Advanced Fuel Cycle Cost Basis Report:

E MODULES

Storage of Spent Nuclear Fuel and Recycled Products

Nuclear Fuel Cycle and Supply Chain

Prepared for
U.S. Department of Energy
Systems Analysis and Integration
March 2021
INL/EXT-21-61994
Revision 1

CONTENTS

FIGURES

TABLES

ACRONYMS

APSF	Actinide Packaging and Storage Facility
BNFL	British Nuclear Fuels plc
CFTC	Consolidated Fuel Treatment Center
CPP	Central Processing Plant
CSF	consolidated storage facility
CWCCIS	Civil Works Construction Cost Index System
DOE	Department of Energy
DOE-STD	DOE Standard
DR	digital radiography
FCRD	Fuel Cycle Research and Development R&D
INL	Idaho National Laboratory
ISFSIs	interim spent fuel storage installations
LCC	lifecycle cost
LWRs	light-water reactors
MA	minor actinides
MC&A	material control and accountancy
MOX	mixed-oxide
MT	metric tons
NDA	non-destructive assay
NEA	Nuclear Energy Agency
O&M	Operations and Maintenance
OCW	outer can welder
OECD	Organization for Economic Cooperation and Development
P&RS	Product & Residues Store
Pu	plutonium
ROM	rough order of magnitude
RRP	Rokkasho recycling plant
SNF	spent nuclear fuel
SRS	Savannah River Site
THORP	Thermal Oxide Reprocessing Plant
TPC	total project cost
TRL	technology readiness level
TRU	transuranic

UFD	Used Fuel Disposition
UO_3	enriched reprocessed uranium
VOA	Voluntary Offer Agreement
WBS	work breakdown structure
WIT	What-It-Takes

ACKNOWLEDGEMENT

This latest version of *the Module E Modules Storage of Spent Nuclear Fuel and Recycled Products* is the result of the cumulative effort of many authors that have contributed to the Advanced Fuel Cycle Cost Basis Report (AFC-CBR). It is not possible to identify and acknowledge all those contributions to the AFC-CBR and this module. All the authors, including the four primary authors, fifteen contributing authors, the twelve contributors acknowledged, and the many other unacknowledged contributors in the 2017 version of the report may have contributed various amounts to the development and writing of this module prior to this current revision. Unfortunately, there is not a consolidated history that allows us to properly acknowledge those that built the foundation that was updated and revised in this latest revision.

This update reformats previous work to the current format for rerelease of the entire report as individual modules so there is no primary technical developer or lead author. J. Hansen (INL) and E. Hoffman (ANL) can be contacted with any questions regarding this document.

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Suggested Citation:

DOE-NE Systems Analysis & Integration Campaign, *Advanced Fuel Cycle Cost Basis Report: E MODULES Storage of Spent Nuclear Fuel and Recycled Products*. Idaho Falls: Idaho National Laboratory, 2021.

Module E1: Wet Storage of Spent Nuclear Fuel

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E1 REVISION LOG

Rev.	Date	Affected Pages	Revision Description
	2004	All	Version of AFC-CBR in which Module first appeared: 2004 as Module E1. In 2012 it was decided to discontinue this module since wet (pool) storage of SNF is generally part of reactor Operations and Maintenance costs. Also, no new centralized or on-site SNF storage pools are planned, so no present need for this Module exists.
	2009	All	Latest version of module in which new technical data was used to establish unit cost ranges: 2009 (It should be noted that the unit costs in the 2009 version may be indicative of a stand-alone wet storage pool.)
			New technical/cost data which has recently become available and will benefit next revision: New cost data on dry interim onsite storage should now be available from the USDOE-NE Used Fuel Disposition Campaign. In future versions of the AFC-CBR this data would appear in Module I (formerly Module E2)
	2021	All	See Module I, Interim Storage, for wet storage of spent nuclear fuel
	2021	All	Re-formatted module consistent with revised approach to release of the AFC-CBR and escalated cost estimates from year of technical basis to escalated year 2020. Cost estimates are in US dollars ($) of year 2020.

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Module E2: Dry Storage of Spent Nuclear Fuel

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E2 REVISION LOG

Rev.	Date	Affected Pages	Revision Description
	2004	All	Version of AFC-CBR in which Module first appeared: 2004 as Module E2. In the 2012 AFC-CBR this Module was renamed Module I (Consolidated Interim Storage). This concept is one of those discussed in the recently-released "Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste" (U.S. Department of Energy, January 2013). The data in Module I also benefit from the *USDOE-NE Used Fuel disposition Storage, Transportation, and Disposal Interface Cost Study* and the *Used Fuel disposition System Architecture Report*.
	2012	All	Latest version of module in which new technical data was used to establish unit cost ranges: 2012
			New technical/cost data which has recently become available and will benefit next revision: USDOE-NE Used Fuel Disposition (UFD) studies on the cost and schedules for both on-reactor-site interim spent fuel storage installations (ISFSIs) and consolidated centralized storage should now be available.
	2021	All	See Module I, Interim Storage, for dry storage of spent nuclear fuel
	2021	All	Re-formatted module consistent with revised approach to release of the AFC-CBR and escalated cost estimates from year of technical basis to escalated year 2020. Cost estimates are in US dollars ($) of year 2020.

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Module E3

Storage of Combined Recycled Product of Mixed Plutonium, Minor Actinides, and Uranium Product

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E3 REVISION LOG

Rev.	Date	Affected Pages	Revision Description
	2004	All	Version of AFC-CBR in which Module first appeared: 2004 as Module E3.
	2009	All	Latest version of module in which new technical data was used to establish unit cost ranges: 2009 for stand-alone storage facilities, 2015 for co-located storage facilities
			New technical/cost data which has recently become available and will benefit next revision: None identified.
	2021	All	Re-formatted module consistent with revised approach to release of the AFC-CBR and escalated cost estimates from year of technical basis to escalated year 2020. Cost estimates are in US dollars ($) of year 2020.

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Module E3
Storage of Mixed Recycled Plutonium, Minor Actinides, and Uranium Product

E3-MD. SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year 2020 for this FY21 update.**
- **Nature of this FY21 Module update from previous AFC-CBRs:** Escalation only.
- Estimating Methodology for latest AFC-CBR technical update from which this FY21 update was escalated:
  - 2009 for stand-alone storage facilities. Most cost bases are pre-conceptual cost estimates for proposed storage facilities. Some data on completed facilities was available.
  - 2015 for storage facilities co-located with reprocessing plants. Bases are pre-conceptual cost estimates for proposed storage facilities.

E3-1. BASIC INFORMATION

Recycled product storage facilities would safely store the pure or mixed transuranic actinide products or by-products produced from the reprocessing of thermal reactor and/or fast reactor fuels. The particular aim of this module is to provide best estimates of the cost of storage for mixed plutonium and other transuranic minor actinides as proposed for advanced fuel cycles with higher actinide incineration. These costs estimates are extrapolated from existing facility experience and design studies for storage plants for plutonium oxide and metal. The cost estimates are provided in two forms: a fixed quantity based on a nominal storage period suited to an established advanced fuel cycle and a time-dependent quantity for use where the advanced fuel cycle system is far from steady state.

Several operating facilities and/or design studies were found that represent plutonium storage: the Mayak fissile material storage facility in Russia, the Actinide Packaging and Storage Facility (APSF) at the Savannah River Site (SRS), the Unirradiated Fuel Storage Facility – Central Processing Plant (CPP) 651 at Idaho National Laboratory (INL)[a], the Sellafield Product and Residues Store (P&RS) in the U.K., and the Pu product store at AREVA Cap La Hague in France. The Savannah River Site provided information for a U.S. consolidated storage facility (an unpublished study) and information from the Engineering Alternative and Follow-on Engineering Alternative Studies conducted for the Consolidated Fuel Treatment Center (CFTC). Generalized cost information was also obtained from a nuclear fuel cycle economics study reported by the Organization for Economic Cooperation and Development (OECD)-Nuclear Energy Agency (NEA).

There are two main storage approaches that must be accommodated. For the eventual well-developed and deployed nuclear fuel cycle with full plutonium recycle and possibly also the partial/full recycle of minor actinides, short (several year, storage of plutonium, etc.) is required to act as buffer storage between facilities and to accommodate the logistics of the overall fuel cycle, which may be in a transient phase of growth. Given the delay in deployment of fast reactors compared to earlier plans, the current general approach being adopted for excess fissile materials by both commercial and defense sectors is that

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[a] Beginning February 1, 2005, the name of the Idaho National Engineering and Environmental Laboratory (INEEL) was changed to Idaho National Laboratory (INL).

of stabilization (if needed), packaging and placement into stores with the capability of 50-year secure storage life, possibly extendable to 100 years. Nuclear materials can be withdrawn at any time consistent with the facility material handling rate, but the time period and costs are sufficient to deter unnecessary movements.

Module E3 examines cost estimates for material handling and short-term storage as well as material stabilization, handling, and long-term storage. Approximate costs are determined for the case of plutonium: both commercial oxide storage and defense oxide and metal storage. The costs of storage of other transuranic products and blends, which may arise from advanced fuel cycles, are estimated by extrapolation from plutonium storage costs by consideration of decay heat levels, criticality safety, and dose levels.

