister. Radionuclide inventories of the SHLW casks used for this study are shown in Table 19.13.

Containment and Shielding Systems

SHLW shipping casks require substantial shielding to reduce the surface dose rate to below 200 mR/hr. The conceptual rail cask has 20 cm (8 in.) of lead gamma shielding and 10 cm (4 in.) of borated water neutron shielding. The conceptual truck cask has 16.8 cm (6.6 in.) of lead shielding and 11.4 cm (4.5 in.) of water shielding.

TABLE 19.13. Radionuclide Inventories of the Conceptual HSLW Truck and Rail Casks^(a)

Radionuclide	Curies/Cask	
	Truck	Rail
Iodine	8.8E-4	7.9E+3
Semi-volatile fission products	2.05E+5	1.8E+6
Tritium	6.5E+2	5.9E+3
Other fission products	4.6E+5	4.1E+6
Uranium plus lower actinides	2.8	2.5E+1
Transuranic nuclides	<u>2.2E+45</u>	<u>2.0E+5</u>
Total	6.9E+5	6.2E+6

(a) Based on the radionuclide inventory of the conceptual SHLW canister shown in Table 16.B.1; 1 canister/truck cask, 9 canisters/rail cask.

19.9 TRANSPORTATION OF CONTACT-HANDLED TRANSURANIC WASTE

Transuranic wastes (TRUW) are produced in nuclear fuel cycles involving plutonium and other elements heavier than uranium. TRUW are defined as wastes contaminated with alpha-emitting TRU radionuclides, with the level of contamination exceeding ten nanocuries per gram^(a). These radionuclides generally have long half-lives and high radiotoxicity. These wastes are divided into two general categories: contact-handled (CH) and remotely-handled

(a) It was recently proposed to raise this figure to 100 nanocuries per gram.

(RH). The transportation system for hauling CH-TRUW is discussed in this section. The representative RH-TRUW transportation system will be discussed in Section 19.10.

Contact-handled TRUW are defined as wastes that emit low levels of penetrating radiation. They can be handled in a hands-on manner with conventional equipment (e.g., forklifts) operated to minimize operator exposure to radiation. CH-TRUW are distinguished from RH-TRUW on the basis of the surface-dose rate. Waste canisters with dose rates no higher than 200 mR/hr are designated CH-TRUW and those with surface dose rates exceeding 200 mR/hr are designated RH-TRUW.

The commercial nuclear fuel cycle facilities that produce CH-TRUW include the fuel reprocessing plant and the mixed oxide fuel fabrication plant. These facilities package CH-TRUW and ship them to the geologic repository described in Section 17.0 of this study. If CH-TRUW must be shipped before the repository is available, they will be shipped to the transuranic waste interim storage facility described in Section 16.C.

19.9.1 Physical Description of the CH-TRUW Shipping System

The transuranic package transporter (TRUPACT) is the representative shipping container for hauling CH-TRUW. These shipping containers are being developed by the U.S. Department of Energy (DOE) to transport defense TRUW similar to those expected to be generated in the commercial fuel cycle. The TRUPACT is designed specifically to provide the containment required in the regulations for hauling Type B quantities of CH-TRUW. Also described in this section are the different waste packagings that are used as primary containers (i.e., canisters) for the waste form. These are containers for Type A quantities of radioactive materials that are placed inside the TRUPACT for shipment.

Transuranic wastes consist of a variety of materials that are contaminated with trace amounts of transuranic radionuclides. Some of these materials are metal scrap, paper, rags, sludges, filters, gloves, plastics, and rubber. A wide variety of Type A canisters have been used for storage and transport of CH-TRUW. The most commonly used is the DOT-17C or DOT-17H 210-L (55-gal) steel drum measuring 61 cm (2 ft) in diameter and 89 cm (35 in.) high. This drum has a full-open-head which is closed with a 2.6-mm (0.1-in.) locking ring with drop-forged lugs and a 16-mm (0.625-in.) diameter bolt and nut. The drum is lined with a 2.5-mm (0.1-in.) rigid polyethylene liner. Closure is by means of a snap-in lid sealed with adhesive or by welding. A 0.13-mm (0.005-in.) inner liner is used for contamination control during drum loading and inspection. Weights of full drums stored at the INEL^(a) range from 43.6 to 500 kg (100-1100 lb) (Shefelbine 1978 and ERDA 1976) with the average weight being about 145 kg (330 lb).

(a) INEL: Idaho National Engineering Laboratory - This is a storage location for the greatest amount of retrievably stored CH-TRUW.