Recycled actinide products, actual and proposed, for potential storage include the following cases:

1. Recycled uranium oxide
2. Plutonium oxide
3. Blended uranium and plutonium oxides
4. **Mixed plutonium, transuranic minor actinides and uranium (1)** – Near pure uranium and blended uranium, plutonium, and/or minor actinide (Np, Am, and Cm) metals or oxides from various electrochemical processing ("pyro") flowsheets
5. **Mixed plutonium, transuranic minor actinides and uranium (2)** – Selected pure or mixed streams of U, Pu, Np, Am, and Cm oxides as delivered as by-products from various proposed UREX+ flowsheets
6. Mixed americium and curium oxide from aqueous reprocessing.

From recycling of thermal oxide fuel, the first two cases, UO_3 and PuO_2 storage, have been routinely performed by commercial industry for decades although the technical demands of plutonium oxide storage have increased with increasing fuel burnup, especially in light-water reactors (LWRs). The third case has recently started at the Rokkasho recycling plant (RRP) in Japan where deliberate mixing of U and Pu products has been practiced to improve safeguards. In the last decade, increased attention has been paid to long-term storage of metallic and oxide plutonium within the defense sector.

Regarding Case 1, the recycled uranium from LWR fuels is often low enriched, ~1% U-235, and has higher U-232 concentration than natural uranium. In LWR recycled uranium, the β-γ dose increases by several orders of magnitude after several years' storage. This is due to daughters of Th-228 created by α decay of U-232. Commercial recycling plants use mainly automated remote equipment for uranium finishing (formation of dry solid oxide product), sampling, metering to drums, closure, decontamination, transport, and storage. This equipment is normally maintained, following inventory removal, using contact maintenance. This modern approach is consistent with conventional industrial automation, and dose and toxic inhalation/ingestion minimization. The heat generation rate of a filled drum is low. Drum capacity varies between around 180 kg for ~1% enriched reprocessed uranium (UO_3) from LWR fuel to around 800 kg for ~0.2% enriched uranium from first generation, natural uranium fuelled, gas reactors. The overall storage cost of once reprocessed uranium is low and is included in Module K-2. The cost values in this module concern the higher costs of storage of transuranic material, which may also be mixed with uranium.

The first three cases, 1–3, were generally planned for buffer storage purposes, but the delayed introduction of fuel recycling has increased the need for planned long term storage (e.g., up to 50 years and with potential for extension to 100 years). Plant capital and operating costs for Cases 2 and 3 are known in principle, but may be commercially restricted information. The perceived excess of fissile materials for defense purposes and the need to treat and/or recover fissile materials from process wastes has also created the need to provide capabilities for stabilization of mixed fissile materials (e.g., plutonium bearing metals and oxides with ≥30 wt% plutonium plus uranium).

Plutonium separated from LWR fuels in commercial oxide reprocessing plants is separated by solvent extraction in pulse columns or centrifugal contactors in cells, further purified by additional polishing trains in gloveboxes, and finished to solids using remotely equipment in gloveboxes with contact maintenance. It is sampled, metered (~4–8 kg Pu) to stainless steel canisters, and the canister seals are remotely welded and contaminated ends cut off (by laser for Sellafield). Again, gloveboxes are used with contact maintenance. The higher burnups of LWR fuel give higher decay heats and penetrating doses from Pu products with isotopic spectrum of increased higher radionuclide atomic masses. These necessitate an all metal storage canister and an automated loading system for the Pu store. Criticality safety and package cooling require engineered features.

Mixed U and Pu product (MOX blend) (i.e., Category I) as used by Rokkasho, can be stored under similar conditions to Pu oxide storage. Costs per kg(Pu) may be greater due to the U diluent essentially increasing the number of canisters and storage positions needed. A conservative approach to design would be to retain the same size canisters for U-Pu oxide product as for PuO_2 product as a form of geometric limitation (take no safety credit for U dilution). The CFTC studies adopted an approach of limiting the Pu content to the same quantity (4.4 kg Pu) as allowed by the DOE 3013 package standard (Jones 2007) but allowing an increased canister length to accommodate the additional U diluent. This "elongated" 3013 canister was accommodated in an APSF style vault with very little increase in building height/size.

Cases 4–6, especially Case 5, are the particular focus of this module and cost estimates. The needs for buffer, medium or long storage of the latter three products, by-products and/or wastes, Cases 4–6, are not yet established and will depend on the requirements, processes, and properties of the projected advanced fuel cycles. These cases are being studied under the DOE, Office of Nuclear Energy, Fuel Cycle Research and Development (R&D) (FCRD) program for potential future industrial deployment and have much higher penetrating dose rates and decay heats than Pu. The whole of the UREX+ flowsheet operation is likely to be deployed within fully remote-maintained hot-cells. Finishing of the actinide product, sampling, metering, canister welding, checking, and swabbing will be performed remotely with full remote maintenance of equipment. The actinide products (e.g., separated streams or blends of U, Pu, Np, Am, and Cm oxides) generally have higher decay heat, penetrating radiation, and still with high-fissile content. A proposed U content is 67 wt% and the higher actinide content 33% by weight. Again there will be an increased number of canisters for storage relative to pure PuO_2 product. This may stem from dilution with UO_3 and either criticality safety or heat rating limitation. There is little cost data for storage of products and wastes for Case 4 and so no strong reasons to consider these separately from Case 5.

For Cases 4 and 5, the CFTC studies adopted an approach of limiting the TRU content to the same quantity (4.4 kg) of Pu allowed by the DOE 3013 standard (Jones 2007). This "elongated" 3013 canister was accommodated in an APSF style vault with very little increase in building height/size. The increased heat and radiation dose was determined to be acceptable for an air cooled vault when the TRU content did not exceed 35% (Jones 2007).

For Case 6, the specific heat (watts/gram) of the Am/Cm oxide is much greater than the U/TRU product due to the loss of the diluent such that the CFTC studies adopted an approach of using a high-heat source type package as allowed by DOE standard 5320. The EP-60 product container associated with this standard is much smaller (1.7-inch ID × 14.4-inch length) and the capacity is limited to 0.25 kg Am/Cm. The CFTC studies envisioned a separate vault for the cases in which high decay heat materials were stored.

Although actual cost data is not readily available for large-scale mixed transuranic storage, the design development, operational, and cost data from existing plutonium storage is considered adequate for extrapolation to the higher decay heat, criticality hazards, and radiation levels posed by the higher actinide mixtures of Cases 4–6.

E3-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

E3-2.1 General Transuranic Storage Requirements

Storage facilities containing fissile transuranic material shall provide safe, secure storage while satisfying national safeguards, and if commercial in use, may need to satisfy international (International Atomic Energy Agency [IAEA]) safeguards. The facilities are fabricated with massive secure structures, which maintain geometric integrity to avoid criticality accidents and minimize damage to canisters from external hazards such as aircraft impact, seismic events, extreme weather, and terrorist actions while containing radionuclides and supplying gamma and neutron shielding. They have multiple systems of containment, material control and accountancy (MC&A), and surveillance to provide an effective safeguards system.

In principle, there are three main storage options for actinides: storage in vented canisters, sealed canisters, or in an immobilization medium such as glass or ceramic. Air-cooled and water-cooled concepts have been proposed. For plutonium, either as metal or as oxide, the approach usually adopted is to stabilize the material, package within multiple nested canisters, and store within an air-cooled store. This may be a secure vault separated into bays by concrete walls where unshielded canisters are hung on walls in a horizontal array and an automated guided vehicle moves canisters to or from storage locations. Alternatively, it may be modules of air-cooled arrays of horizontal (or vertical) sleeves with shield plugs extending through the concrete wall (or floor) of the vault bay. A remotely operated stacker retriever moves canisters to or from storage locations. Facility design should support documented surveillance of all packages for integrity, safety, safeguards, and storage conditions. There may be a trade off between ease of inspection, potential for widespread contamination of canisters, and store. Natural convection may be possible for some vault geometries and has the advantage of passive cooling, but safety analysis may require the use of active ventilation systems with high air flow, treatment by filtration, etc. High reliability of ventilation systems is required and the ability of the canisters and store to survive ventilation failure is often part of the licensing process. Typically, a long-term store (design life of 50–100 years) is likely to accept ~10,000 storage canisters (approx. 40 t of fissile material) or more.

Technical issues concerning safe storage include: criticality safety of each canister and all arrays of canisters, limitation of heat generation (depends on isotopic composition, and mass, and may be limited to 19–30 with a canister depending on store design), drop testing of nested canisters, restriction of moisture and other volatiles content, loss on ignition testing (etc.), specific area limitation on metals (avoidance of thin section metals with pyrophoric hazard), removal of thick oxide coating from metals, pressure rise tolerance of canister, immobilization process (plutonium oxide stabilization in oxidizing atmosphere [e.g., 950°C for ≥2 hours] where significant non-actinide materials are present), use of inert filler gas and avoidance of water adsorption after stabilization, free volume of canister to minimize pressure rise, and potential explosive hazard on re-opening canisters with hydrogen enriched atmospheres (Rothman and Liu 1998).