Other wastes are placed in steel boxes of various sizes. A new waste canister called the modular box has been developed at Sandia National Laboratories. This box is constructed of 1.9-mm (0.075-in.) thick steel plates that are bent and folded to form a box. The modular box is of completely welded construction and provides a leak-proof, noncombustible packaging. The box size is 1.9 m x 1.3 m x 0.9 m (74 in. x 50 in. x 37.5 in.) with an empty weight of about 191 kg (420 lb) and a gross weight limit of 4.5 Mg (10,000 lb). The box dimensions allow its use as the primary waste container or as an overpack container for a 2 x 3 array of 210-L (55-gal) drums. Inner liners for contamination control during loading have not been investigated but liners similar to those being used in the drums and Rocky Flats boxes are also applicable to this waste packaging.

The TRUPACT is a Type B packaging concept currently being developed at the Transportation Technology Center (TTC) to transport different Type A containers carrying a variety of CH-TRUW. The TRUPACT is currently in the final design stages and a prototype is expected to be available in 1986. The TRUPACT conceptual design is currently a large metal box that may be available in both truck and rail transported versions. However, due to uncertainties in the dimensions of the conceptual rail version, the rail TRUPACT system used for this study utilizes two conceptual truck versions shipped on a railroad flatcar. The truck system consists of one truck TRUPACT shipped on a flatbed truck trailer.

As presently conceived, the truck TRUPACT will have inner and outer steel frameworks made of rectangular tubing. Steel sheets covering the inner and outer surfaces of the inner and outer frameworks are separated by about 0.3 m (12 in.) of high-temperature insulation and rigid polyurethane foam that provides thermal protection in the event of an external fire as well as a good shock absorbing medium. The foam is about 0.9 m (36 in.) thick on each end. The inner liner is constructed of stainless steel sheets; the outer shell is carbon steel. In the inner wall between the two frameworks is a stainless steel plate designed to prevent puncture of the inner liner. Access to the inner cavity is through two hinged closures. The inner closure has one inflatable elastomeric seal and one that is not inflatable to provide a seal after loading. The outer hinged door is also sealed with elastomeric gaskets when bolted shut and acts as an impact limiter and provides puncture and fire protection. The TRUPACT shipping container is shown in Figure 19.18. A TRUPACT aboard a truck trailer is shown in Figure 19.19.

The truck version of the TRUPACT is currently proposed to have outside dimensions of 2.4 m x 2.7 m x 7.5 m (8 ft x 9 ft x 24.5 ft), as shown in Figure 19.18. The current dimensions proposed for the inner cavity are 1.8 m x 2.1 m x 5.6 m (6 ft x 7 ft x 18.5 ft). It is designed to hold thirty-six 210-L (55-gal) drums or six 6-pack configurations (six 2 x 3 arrays of 55-gal drums) or six modular boxes. The maximum payload for a single TRUPACT is 13 Mg (28,000 lb). One of these containers is transported on a truck trailer and two are transported on a railcar.

PAYLOAD LIMIT: 13,000 KGS
EMPTY WEIGHT: 10,000 KGS

FIGURE 19.18. TRUPACT Shipping Container for CH-TRUW

FIGURE 19.19. TRUPACT on a Truck Trailer

The inner cavity of the TRUPACT is made secure by closing and bolting in place a hinged, 13-cm (5.1-in.) thick door that has a tubular steel framework and is filled with rigid polyurethane foam. The door is sealed by dual elastomeric seals, one inflatable and the other not inflatable. The exterior door is hinged, 90 cm (36 in.) thick, utilizes a single elastomeric seal, and is also bolted in place for transport.

The truck transport system is being designed to operate as a legal-weight system throughout the United States, i.e., at a gross vehicle weight (GVW) of 33.3 Mg (73,280 lb) or less. The positioning of the TRUPACT on the trailer is adjusted to uniformly distribute the load between the tractor and semi-trailer axles, as shown in Figure 19.19. The shipping container is attached to the trailer bed using standard container-mounting hardware that has been installed into the trailer frame to provide rigid attachments. The tie-down hardware is designed to regulations to provide a static strength equal to 2, 5 and 10 times the maximum weight of the loaded TRUPACT in the horizontal, vertical, and longitudinal directions, respectively (10CFR71.31(d)).

The railcar-transported TRUPACT option provides a system that can be used throughout the U.S. without restrictions. Two TRUPACT packages are shipped on a railcar with the closure ends of the packages facing each other. The railcars used are equipped with similar tie-down hardware as for the truck trailer. The overall length of the railcar is approximately 27.4 m (90 ft).