Significant industrial factors include: operational efficiency, maintenance and repair, environmental impact, radiation and criticality safety, safeguards and security, heat removal, waste minimization, national and international inspection, and construction and operational costs.

Plant activities include: receipt and shipping, material handling, monitoring, system engineering (e.g., potentially radioactive ventilation), and physical security. Remote and/or automated handling of canisters is needed to minimize radiation dose to operators and provide geometric controls for criticality safety.

Heat generation (self-heating) from radioactive decay during storage and handling is a serious issue for design and operation. The decay energies for various grades (isotopic compositions) of plutonium and other transuranic isotopes differ markedly. The principal ones relevant to storage of uranium and plutonium products and storage of pure or mixed transuranic by-products are shown in Table E3- 1. Reprocessing of thermal oxide fuel (e.g., LWR type) is assumed to take place after 5–10 years or more of cooling so that short-lived isotopes are assumed to have decayed. These values are approximate since in some cases they incorporate increases due to decay of short-lived daughters and are time dependent (DOE 2004).

Table E3- 1. Specific heat generation rates for actinide products and by-products of reprocessing.

Nuclide or Mixture	Grade	Half Life/ Composition	Heat Generation Rate, W/kg
U-235	-	710×10^6 yr	60×10^{-6}
U-238	-	4.51×10^9 yr	8×10^{-6}
Pu	Weapon Grade	0.05%, 93.5%, 6.0%, 0.40%, 0.05% wt. for Pu isotopes 238 to 242.	2.8
Pu	Fuel Grade	0.1%, 86.1%, 12.0%, 1.6%, 0.2% wt. for Pu isotopes 238 to 242.	4.5
Pu	Power Grade (≥19% Pu-240)	1.0%, 63.0%, 22.0%, 12.0%, 3.0% wt. for Pu isotopes 238 to 242.	19
Pu-238	-	86 yr	570
Pu-239	-	24.4×10^3 yr	1.9
Pu-240	-	6.58×10^3 yr	7.1
Pu-241	-	13.2 yr	13
Am-241	-	458 yr	110
Cm-242	-	163 days	120×10^3
Cm-244	-	17.6 yr	2.8×10^3

For fuels, a high content of Pu-239, the high specific thermal output of Pu-238, together with in-growth of Am-241 from Pu-241 dominates overall self-heating. The thermal outputs of civil plutonium derived from 30 and 60 Gw.d/MT(U) burnup fuels are approximately 10 and 30 W/kg, respectively. The Pu-238 content increases with increasing fuel irradiation and exceeds 2% wt for burnup levels of 40 GW.d/MT(U) in LWRs. Canister wall temperatures may exceed 100°C and possibly reach twice this value at the center of the nuclear material. To provide defense-in-depth and aid handling, multiple nested metallic canisters, normally 300 series stainless steel, are used that must retain high conductivity for heat rejection, often by close fit. Canisters are frequently rated for an internal pressure rise of around 20 bar. Criticality safety limits and allowances are examined to store design and process operation (including mal-operation) to accommodate flooding, composition ranges, heterogeneity, batch doubling, analysis accuracy, etc. In some cases, a critical dimension method is used. The need for free volume in storage canisters is in competition with the use of the canister to provide geometric control.

For decay storage of curium, Case 6, the heat generation is dominated by Cm-244 and Cm-242 (depending on cooling period), which gives an overall decay power of about 2 kW/kg and is about two orders of magnitude higher on a mass basis than that for LWR plutonium. The quantity of plutonium from reprocessing of LWR uranium fuel is about two orders of magnitude greater than that of curium. Also, selected isotopes of the higher actinides have lower criticality safety mass limits than Pu-239. Commonly, engineered glasses are used for the storage and transport of separated curium and americium. Engineered glasses are designed to minimize the rate of dissolution in groundwater, provide dilution and reduced storage temperatures, and enable actinide recovery by glass dissolution in, for example, strong acid. (The CFTC studies assumed an Am/Cm oxide product). In broad terms, the number of packages to store curium from a given amount of LWR fuel may be quite similar to the number to package plutonium from the same fuel. This would make curium storage some 100 times more expensive per kg of actinide than plutonium storage. An alternative waste form and store design may be cost effective for long-term storage of curium.

E3-2.2 Examples of Plutonium and Higher Actinide Storage Facilities

These facilities are not chemical or manufacturing process plants. Defense stores may accommodate stabilized high Pu-bearing materials whereas Pu product oxide from commercial reprocessing has exceptional chemical purity. In the commercial nuclear fuel cycle, material stabilization and packaging in canisters is often performed in the reprocessing plant, which reduces the complexity of the store. Although improbable, any breached canisters are likely to be over-packed and returned to the reprocessing plant. In the defense sector, plants for storage of excess plutonium and components may include process stages for inspection, material analysis, monitoring, material stabilization and/or testing prior to canister filling, and closure. Material from any breached canisters is likely to be repackaged within the storage facility. The main technology considerations to note for storage aspects are the security, remote handling, maintenance, inspection, and material control and accountability aspects. Commercial stores are likely to provide international safeguards arrangements. Descriptions of the Mayak Storage Facility, Actinide Packaging and Storage Facility, CFTC storage facilities, Idaho Central Processing Plant-651, Consolidated Storage Facility, and Sellafield Product & Residues Store are now given.

E3-2.2.1 Mayak Storage Facility at Ozersk, Russian Federation

The Mayak facility, completed around 2003, is a central storage facility to provide safe and secure storage of nuclear materials from disassembled Russian nuclear weapons and is operating at Mayak (Chelyabinsk-65). Originally, the storage site was planned as a two-wing facility, which was expected to provide secure, centralized storage for fissile material from approximately 12,500 dismantled nuclear warheads with a service life of 100 years. Using two wings, it was designed to store 50,000 canisters capable of holding 50 metric tons (MT) of plutonium (Pu) and 200 MT of highly enriched uranium. After the 1999 GAO evaluation of the project, the United States decided to bear most of the costs, which had increased from \$275M to \$413M. Design, construction, and specialized equipment for the storage facility were funded, mainly, by the U.S.A., and the design and construction contract was won by Bechtel, a U.S. company. Up until 2004 at least, the construction of the facility was limited to a single wing to store 25,000 canisters. The facility is surrounded by a concrete wall and rows of barbed wire and has three guardhouses. The walls of the facility itself are said to be 8 m thick, and the roof is covered with 4 m of concrete, tar, and gravel. The facility was designed to withstand an earthquake measuring 8 on the Richter scale, and to survive a flood or the impact of a jet plane crash. It is not designed to withstand the effects of a nuclear bomb. The facility area, degree of remote operation, etc., is not available, but the store was described in 2003 as having state-of-the-art security and dose protection (NTI 2009).

E3-2.2.2 Actinide Packaging and Storage Facility (APSF) – Design Study

Department of Energy (DOE) proposed and designed a semi-automated state-of-the-art facility, called the Actinide Packaging and Storage Facility (APSF) to safely store and monitor all excess plutonium for an indefinite (long) period at the Savannah River Site (ReFalo and Wong 1998). Also proposed was a plutonium stabilization facility to provide a near-term disposition pathway for excess plutonium not designated for mixed-oxide (MOX) fuel. Plutonium oxide and metal were the primary materials stabilized and stored in the APSF. Plutonium to be stored in the facility was intended to be placed under international safeguards under the Voluntary Offer Agreement (VOA) with IAEA.

This project, which was suspended in 1999 (Richardson 1999) prior to start of construction, consisted of a hardened, underground material access area totaling 49,300 ft^2 and a 20,000-ft^2 surface concrete utility building. It was sized for 2,000 storage positions with a potential to hold 5,000. Each position would hold a “3013” canister with a loading of 4.5 to 4.8 kg plutonium (or HM). The 3013 designation referred to the DOE Standard (DOE-STD-3013-96), “Criteria for Preparing and Packing of Plutonium Metals and Oxides for Long Term Storage,” which required the packaging of stabilized plutonium in two welded canisters (an inner and an outer canister) for safe storage up to 50 years. DOE-STD-3013-96 is now superseded by DOE-STD-3013-2004, April 2004.

This report assumes the 2,000-canister capacity, which equates to approximately 9.5 MTHM. Although a portion of the facility is used for plutonium oxide calcining and packaging, it is classified as a storage facility.

E3-2.2.3 Consolidated Storage Facility – SRS Design Study

After cancellation of the APSF, DOE conducted a systems engineering evaluation of plutonium material management to determine the benefits of integrating plutonium storage/disposition facilities. The SRS team evaluated the design and construction of a consolidated storage facility (CSF) for DOE surplus non-pit plutonium. This APSF-style, enlarged facility with surveillance capability was the subject of an unpublished study by SRS in 2001 (Boore 2004). The storage capacity was 10,000 storage positions with declared capacity of 45 MT(Pu). Each storage spot was to hold a “3013” canister (DOE 2004) with a similar quantity of HM/canister as APSF (4.5–4.8 kg). Due to variations between the canister capacities of defense and commercial stores, CSF may be taken as a nominal 50 MT(Pu) capacity store for commercial product purposes.

E3-2.2.4 Consolidated Fuel Treatment Center (CFTC) U/TRU Storage Concepts – Design Study

The CFTC U/TRU storage concepts built upon the state of the art APSF and consolidated storage facility designs. The basic design has a below-ground concrete storage vault, which includes storage racks and shield plugs, air inlet, and exhaust shafts, and an above-grade operating area and building support systems, including structures and services required for the building and equipment operations. The vault portion of the building is an underground (~11 ft) reinforced concrete construction structure. The roof slab is nominally 5 feet thick and the walls and floor 3 feet thick. The vault capacities varied with the specific CFTC alternative.

The 3,000 MT/yr Engineering Alternative Study (EAS) for a UREX +1 application had a requirement for 10 years of U/TRU product storage. The design proposed 3 storage vaults with each vault providing storage for nearly 38,000 “elongated” 3013 cans or nearly 170 MT TRU.

The 800 MT/yr Follow-on Engineering Alternative Study (FOEAS) for a UREX + 1 application reduced the storage requirement to 3 years capacity and proposed a 10,000 canister vault similar to the Consolidated Storage Facility.

The 800 MT/yr FOEAS for a Co-Ex application in which only U/Pu is recovered proposed a 3 year capacity with a 10,000 canister vault. The overall dimensions were slightly smaller to reflect the smaller storage container.

The 800 MT/yr FOEAS for a UREX+3 application requires the storage of both a U/Pu/Np oxide and an Am/Cm oxide. The same U/Pu storage described above is required as is an additional storage vault with enhanced decay heat removal systems for the Am/Cm oxide storage.

E3-2.2.5 Unirradiated Fuel Storage Facility at INL[b] (CPP-651) – Concept for Upgrading Existing Facility

The CPP-651 was constructed in 1984 and is used to store un-irradiated fuel. It is an entirely hands-on operation with heightened security capabilities. The hardened area for this facility is 4,960 ft^2. DOE considered the store in 2000 for upgrading to a capability for storage of cans containing 6 kg of Np-237 as NpO_2 for possible future conversion by irradiation of Np to Pu-238. In the event, CPP-651 was not selected for this purpose.

E3-2.2.6 Sellafield Product & Residues Store (P&RS) – U.K.

This robust store for plutonium oxide product and plutonium wastes is nearing completion of construction in 2009. It is a stand-alone facility that will provide storage for cans of products and residues from Magnox and Thermal Oxide Reprocessing Plant (THORP) reprocessing plants, and the Sellafield MOX plant as well as replacement for older Pu stores at Sellafield, U.K. P&RS possesses no facilities for immobilization or other treatment of plutonium material as this is performed in other facilities on the Sellafield site. Historically, Magnox-derived plutonium oxide was stored in aluminum cans each holding about 5.5 kg Pu. THORP-derived plutonium oxide is stored in stainless steel, nested triple canisters, each holding 7.5 kg Pu. The cost of P&RS is quoted as £220M for construction only (Cabinet Office 2009). The Pu inventory on the Sellafield Site is around 100 MT(Pu) and may increase to 130 MT(Pu) as reprocessing continues and depending on the rate of shipment to utility customers.

The new store has a design capacity of 9,600 plutonium canisters and comprises 128 storage modules [each containing 75 canisters ≡ 560 kg (Pu)]. The design lifetime for storage is 50 years extendable to 100 years. The nominal maximum capacity of the store is estimated as about 72 MT(Pu). So far, no cost data has been obtainable for this facility although it should continue to be sought as this store probably represents the newest generation facility constructed in a western nation following a decade of increased international security concerns. Besides the new plutonium store, Sellafield has spent an additional $100M in the period of 2003–2009 on enhanced physical security, contingency planning, and consolidation of nuclear material for transfer to the Sellafield P&RS (Hansard 2009; Cabinet Office 2009).

E3-3. PICTURES AND DIAGRAMS

Figure E3-1 shows a schematic of the design concept of modules of air-cooled arrays of horizontal storage sleeves with shield plugs extending through the concrete wall of the vault bay. A remotely operated stacker retriever moves canisters to or from storage locations (Forsberg 1995).

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[b] Beginning February 1, 2005, the name of the Idaho National Engineering and Environmental Laboratory (INEEL) was changed to Idaho National Laboratory (INL).

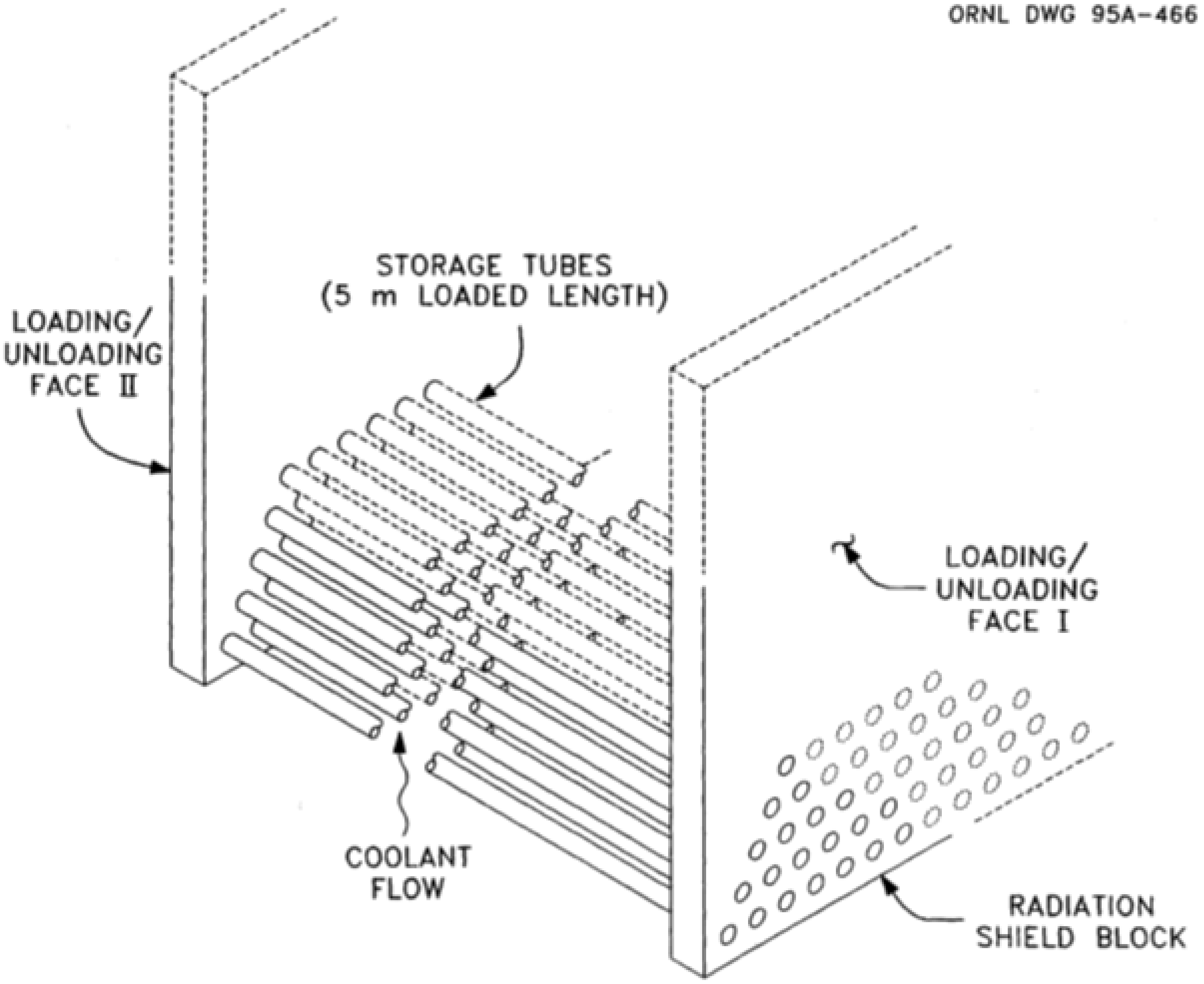

Figure E3-1. Storage module concept using horizontal sleeve geometry within plutonium storage vault.

A buffer plutonium oxide store, see Figure E3-2, is attached to the THORP reprocessing plant at Sellafield in the U.K. It was first operated in 1994. The Pu canister loading/unloading system uses a trolley propelled by chain along a rail with lifting action for package deposition or retrieval. A second trolley provides equipment for in situ inspections of packages. The degree of automation appears quite basic, but is part of the criticality safety case. A cable is used, as necessary, for retrieval of a failed trolley.