Forklift equipment cannot be used inside the TRUPACT because the floor is made from thin steel sheets supported on and attached to fairly wide-spaced tubular steel framework members and because the rigid foam between the inner and outer frameworks has a low compressive strength. Two potential modifications have been suggested to compensate for this difficulty; pallets

and roller conveyors. Space is available for the use of single or multiple pallets that operate on the floor of the TRUPACT. The pallet is placed into the open end of the TRUPACT and moved into position along the floor using wheels, rollers, or air bearings. For unloading, the pallets are moved out of the TRUPACT along the floor to allow forklift access to the waste canisters. Alternatively, sufficient room in the height of the TRUPACT has been allowed so that a pair of roller conveyors can be installed to expedite loading and unloading. The roller conveyors would be retracted to allow forklift access for waste removal. Dunnage, cribbing, and/or blocking of the load secures the waste canisters inside the TRUPACT inner cavity and prevents shifting of the load. Air bags and internal tie-downs are also being considered to prevent movement of the load inside the TRUPACT during transit.

The final TRUPACT design will have some provision for venting differential pressure between the sealed TRUPACT interior and the ambient atmospheric pressure for safe loading and unloading. This same vent may be used for radiological measurements of the internal cavity atmosphere, prior to opening, for the protection of facility and operating personnel. This vent will be protected from normal and accident conditions so that there can be no inadvertent venting of the internal atmosphere.

The TRUPACTS are still under development at this time and are expected to become available for routine use during 1986. Dimensions and some structural details could change before the design is completed.

19.9.2 Safety-Related Information

This section discusses the safety-related information important to identification of the risks associated with transportation of CH-TRUW in the TRUPACT.

Operating and Maintenance Requirements

Unloading of CH-TRU waste canisters from the TRUPACT consists primarily of inspection, leak rate testing, internal pressure venting and atmosphere sampling, detaching external tie-downs, removing the TRUPACT from the transport vehicle by crane, opening the outer and inner doors, removing dunnage and/or internal tie-downs, withdrawing the internal pallets or roller conveyors, and removing the waste packages. Loading is essentially the reverse of unloading operations. An alternative operational scheme for unloading truck systems is to leave the TRUPACT attached to the trailer bed and drive forklifts on the bed for access to the waste canisters.

TRUPACT maintenance requirements include repairs and decontamination. Repairs may be necessary for damaged door seals, external and internal plate structures, the foam, and external and internal tie-downs. Damaged external plates could be replaced by patch plate sections riveted to the existing plate, with appropriate provisions for sealing. However, repairing the interior stainless steel lining is a more difficult problem since a seal must be maintained for containment purposes. As a result, a patch plate for the inner

lining may have to be welded. This welding, if required, could necessitate removal of external skin and foam in the repair region to prevent contamination of the welds and degradation of the foam. Thus, welding would seem to be a relatively major repair procedure.

The inner shell of the TRUPACT is constructed of stainless steel which is easily decontaminated. Contamination of the TRUPACT is dependent, in part, upon the containment integrity of the waste canisters and how well they are secured within. The other internal gear (dunnage and/or tie-downs and removable pallets or roller conveyors) will be designed to facilitate decontamination.

Inventories of Hazardous Materials

The TRUPACT shipping container used in this study has the capability of transporting thirty-six 210-l (55-gal) drums or six modular steel boxes. Radionuclide data for CH-TRUW stored in drums and boxes are shown in Table 19.14. TRUW nuclide contents given are overall averages for wastes as discussed in Section 16.C of this report. Table 19.15 presents the radionuclide data for a typical TRUPACT cargo. It should be noted that these wastes are most likely contaminated by other radionuclides than those listed in the Tables, e.g., ^{233}U , ^{235}U , ^{236}U , ^{238}U , and some mixed fission products. However, their average and maximum quantities per drum and box are not available.