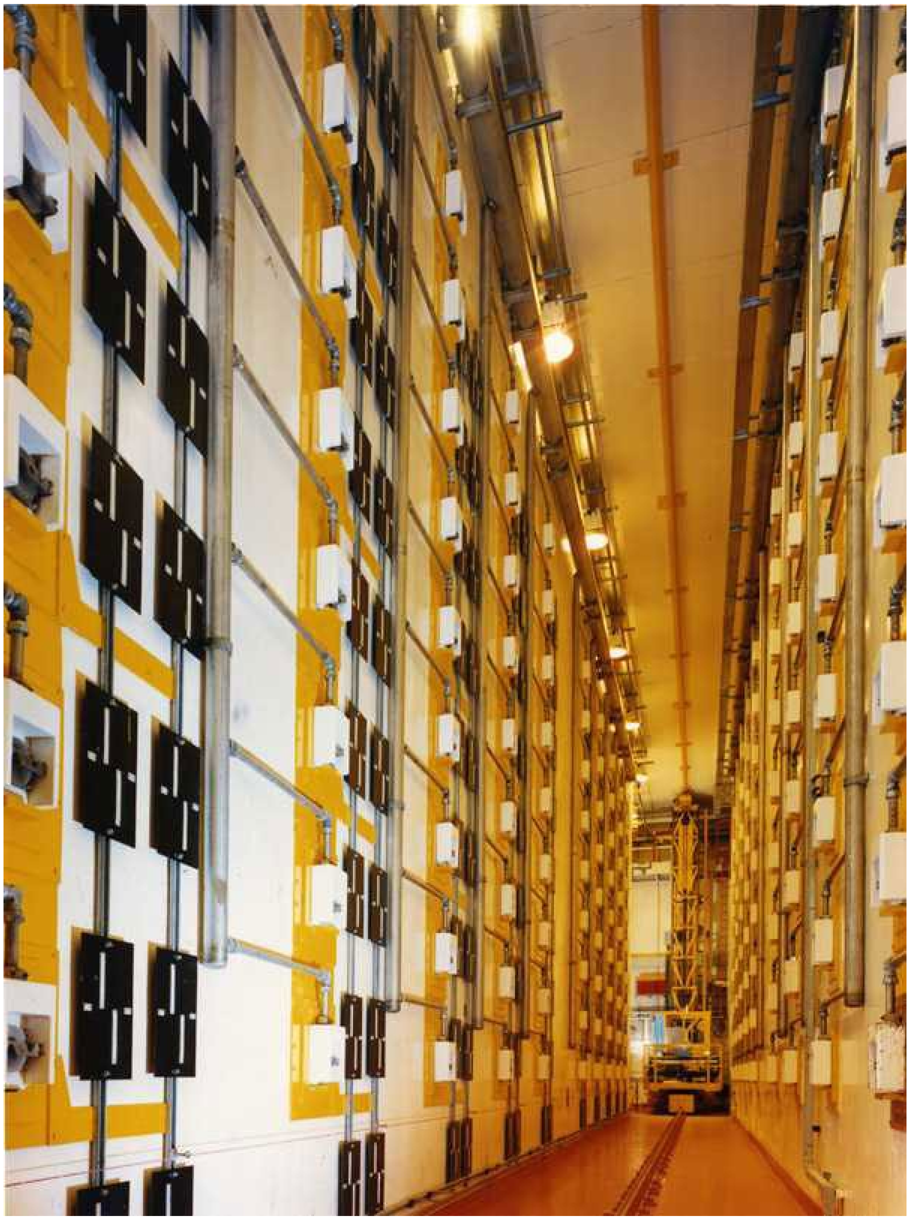

Figure E3-2. Remotely operated store for pure PuO_2 in welded stainless steel canisters in the THORP facility at Sellafield, U.K.

Information on the British Nuclear Fuels plc (BNFL) canister design is given in Table E3-2 (DOE 2004). This canister type is used by the U.S. DOE. A welded closure is preferred because it is believed to provide the best combination of features such as design qualification test performance, ease of assembly under production conditions in a glove box, canister (package) payload capacity, and achievement of a 50-year lifetime. The material canister (convenience canister) is the inner canister that is used to transfer plutonium-bearing material. A material canister is not required in packaging and is not considered an isolation barrier by the DOE Standard (DOE-STD-3013-96). Use of a material canister can reduce the potential for contamination during loading and closure of the “middle” canister, facilitate packaging, and provide an additional material barrier. The specified design pressure of 4,927 kPa (699 psig) for the welded outer canister is sufficient to contain the pressure generated by the mass of oxide specified in

Section 6.3.2 under DOE "worst case" conditions of 0.5 wt% moisture, 19 w heat generation, and 211°C (412°F) gas temperature (DOE 2004). Use of low-carbon stainless steels, such as 304 L and 316 L, is recommended with 316 L being preferable to 304 L because of its greater corrosion resistance.

Table E3-2. BNFL canister volumes.

Component	Interior Volume in liters	Mass in grams	Material Volume in liters	Free Volume in liters
Convenience Can	1.839	1580	v_{cc} = 0.198	V_c = 1.839
Inner Can	2.266	1600	v_i = 0.200	2.068
Outer Can	V_o = 2.602	4026		2.204

Pictures of nested BNFL MOX canisters are shown in Figure E3-3.

Figure E3-3. Nested Stainless Steel BNFL Canisters with Welded Closure for Storage of Pure PuO_2 at Sellafield, U.K. (similar used for DOE at SRS).

For the APSF, the handling of the Pu loaded 316 SST canister with laser-etched bar code in the material accountancy stage was designed to be fully remote. The design for the top of the canister, see Figure E3-4, gave challenges for item handling over a large area and insertion and removal from the measurement instrumentation. The latter did not allow canister gripping from the side and had to be gripped from the top with all tooling, etc., sufficiently small for insertion into the instrumentation channel. Two and three-finger grippers were designed, fabricated, and tested. The design goal of the gripper tooling was to maximize positioning tolerance while maintaining handling capability for the 25-30 lb (11.4–13.6 kg) canister (ReFalo and Wong 1998).

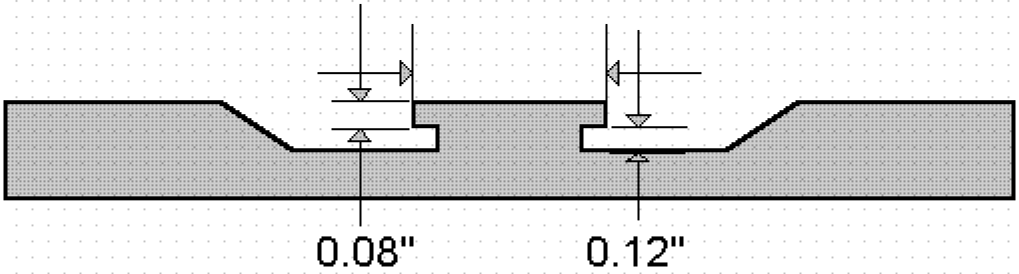

Figure E3-4. Canister top for remote handling of "1330 Canister."

In design and development for the APSF, emphasis was placed on shared use material accountancy as material is loaded. In addition to DOE-STD-3013-96 storage requirements, the facility design met the most recent radiation exposure limits of 500 mrem (5 mSv)/year/employee. A balanced approach in terms of automation, exposure control, processing rates, storage requirements, accountability requirements, general nuclear facility design requirements, manual operation, and cost control was adopted. New concepts are in development, including non-destructive assay (NDA) equipment/software development, automated handling of the canisters, development for new IAEA equipment/software, and testing of the IAEA conceptual safeguard's approach. These features needed mock-up and testing to provide key details for facility design, see Figure E3-5Figure E3-5Figure E3.5.

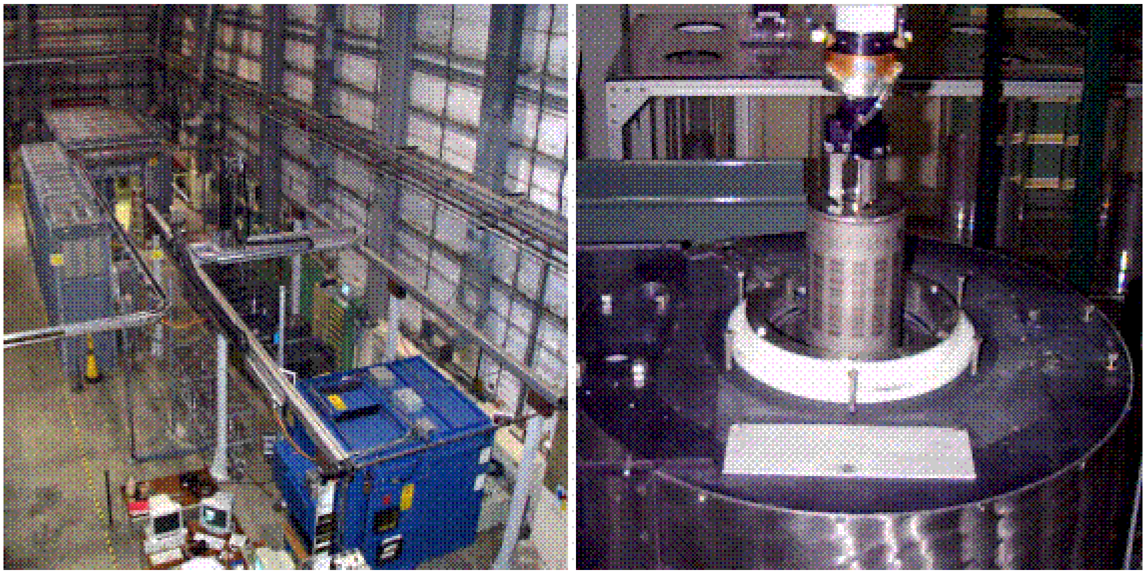

Figure 1. APSF Mockup **Figure 2. Crane Loading ITSC into Neutron Counter**

Figure E3-5. Mockups at SRS for Actinide Packaging and Storage Facility (ReFalo and Wong 1998).

Plutonium metal and oxide have been stored at SRS for several years. In order to be compliant with the DOE 3013 standard, the FB-Line Facility at SRS is packaging plutonium metal and stabilized plutonium oxide into outer canisters using the outer can welder (OCW) system. After welding the weld parameters are reviewed. Next, the outer canisters undergo a helium leak test, see Figure E3-6.

Figure E3-6. Test of leak detector installed in FB-Line Facility at SRS (Hudlow 2004).

Then, the outer canisters undergo a visual examination by an ASME qualified visual examiner. Last, the outer canisters undergo screening by the digital radiography (DR) system for weld porosity (see Figure E3-7) (Hudlow 2004).

Figure E3-7. Test of digital radiography system installed in FB-Line Facility at SRS (Hudlow, 2004).

E3-4. MODULE INTERFACES

The actinides that would be stored in these facilities would be received from “Aqueous Reprocessing,” Module F1, or “Electrochemical Reprocessing,” Module F2/D2. One path out of this module is Module O1, “Shipment of Materials to a Mixed Oxide Fuel Fabrication Facility,” (if fuel fabrication is not integrated at the reprocessing facility). Depending on the fuel cycle eventually adopted, further path options include waste management, such as Modules G, I, F2/D2, L, M, and O Series.

E3-5. SCALING CONSIDERATIONS

To report the costs in near present day (2005) dollars, the Engineering News Record Construction Cost Index and Building Cost Index were used to escalate the Work Breakdown Structure Level 2 costs to 2005. Work Breakdown Structure 1 and 3–7, where available, were escalated to 2005 using the U.S. Department of Labor, Bureau of Labor Statistics, Consumer Price Index—All Urban Consumers. Escalation to 2007 for all Work Breakdown Structure levels was also based on this index. Further escalation was performed using the U.S. Army Corps of Engineers, 2007, Civil Works Construction Cost Index System (CWCCIS), using CWBS Feature Code – 07 Power Plant (U.S. Army Corps of Engineers 2009). Table E3-3 summarizes information for five selected facilities; also see the cost data from Section E3-6.

Table E3-3. Recycled product storage summary.

Store	Mayak, R.F.	APSF	CPP-651	CSF	P&RS, UK
Status	Operated 2004	Design, 1995	Not used	Design, 2001	Construction
Hardened Area	Unknown	49,300 ft^2	4,960 ft^2	Unknown	Unknown
No. of Canisters	25,000	2,000	-	10,000	9,600
Storage Capacity	100 MT(Pu)	9 MT(Pu)	1 MT(Pu)[a]	45 MT(Pu)	72 MT(Pu)
Total of WBS Levels 1–6 in 2005 $K	$570,600	$319,300	$9,320	$634,000	~$400,000 (capital)
Capital Cost/Unit of Material	$5,700/kg	$35,500/kg	$9,320/kg	$14,100/kg	$5,500/kg[b] ($9,200/kg)
Store	CFTC EAS UREX+1	CFTC FOEAS UREX+1	CFTC FOEAS U/Pu	CFTC FOEAS UREX+3	
Status	Design Study	Design Study	Design Study	Design Study	
Hardened Area	475,000	28,000	24,000	52,000	
No. of Canisters	114,000	10,000	10,000	20,000	
Storage Capacity	500 MT TRU	44 MT TRU	44 MT Pu	44 MT TRU	
Total of WBS Levels 1–6 in 2007$	$6.5B–9.0B	$0.7B–1.0B	$0.65B–0.9B	$1.5B–2.1B[d]	
Capital Cost/Unit of Material	$13,000–$18,100/kgTRU	$17,800–$24,700/kg TRU	$16,100–$21,700/kg TRU	$37,000–$51,400/kg TRU	

a. Capacity estimated from the APSF on the basis of hardened area.

b. This figure is based on 7.5 kg/canister, which becomes $9,200/kg based on 4.5 kg/canister as used by APSF and CSF. P&RS does not include plutonium immobilization as do APSF and CSF. P&RS has eight-fold greater capacity than APSF and 60% greater capacity than CSF.

c. Figure includes a distributed share of the infrastructure and balance of plant cost associated with the reference reports WSRC 2007 and WSRC 2008.

d. Includes the cost of two vaults one for the U/Pu oxide and a second for the Am/Cm oxide.

E3-6. COST BASIS, ASSUMPTIONS, AND DATA SOURCES

E3-6.1 Basis and Assumptions

The facilities, other than the CFTC design studies, referenced here are “stand-alone” operations, not dependent on other facilities for shared services, except utilities including transport. It is not known how these estimated costs were developed; “top-down” or “bottoms-up,” except the capital costs for Mayak and CPP-651, are reported as the actual costs. The latter is likely to have needed substantial upgrading expenditure for remote handling, improved cooling, modern dose standards, enhanced material controls, and accountancy for the proposed new storage mission. Furthermore, the designs of the facilities are different and include other functions, especially some with remote material stabilization and canister welding and inspection.

The CFTC design study estimates are “bottoms-up” estimates. The facility is assumed to be collocated on the reprocessing site (within the security PIDAS) and a portion of the site infrastructure and balance of plant costs have been allocated to this facility segment. No functions other than storage are conducted in this building.

Since this module may receive for storage a mixed U-TRU product, the relative cost difference due to storage of a product high in U content versus a pure TRU product is needed. The relative cost factor evaluation shown in Table E3-4 is generally based on the commercial store receiving intact and sealed canisters, returning defective ones, and eventually exporting intact cans to downstream plants. For uranium, the heat emission is negligible and realistic changes hardly affect store cost. The relative cost per kilogram for storage of reprocessed LWR 1% ^{235}U uranium product, although low, is judged to be several times greater on a mass basis than for reprocessed 0.2% ^{235}U product (e.g., Gen I gas reactors). This is due to the greater number of drums, the increased shielding required, the greater number of storage positions, the need for outrigger spacers (“bird cage”) for geometric safety, and the greater degree of remote handling. The unit costs of storage for pure uranium trioxide are judged to be orders of magnitude lower than for materials with high transuranic contents. For this module, the effects of uranium storage are mainly important when contributing to mass/canister increases to mixed actinide storage.

The OECD-NEA estimated unit charges for commercial plutonium storage, which have the following broad technical context (OECD-NEA 1994). A representative commercial PUREX plant with LWR oxide spent nuclear fuel (SNF) throughput of 800 MT(HM)/year has a plutonium arising rate of about 1% of initial heavy metal input to reactor, which equates to 8,000 kg(Pu)/year or 1,100–1,500 PuO_2 canisters/year [~5 canisters/working day with ~6 kg(Pu)/canister]. For a steady-state buffer store of ~8,000 canister [~50 MT(Pu) capacity servicing a typical PUREX oxide recycling plant, this then gives a mean residence time of 6 years, which would be greater than the expected buffer period for a well-developed closed nuclear fuel cycle near steady state, but not excessive for present conditions where thermal MOX fuel for use in LWRs remains the exception for utilities. [Note that the CSF (defense store) assumes 4.5 kg(Pu)/canister, whereas commercial fuel cycle stores with different Pu feed may use values of 5.5–7.5 kg(Pu)/canister]. For recycling plutonium oxide to current designs of thermal or fast reactors, without the presence of the minor (higher) actinides, a period of about 2-years storage (preferably ≤1 year) post reprocessing may be the upper limit prior to fuel fabrication using hands-on techniques in gloveboxes and the commencement of irradiation in reactor. This is because fuel/power grade plutonium materials have Pu-241 contents of 1.6%/12% that decay by β-decay to Am-241 with a half-life of 13.2 years and increase dose levels. OECD-NEA quotes the possible requirement for purification of long-stored plutonium due to americium in-growth. The need depends on the source of Pu, period of storage, and design of MOX fuel fabrication plant. The cost range is given as $10–28/g(Pu) with a value of $18/g(Pu) for a 2,000 kg(Pu)/year plant (OECD-NEA 1994).

For advanced fuel cycles with burning of minor actinides, the recycle fuel is likely to need to be fabricated remotely and remotely handled during reactor refueling. The period of storage of plutonium, and possibly uranium, mixed with minor actinides (MA) is less significant although clearly, due to cost, all storage for recycle should be minimized unless it contributes to optimization of waste management. The nominal value assumed here for storage of Pu-MA-U is 4 years.

It is seen that commercial Pu or Pu-MA-U stores in a developed fuel cycle may have continuous loading and unloading of fissile material, which implies a higher level of operations than for defense stores, which may load for a period (e.g., 5 years) and maintain storage for a long period (e.g., 40 years and then unload), for example, over the next 5 years.