Containment and Shielding Systems

Double containment is required by NRC regulations for TRUW shipping packages containing more than 20 Ci of plutonium. This rule may be implemented by either providing a separate inner containment as part of the shipping package design or providing disposable overpacks for waste canisters. The TRUPACT shipping package is being designed with a single level of containment. It is proposed on the conceptual design that the inner skin is a containment boundary and that the TRUW forms will be exempted from this double containment requirement. If this exemption is not granted, sealing of individual waste packages (e.g., drums and boxes) to meet Type B requirements, or to meet

TABLE 19.14. Average Radionuclide Content per Drum and Box of CH-TRUW

<u>Radionuclide</u>	<u>Average Ci/drum</u>	<u>Average Ci/box</u>
Fission products	Negligible	Negligible
Actinides		
^{239}Pu	5.2E-1	1.3E-4
^{241}Pu	2.3E2	5.8E-2
Other Pu	8.6	2.2E-3
All other actinides	<u>7.7E-1</u>	<u>9.8E-5</u>
Total	2.4E2	6.0E-2

TABLE 19.15. Average Radionuclide Contents in a Fully Loaded TRUPACT Shipping Container

Radionuclide	Average Radioactivity Content ^(a)	
	Cargo Type	
	Drums (Ci)	Boxes (Ci)
²³⁹ Pu	1.9E1	7.8E-4
²⁴¹ Pu	8.3E3	3.5E-1
Other Pu	3.1E2	1.3E-2
Other actinides	<u>2.8E1</u>	<u>5.9E-4</u>
Total	8.7E3	3.6E-1

(a) Calculated by multiplying average activity in drums or boxes shown in Table 19.14 by 35 drums/TRUPACT or 6 boxes/TRUPACT.

secondary containment requirements when considered as a component of the TRUPACT structure, may be a more practical approach than adding a second or inner containment to the TRUPACT structure.

No shielding, neutron absorbers, or moderators are required for the types of materials shipped in the TRUPACT. Regulations allow dose rates up to 200 mR/hr on the exterior surface of the TRUPACT and 10 mR/hr at a distance of 1.9 m (6 ft).

19.10 TRANSPORTATION OF REMOTE-HANDLED TRANSURANIC WASTE

Some of the TRUW produced in the nuclear fuel cycle are contaminated with sufficient quantities of transuranic nuclides and fission products to produce external dose rates that require wastes to be handled remotely. Wastes that have surface dose rates higher than 200 mR/hr are designated as RH-TRUW. These wastes require remote handling operations from behind shielded walls to minimize the radiation exposure to workers. RH-TRUW are produced at fuel reprocessing plants and will eventually be shipped to a geologic repository for disposal. If the repository is not available, these wastes will be shipped to an interim storage facility until the repository is operational.

Two Type A waste canisters are used in this study to contain RH-TRUW. The first is a 60-cm (24-in.) diameter and 3.1 m (10-ft) long cylindrical canister that contains short lengths of fuel cladding, fuel assembly hardware, and contaminated failed equipment. The second canister is the 210-l (55-gal) steel drum that is used for solidified wet wastes, contaminated, failed equipment and assorted solid wastes.

Casks designed for shipment of fuel residues (cladding and hardware) have not yet been built. A conceptual shipping system described in document DOE/ET-0028 (DOE 1979) for transporting the larger fuel residue canisters is used as the basis for the fuel residue canister shipping system. The shipping system for transporting drums is one that is in use today. Both transport systems are designed as Type B packaging systems.

19.10.1 Physical Description of the RH-TRUW Shipping System

Truck and rail versions of two distinctly different conceptual RH-TRUW shipping systems are described in this section. The first is a conceptual packaging designed to transport fuel cladding hulls, fuel assembly hardware, and contaminated equipment that would be contained in 61 cm (24-in.) diameter by 3.1-m (10-ft) long canisters. The second conceptual system is designed to transport three 210-l (55-gal) drums containing failed equipment and other solid TRUW and solidified wet wastes (TRU-contaminated ion exchange resins, filter cartridges, etc.).

Transportation of Fuel Residues

This section describes the conceptual transportation system for fuel residues generated at a fuel reprocessing plant (FRP). This system is also used to transport TRU-contaminated failed equipment too large to fit in 210-l (55-gal) drums. The conceptual waste canister for these wastes is assumed to have a diameter of approximately 61 cm (24 in.) and a length of 3.1 m (10 ft). It is made of stainless steel with a wall thickness of about 0.6 cm (0.25 in.), end thickness of 1.3 cm (0.5 in.), and welded closures.

The truck and rail casks for transporting the fuel residue canisters are expected to resemble those currently used for transporting spent fuel. These casks would conform to the requirements for Type B packaging and would provide adequate gamma shielding to meet the radiation dose requirements of 49CFR173. They are expected to be somewhat more simple in design because heat removal requirements would be reduced and neutron shielding would not be required. Decay heat loads for RH-TRUW are low enough that cooling fins will most likely not be required. Impact protection for the casks would be provided by steel-clad balsa wood impact limiters similar to those described for spent fuel casks.

The conceptual rail cask (postulated in DOE 1979) is assumed to transport fuel residue canisters and weighs about 65 Mg (71.5 ton). The cask is a lead-filled, double-walled stainless steel cylinder. An insert would serve to position three RH-TRUW canisters inside the cask cavity and would act as a heat conduction path from the waste canister to the inner surface of the cavity wall. The cask would not be pressurized and the inner cavity would be shipped dry (no water coolant). Figure 19.20 shows a cross-section of the conceptual rail cask for transporting fuel residue canisters. Table 19.16 provides the basic design parameters for this cask. It should be noted that the conceptual

FIGURE 19.20. Cross-Section of the Conceptual Rail Cask for Transporting RH-TRU Fuel Residue Canisters

cask described in DOE/ET-0028 (DOE 1979) transports three fuel residue canisters that are 76 cm (30 in.) in diameter. Cask dimensions shown in Table 19.16 are scaled down to accommodate the smaller canisters used in this study. Cask weights and shielding thicknesses are reported as they are in DOE (1979) and are expected to be conservative values due to the lower heat generation rates and radioactivity contents of the smaller canisters used for this study.

The conceptual truck cask used for this study is based on the rail cask described in DOE (1979). No descriptive information was given in this document concerning the truck casks. As a result, the basic cask design parameters are obtained by scaling the rail cask down to a single-canister-capacity truck cask. The following assumptions were made to facilitate calculations of these design parameters. First, the gamma shielding thickness for the conceptual fuel residue cask is the same as that for the spent fuel truck cask. Second, the cavity shell and cask outer shell thicknesses are assumed to remain the same. The third and final assumption is that the decay heat can be adequately removed by conduction through the cask insert, cavity walls, lead gamma shielding, and outer shell and by radiation and convection to the atmosphere. A cross-section of the truck cask is shown in Figure 19.21. The basic design parameters for the cask are given in Table 19.17.

Transportation of RH-TRU Waste Drums

This section describes the representative system for hauling RH-TRUW contained in drums. These shipping containers are constructed to Type B

TABLE 19.16. Details of the Conceptual Rail Cask^(a)
for Transport of RH-TRU Fuel Residue Canisters

Length of structural steel	3.9 m (152 in.)
Diameter of structural steel	1.8 m (70 in.) ^(b)
Length of cask cavity	3.4 m (132 in.)
Diameter of cask cavity	1.4 m (55 in.) ^(b)
Number of canisters transported	3
Cask weight	
Net weight	65 Mg (71.5 ton)
Loaded weight ^(c)	72 Mg (80 ton)
Cavity shell	
Material	Stainless steel
Thickness	2 cm (0.75 in.)
Gamma shield	
Material	Lead
Thickness	15 cm (6 in.)
Outer shell	
Material	Stainless steel
Thickness	5 cm (2 in.)
Impact absorbers	
Internal	Honeycomb
External	Steel-clad balsa wood
Thermal power per cask ^(d)	600 W

(a) This table is based on information given in DOE/ET-0028 (DOE 1979, Table 6.4.2) and the assumptions outlined in the text.

(b) This value is based on the space required to transport three 60-cm (24-in.) diameter canisters. This information is different than what is shown in DOE 1979, Table 6.4.2.

(c) Assumes canisters weigh 625 kg (1380 lb.) empty and they are filled with 850 kg of uncompacted hulls and hardware and 850 kg of sand to fill void spaces.

(d) This value is the expected heat generation rate of three canisters of fuel residues.

FIGURE 19.21. Cross-Section of the Conceptual Truck Cask for Transporting RH-TRU Fuel Residue Canisters

package standards and are designed with shielding. Reusable transport shields are typically smaller than unshielded shipping containers such as the TRUPACT. The container volume in reusable transport shields is limited so that the gross vehicle weight of the transport shield, waste canisters, truck trailer and tractor do not exceed legal weight limits.

The representative reusable Type B transport shield for the purposes of this study is the HN-200 shipping container. This container is a top-loading system that has a capacity of three 210-L (55-gal) drums. The inner configuration consists of one tier of three drums positioned on a steel pallet. The maximum decay heat load is 34 W (116 Btu/hr). One of these containers can be shipped on a flatbed truck trailer and three can be shipped per rail car. The empty weight of the HN-200 is approximately 16.6 Mg (18.3 ton).