The relative storage unit capital cost (cost per kg) for high fissile material such as pure plutonium and high plutonium blends with transuranic (TRU) minor actinide and uranium elements are judged in Table E3- 1 as based on the two multiplying factors relative to the base case of storage of pure PuO_2 in canisters as was widely used for the plutonium product of commercial PUREX reprocessing of electric utility fuels. The extrapolating factors are based first on the likely increased numbers of canisters and storage positions, and second on the increased radioactivity, dose, ventilation, handling, and criticality safety requirements.

Table E3-4. Judged storage cost increase factors relative to depleted UO_3 and PuO_2.

Stored Material	Relative Factor-Drums/Canisters Needed	Relative Factor—Heat, Activity, Dose, Handling	Store Cost Factor— c. f. kg(U) or kg(Pu)	Nominal Period of Storage, Years
UO_3 drums (0.2% 235U)	1 (800 kg Magnox GCR)	1	1	-
UO_3 drums (1% 235U)	3 (180 kg THORP LWR)	2	5	-
PuO_2 canisters	1 (4.4 kg)[c]	1	1	2
UO_3-PuO_2 (50%)	1.2[d]	1	1.2	2
$TRUO_2$ (33%)-UO_3	1.5[d]	1.3	2[e]	4
CmO_2 (100%)	10[f]	2	20[g]	~100

E3-6.2 Storage Facility Costs

E3-6.2.1 Mayak Fissile Material Storage Facility

The Mayak fissile material storage facility was reported to have cost $458M U.S. ($413M provided by the U.S. Department of Defense [DOD] and $45M provided by Russia), with a cost of $223M (≡ 12% of capital/year) to load the facility with the actinide product. This covered the construction of the first wing only. Loading the facility began in July 2006 and is anticipated to take 4 years to complete. Annual operating costs thereafter were estimated by a U.S. contractor to be $13M. The operating cost during storage is judged to be low and perhaps represents staff costs, which are much lower than U.S. costs. These costs are in 1999 dollars. As of May 1999, the Fissile Material Storage Facility Budget through 2001 stood at $397.6M. That amount is broken down in Table E3-5 (NTI 2009):

Table E3-5. Mayak store cost breakdown ($M).

Design	$9.1
Construction	$175.0
Equipment Purchases & Installation	$171.5
Transportation	$6.5
Cooperative Threat Reduction Logistics Support	$2.1
Project Support	$33.3

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[c] DOE-STD-3013-2004, p. 9, "The total mass of plutonium and other fissile species within either metal or oxide contents shall not exceed 4.40 kg (9.70 lb.). The total mass of the package contents, whether metal or oxide, shall not exceed 5.00 kg (11.02 lb.)." Commercial reprocessors may use higher values (5.5–7.5 kg of plutonium depending on fuel burnup, etc.).

[d] EAS studies assumed the 4.4 kg TRU per container will apply and the total mass could be increased so this factor is 1.0

[e] EAS studies used a factor of about 1.2 based on the ratio of the hardened facility area from Table E3-3

[f] EAS studies used a factor of 4.4/0.25 or 17 for Am/Cm oxide

[g] EAS studies used a factor of about 1.2 based on the cost per unit area

An additional $15M was spent on the early design of the facility.

Taking the unloading cost to be the same as loading and taking 42 years of operation at $13M/year (~3% of capital per year), and adding 20% for decommissioning and contingencies, the lifecycle cost (LCC) is found to be ~$1.45B, which is approximately 3.2 times the total project cost (TPC), which broadly equates to capital cost. The total cost for work breakdown structure (WBS) Levels 1–6 is given in Table E3-3 as $570.6M in 2005 dollars.

E3-6.2.2 Actinide Packaging and Storage Facility (APSF)

The reported capital cost is $330M in 2000 dollars.

E3-6.2.3 Unirradiated Fuel Storage Facility (CPP-651)

Its reported capital cost in 1984 was $5.16M. Significant costs for upgrading are judged likely to achieve modern standards for plutonium and greater for Pu-MA-U storage.

E3-6.2.4 Consolidated Storage Facility (CSF)

The SRS CSF design study derived a rough order of magnitude (ROM) cost estimate in 2005 U.S. dollars of $600M in capital and $75M/yr for operating expense (≡12.5% of capital/year) throughout the period of operation.

E3-6.2.5 CFTC U/TRU Storage Facilities

The SRS EAS derived four estimate ranges for various product storage vaults and capacities including:

- $6.5B–9.0B for a U/TRU oxide product vault with a capacity 500 MT of TRU
- $0.7B–1.0B for a U/TRU oxide product vault with a capacity of 44 MT of TRU
- $0.65B–0.86B for a U/Pu oxide product vault with a capacity of 44 MT of Pu or Pu/Np
- $0.68B–0.90B for an Am/Cm oxide product vault with a capacity of 2.4MT of TRU.

In addition the EAS determined the steady-state operations cost (continuous loading and unloading) operations as:

- $26M–39M/yr for a U/TRU oxide product vault with a capacity 500 MT of TRU
- $7M–10M/yr for a U/TRU oxide product vault with a capacity of 44 MT of TRU
- $7M–10M/yr for a U/Pu oxide product vault with a capacity of 44 MT of Pu or Pu/Np
- $7M–11M/yr for an Am/Cm oxide product vault with a capacity of 2.4MT of TRU.

E3-6.2.6 Sellafield Product & Residues Store (P&RS)

The cost of Sellafield P&RS has been quoted as £220M (pounds sterling) (Cabinet Office 2009). This is for construction only and does not include operation, decommissioning, or interest during construction. The money value year was not given, but much of the construction has been completed and was performed during the past 5 years when the average pound sterling exchange rate was about 1.00£:US$1.80. A broad capital cost value of $400M (2007 dollars) is estimated. The P&RS specific capital cost is lower than APSF or CSF, due partly, it is judged, to the larger canister inventory used by P&RS, the greater economy of scale of P&RS (60% greater capacity than CSF), and the smaller scope with no plutonium immobilization plant. The anticipated operating costs of P&RS have not been disclosed. After adjustments, the P&RS capital cost is considered to be broadly consistent with that for CSF (see Table E3-3).

E3-6.3 Cost Correlations

Figure E3-8 shows the capital costs of four facilities in Table E3-3. A power trend line to these data was determined to be:

Capital cost (millions of 2007 dollars) = 16.123 (Pu Capacity in MT) $^{0.912}$.

This is based on a very small data sample, but indicates that some benefits of scale may accrue to this type of facility. A somewhat smaller exponent showing greater economies of scale might be expected. The low cost of CPP-651, its age, and the lack of adjustment of the capital cost to represent a totally remote-operated facility is considered to unrealistically skew the correlation downwards near the origin.

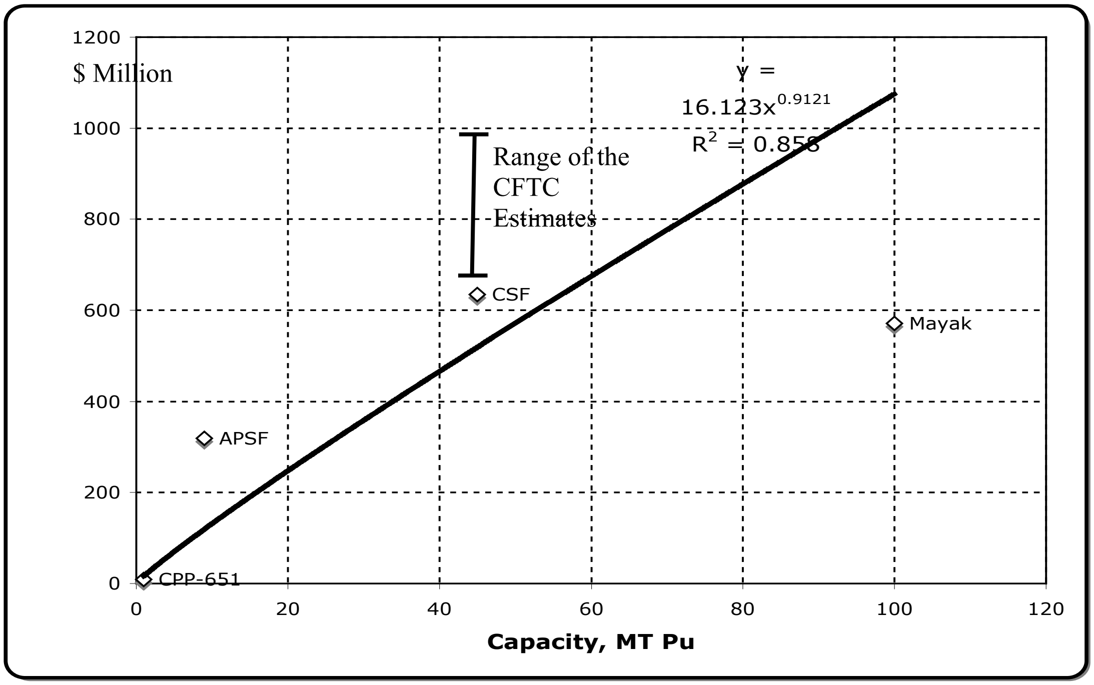

Figure E3-8. Costs of various sizes of plutonium storage facilities.