The HN-200 shipping container consists of a steel-lead-steel annulus cask fabricated in the form of a right circular cylinder. The shipping container is 168 cm (66.3 in.) in diameter and 190 cm (74.5 in.) in height. The cask is closed at one end and is provided with a lid closure at the other end. The cask wall consists of a 1 cm (0.38 in.) outer shell and a steel flange connecting the two shells. The cask outer shell is surrounded by a 2.5-cm (1-in.) thick layer of insulation contained within a 3.2-mm (14-gauge) steel shell.

The lid, sealed by O-ring gaskets, is of similar construction and is bolted to the cask body. A cylindrical shield plug is located in the center of

TABLE 19.17. Details of the Conceptual Truck Cask^(a) for Transport of RH-TRU Fuel Residue Canisters

Length of structural steel	3.9 m (152 in.)
Diameter of structural steel	0.9 m (35 in.) ^(b)
Length of cask cavity	3.4 m (132 in.)
Diameter of cask cavity	66 cm (26 in.) ^(b)
Number of canisters transported	1
Cask weight	
Net weight	unknown
Loaded weight ^(c)	33.3 Mg (73,280 lb) max.
Cavity shell	
Material	Stainless steel
Thickness	2 cm (0.75 in.)
Gamma shield	
Material	Lead
Thickness	16 cm (6.63 in.)
Outer shell	
Material	Stainless steel
Thickness	5 cm (2 in.)
Impact absorbers	
Internal	Honeycomb
External	Steel-clad balsa wood
Thermal power per cask	200 W

(a) Based on conceptual fuel residue rail cask described in DOE/ET-0028 (DOE 1979) and assumptions outlined in the text.

(b) Assumes inner cavity is designed with a 5-cm (2-in.) allowance for the cavity insert. Structural shell diameter is sum of the shell, shielding, and cavity thicknesses.

(c) The shipping system (cask, tie-downs, peripheral equipment, semi-trailer and tractor) is assumed to be legal-weight and is restricted to a maximum gross vehicle weight (GVW) of 33.3 mg (73,280 lb).

the cask lid and is sealed by O-rings. Lifting and tie-down devices are attached to the cask body and impact skirts, consisting of removable rings of shock-absorbing foam, are attached to the ends of the cask.

19.10.2 Safety-Related Information

This section contains important safety-related information concerning the two different RH-TRUW shipping systems. This information is important to identification of the risks associated with transporting RH-TRUW.

Operating and Maintenance Requirements

Operating and maintenance requirements for the fuel residue cask are similar to those for the spent fuel and solidified high-level waste casks. The containment vessel is dry (no free water) when delivered for transport. Seals must be tested prior to leaving a facility with cargo and will most likely be replaced every 12 months. Both of these requirements also pertain to the HN-200 shielding transport package for RH-TRUW drums.

Both of the RH-TRUW shipping containers have steel external surfaces for ease of decontamination. Decontamination operations are assumed to be similar to those described for spent fuel shipping casks.

Inventories of Hazardous Materials

This section contains radionuclide content data for the conceptual fuel residue cask and the HN-200 reusable transport shield for RH-TRUW drums. Radionuclide data for the truck and rail transport configurations of these shipping containers is shown in Table 19.18. The information in this table is based on the radionuclide contents of typical fuel residue canisters and RH-TRUW drums described in Section 16C.

Containment and Shielding Systems

Both types of transportation systems described in this section are designed with shielding systems to reduce the dose rate at their surfaces below 200 mR/hr. The fuel residue rail cask incorporates 15 cm (6 in.) of lead for gamma shielding and 7 cm (2.75 in.) of stainless steel. The truck cask is assumed to incorporate 16 cm (6.6 in.) of lead and 7 cm (2.75 in.) of stainless steel. The shielded overpack for RH-TRUW drums contains 9.5 cm (3.75 in.) of lead shielding and 3.5 cm (1.4 in.) of stainless steel.

Seals are provided for the lid closures of both the fuel residue cask and RH-TRUW shipping Container. O-ring seal gaskets are provided in both cases.

19.11 TRANSPORTATION OF LOW AND INTERMEDIATE LEVEL NON-TRANSURANIC WASTE

Low- and intermediate- level non-transuranic wastes (non-TRUW) are generated at all nuclear facilities. They include a wide variety of materials with a wide range of radioactivity contents. These wastes include such items

TABLE 19.18. Radionuclide Data for the Truck (a) and Rail Fuel Residue Casks and RH-TRUW Drum Shipping Container

Radionuclide	Radionuclide Inventory (Curies)			
	Fuel Residue Cask		Shielded Drum Container	
	Truck	Rail	Truck	Rail
Tritium	1.7E+2	5.1E+2	2.5E-1	7.6E-1
Semi volatile fission products	8.9E+2	2.7E+3	2.4E+2	7.1E+2
All other fission products	1.4E+3	4.2E+3	2.3E+1	6.9E+1
Activation products	2.4E+4	7.2E+4	1.4	4.1
Uranium plus lower actinides	negligible		negligible	
Transuranic nuclides	2.7E+2	8.1E+3	5.1E+2	1.5E+3
Total	2.6E+4	7.8E+2	7.7E+2	2.3E+3

(a) Based on radionuclide data contained in Section 16.C of this study.

as resins and filter cartridges from liquid cleanup systems, HEPA filters, contaminated paper, gloves, protective clothing, wood, and failed equipment. A large fraction of the wastes in the fuel cycle, on a volume basis, can be placed in these two categories of materials. Most shipments of low- and intermediate- level non-TRUW in the past have been made by truck although rail transportation has also been used. Therefore, truck shipments are assumed to be the primary mode of transport for these materials.

19.11.1 Physical Description of Low- and Intermediate- Level Waste Shipping Systems

Prior to shipment, low- and intermediate- level non-TRUW will be packaged in disposable containers. Wastes are treated to immobilize and increase the durability of the waste and compacted to reduce the volume of the wastes. They are packaged in 210- ℓ (55-gal) steel drums, large (3.6-m³) or small (0.45-m³) plywood boxes, or large (4.8-m³) or small (1.4-m³) liners that are then placed inside shipping containers for transport to waste disposal facilities (NRC 1981b). The majority of these wastes are expected to be shipped as low specific activity (LSA) material as defined in 49CFR173.393 or as Type A shipments.

Special placarding is required for these shipments, as it is for all radioactive material shipments, to distinguish these vehicles from other, non-hazardous shipments and to alert persons at an accident scene of the hazardous cargo.

19.11.2 Safety-related Information

Low- and intermediate- level non-TRUW present primarily a radiological health hazard. Loading and unloading operations require cranes and forklifts to handle shipping container lids, internal dunnage, and waste canisters. Very little personnel contact with the potentially contaminated shipping containers and waste canisters will occur.

A wide variety of wastes are expected to be shipped as low- or intermediate- level non-TRUW. Reactor process wastes and general trash are chosen to be representative of this type of waste generated in the nuclear fuel cycle. The isotopic content of process and special wastes and trash from BWRs and PWRs was shown in Section 18.0. Using this data, average radionuclide contents of a rail and truck shipment of both types of reactor wastes were calculated. The rail shipping container was assumed to have a capacity of 120 drums of waste and the truck shipping container was assumed to transport 50 waste drums. The results of these calculations are shown in Table 19.19.

Low- and intermediate- level non-TRUW do not have a double containment regulation. Shielding may be required for certain cargos, e.g., solidified ion exchange resins and spent HEPA filters that have become excessively contaminated. However, relatively modest shielding would be required for these wastes.

A shipping distance for these wastes is conservatively assumed to be 4000 km (2500 miles). This is the approximate distance wastes from a reactor on the east coast of the United States would travel if they were being transported to the commercial low-level waste burial grounds in Richland, Washington, or Beatty, Nevada.

TABLE 19.19. Isotopic Content of Truck and Rail Shipments of General Reactor Trash and Reactor Process Wastes^(a)

Type of Waste and Shipment Mode ^(b)	AP	H-3	SVFP	AOFP	ULA	TRU	Total
PWR Process Wastes							
Truck	2.8E+1 ^(c)	9.4E-2	7.9E-1	7.2E-3	5.2E-5	2.6E-1	2.9E+1
Rail	6.7E+1	2.2E-1	1.9	1.7E-2	1.2E-4	6.2E-1	7.0E+1
BWR Process Wastes							
Truck	6.3E+1	3.1E-1	3.3E+1	7.3E-2	3.5E-5	2.0E-1	9.6E+1
Rail	1.5E+2	7.4E-1	7.8E+1	1.6E-1	8.5E-5	4.7E-1	2.3E+2
PWR General Trash							
Truck	4.5	6.9E-2	5.7E-1	5.1E-3	1.6E-5	5.9E-2	5.2
Rail	1.1E+1	1.6E-1	1.4	1.2E-2	3.8E-5	1.4E-1	1.3E+1
BWR General Trash							
Truck	2.5E+1	1.0E-1	1.1E+1	2.1E-2	4.2E-6	9.6E-2	3.6E+1
Rail	6.0E+1	2.5E-1	2.6E+1	5.2E-2	1.0E-5	2.3E-1	8.6E+1

(a) Abbreviations are as follows: AP: Activation Products, H-3: Tritium, SVFP: Semi-Volatile Fission Products (Ru, Rb, Te, Cs, Sb), AOFP: All Other Fission Products, ULA: Uranium plus Lower Actinides, TRU: TRAnsUranic isotopes

(b) Assumes truck shipment contains 50 drums and rail shipments contain 120 drums of wastes

(c) $2.8E+1 = 2.8 \times 10^1$

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GLOSSARY

The symbols for metric units and the prefixes used with the metric units are defined below. A definition of acronyms and abbreviations used in this study are also listed. Common terms covered in standard dictionaries and commonly used chemical symbols are not included.

Prefixes Used With Metric Units

<u>Prefix</u>	<u>Abbreviation</u>	<u>Factor</u>
giga	G	10^9
mega	M	10^6
kilo	k	10^3
centi	c	10^{-2}
milli	m	10^{-3}
micro	μ	10^{-6}
nano	n	10^{-9}
pico	p	10^{-12}

Symbols for Metric Units

<u>Symbol</u>	<u>Name</u>
$^{\circ}\text{C}$	degree celsius
Ci	curie
d	day
g	gram
h, hr	hour
J	joule = $1 \text{ N}\cdot\text{m}$
L	liter
m	meter
<u>M</u>	gram-mole/liter
min	minute
N	Newton = $1 \text{ kg}\cdot\text{m}/\text{s}^2$
Pa	Pascal = $1 \text{ N}/\text{m}^2$
R	Roentgen
rem	roentgen equivalent man
s	second
W	Watt = $1 \text{ J}/\text{s}$
yr	year

Definition of Acronyms and Abbreviations

AP:	Activation product
ADU:	Ammonium diuranate
AEC:	Atomic Energy Commission
AFR:	Away from reactor (spent fuel storage)
ALARA:	As low as reasonably achievable
AOFP:	All other fission products
ASHRAE:	American Society of Heating, Refrigeration, and Air Conditioning Engineers
ASME:	American Society of Mechanical Engineers
BCL:	Battelle-Columbus Laboratories
BWR:	Boiling water reactor
CCD:	Countercurrent decantation
CFR:	Code of Federal Regulations
CH-TRUW:	Contact-handled transuranic waste
DF:	Decontamination factor
DOT:	Department of Transportation
DOE:	Department of Energy
EPA:	Environmental Protection Agency
ERDA:	Energy Research and Development Administration
FP:	Fission product
FFP:	Fuel fabrication plant
FRP:	Fuel reprocessing plant
GESMO:	Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors, NUREG-0002
HC:	Hydrocarbons
HEPA:	High-efficiency particulate air
HLW:	High-level waste
HLLW:	High-level liquid waste
HTGR:	High temperature gas-cooled reactor
HVAC:	Heating, ventilation, and air conditioning
IAEA:	International Atomic Energy Agency
ILW:	Intermediate level waste
INEL:	Idaho National Engineering Laboratory
K_{eff} :	Effective neutron multiplication factor
LA:	Lower actinides
LANL:	Los Alamos National Laboratory
LLW:	Low-level waste
LLNL:	Lawrence Livermore National Laboratory
LMFBR:	Liquid metal fast breeder reactor
LWR:	Light water reactor
MOX:	Mixed oxide (U and Pu)
MOX-FFP:	Mixed oxide fuel fabrication plant
NG:	Noble gases
NA:	Not available
NDT:	Non-destructive testing
NEG:	Negligible

NRC: Nuclear Regulatory Commission
NWTS: National Waste Terminal Storage Program
ORNL: Oak Ridge National Laboratory
PNL: Pacific Northwest Laboratory
PWR: Pressurized water reactor
RHO: Rockwell Hanford Operations
RH-TRUW: Remote-handled transuranic waste
SAI: Science Applications, Inc.
SHLW: Solidified high-level waste
SNM: Special nuclear material
SNL: Sandia National Laboratories
SURF: Spent unprocessed fuel
SVFP: Semi-volatile fission products
SWU: Separative work units
TBP: Tri-butyl phosphate
TRU: Transuranic nuclide
TRUW: Transuranic waste
UBC: Uniform Building Code
ULA: Uranium plus low actinides
UNH: Uranyl nitrate hexahydrate
WSEP: Waste solidification engineering prototypes

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