Using just the data for APSF and the CSF as relevant to the U.S. and DOE environment (both design studies performed by SRS), a linear equation can be derived:

Capital cost (millions of 2007 dollars) = 241 + 8.74 × (Pu Capacity in MT)

It is not unreasonable that the line intersects the cost axis at several hundred million dollars as the storage capacity tends toward zero, since the facility would still comprises a wide range of capabilities and operations including receipt, handling, MC&A, security, ventilation, health physics, maintenance, inspection, etc., systems while omitting storage modules.

From the reasonable consistency of these data, use of the CSF case and its cost data is selected for further analysis and comparison with the overall unit cost information provided by OECD-NEA.

Using data from the CFTC studies the cost was fitted using the logarithmic relationship:

$$CostofA = CostofB\left(\frac{CapacityofA}{CapacityofB}\right)^n \qquad (1)$$

Where, capacity is expressed as instantaneous design capacity (MT/yr), and the exponential factor is typically in the range of about 0.6. However, due to the inherently high structural costs associated with highly shielded and remotely operated nuclear facilities not found in commercial operations, the power law exponent is expected to be less than 0.6. The preceding equation indicates that a log-log plot of the capacity versus cost should be a straight line with the slope equal to the exponent. Therefore, the CFTC U/TRU vault estimates for different capacities shown in Table E3-3 were used to determine the power law factor was equal to about 0.41 over the range of storage capacities from 44 to 500MT.

E3-6.4 Representative U.S. Plutonium Storage Unit Cost Estimate

Based on the above subsections of E3-6, cost estimates given here use CFTC CO-EX case as the representative U.S. Pu Store Design. The TPC, broadly capital cost, of the CFTC U/Pu oxide storage value was given as $650M to $875M with annual operating cost of $7M to $11M (1% of capital/year) with store capacity of 10,000 canisters [45 MT(Pu) defense], Costs were reported in 2007 dollars. The EAS Studies also reported a 40 year operational Operations and Maintenance (O&M) cost of $360M to $540M when combined with the D&D and TPC gives a total LCC of $1.1B to $1.5B or a levelized cost of $25,000 to $35,000 per kg of Pu for 40 years or $0.62 to $0.90/g(Pu)-year

This is about half the OECD-NEA unit values for commercial plutonium storage charges of $1.5–2.9/g(Pu)·yr in 2007 dollars. The difference may be due to the assumptions of the storage vault being collocated with the reprocessing center. It should be noted that the O&M costs were considerably higher (75M/yr) for the stand alone CSF. If this O&M value is used then the LCC becomes $3.7B to $4.0B or $84,000 to $91,000/kg Pu or $2.10 to $2.30 g(Pu)-year within the range of the OECD-NEA estimate.

E3-6.5 OECD-NEA Unit Charge Estimates for Plutonium Storage

The OECD-NEA report, “The Economics of the Nuclear Fuel Cycle,” Section 4.3.2.6, Plutonium Storage, p. 40, 1994, states that the costs for Pu storage varies widely between countries due to store size and other factors and are usually taken to be in the region of $1 to $2 per gram (1994 dollars) of total Pu per year.

Adjusted to 2007 dollars, the unit plutonium storage charge values are $1.5–$2.9/g(Pu)·yr. This forms an essentially linear capacity versus cost correlation. OECD-NEA also states, “Both BNFL (now NMP) and COGEMA (now AREVA) include the cost of short-term storage as a minor component of the overall recycling price, but some countries requiring longer-term storage are incurring additional prices of this order.” As a price, this includes the reprocessor’s profit element for the service provided.

It was anticipated that operational and maintenance costs for loading/unloading would be appreciably higher from those for long-term storage of essentially undisturbed material. Estimates of operational costs of 10–12% of capital per year during buffer storage with steady filling/emptying and significantly <10% of capital per year during long-term storage, where Pu inventories are retained in full stores pending potential future recycling, seemed reasonable. However, the SRS cost estimates and the OECD-NEA report do not justify this. The low operating cost value for steady long-term storage at Mayak may be due to low wages in the Russian Federation (R.F.) as compared to the higher capital cost of a U.S. designed and equipped plant.

E3-6.6 Converting OECD-NEA Unit Charge to Capital and Operational Costs

In this subsection, OECD-NEA unit plutonium storage charges are broken down into corresponding capital and operational elements. For example, taking the midrange OECD-NEA overall estimate (1994 dollars) of \$1,500/kg(Pu).year cost as based on a 50 MT(Pu) store operated for 50 years and carrying out a broad consistency check.

Overall charge of \$1,500 × 50,000 kg (Pu) × 50 year = \$3.75B (1994 dollars) ≡ \$5.4B (2007 dollars), where 50,000 kg (Pu) at 5 kg (Pu)/container = 10,000 containers.

The period of steady loading/unloading and buffer storage is taken as 50 years at 10% of capital/year ≡ 500% total. The capital cost is then \$5.4B × 10/60 = \$900M (2007 dollars). The unit capital cost is \$900M/50,000 kg (Pu) = \$18,000/kg (2007 dollars).

In undiscounted terms, this gives total operating and capital costs of 600% of capital only (i.e., Annual operating cost = \$5.4B/60 ≡ \$90M [2007 dollars]). This equates to ~450 overall staffing (day/back shifts, radiation workers, physical protection, management, etc.) as based on \$200K/staff, which includes all other elements of operation such as utilities, materials, contracts, projects, etc.

E3-6.7 Comparing OECD-NEA and SRS Cost Estimates

The cost comparisons for the two 50 MT (Pu) stores, each operating for 50 years, are shown in Table E3-6.

Table E3-6. Capital, operational, and lifecycle costs (2007 dollars).[a]

Pu Store Cost Basis	Capital Cost Unit Capital Cost	Operational Cost Unit Operational Cost	Lifecycle Cost Averaged Unit Cost[e]
OECD-NEA, 1994[c] Mid cost range; 50 MT	900 \$M \$18,000 \$/kg(Pu)	90 \$M/yr \$1,800 \$/kg(Pu)·yr	5.4 \$B 2,200 \$/kg(Pu)·yr
CSF Cost Basis[d] CSF scaled to 50 MT	750 \$M \$15,000 \$/kg(Pu)	94 \$M \$1,900 \$/kg(Pu)·yr	5.4 \$B 2,200 \$/kg(Pu)·yr
CFTC U/Pu Vault Cost Basis scaled to 50MT[f]	\$712M–960M \$14,300–\$19,200/kg Pu	\$9M–14M/yr \$180–\$280/kg (Pu)-Yr	\$1.25B–1.7B \$500–\$680/kg (Pu)-Yr

a. Cost escalation derived using: U.S. Army Corps of Engineers, CWCCIS, Using CWBS Feature Code – 07 Power plant, Appendix A, EM 1110-2-1304, Appendix Revised September 30, 2007. Values used: \$1 (1994 dollars) ≡ \$1.31 (2005 dollars) ≡ \$1.38 (2006) ≡ \$1.47 (2007) ≡ \$1.54 (2008) ≡ \$1.56 (2009).

b. Capital and operational expenditure (decommissioning not explicitly included).

c. OECD-NEA, "The Economics of the Nuclear Fuel Cycle," Section 4.3.2.6, Plutonium Storage, 1994, using mid-range overall cost value of \$1.5 (1994 dollars)/g(Pu).year.

d. Capital cost of U.S. SRS design for CSF of \$600M with annual operating cost of \$75M with store capacity of 45 MT(Pu). Costs in 2005 dollars (Boore 2004).

e. Undiscounted sum of costs divided by storage capacity times facility lifetime.

f. Capital cost determined using the logarithmic equation using the 0.41 power factor

For a 50 MT(Pu) capacity store, the unit storage charge range given by OECD-NEA of \$1-2 g(Pu)/year (1994 dollars) ≡ \$1.5–2.9 g(Pu)/year (2007 dollars) implies a store capital and operating cost range of \$3.7B–7.4B (2007 dollars) with reasonable assumptions for operating costs over 50 years. At the same capacity, the SRS design for the U.S. CSF for plutonium oxide has a store capital and operating cost estimate of \$5.4B (2007 dollars) over the same period while the CFTC design has a lower operating cost and LCC of \$1.7B. The unit capital costs for the OECD-NEA and SRS cases are similar in the range of \$14,000 to \$19,000 per kg(Pu) The unit operational costs for the OECD-NEA and CSF cases are also similar at \$1,800 and \$1,900 per kg(Pu).year, respectively while the CFTC O&M costs are an order of magnitude lower. The CSF cost estimate is seen to be in the middle of the OECD-NEA cost range. OECD-NEA is using commercial industry data, which may be more broadly based from worldwide nuclear operators. Also, OECD-NEA does comment on the wide spread of costs. However, design requirements, particularly relating to safeguards, physical security, and non-proliferation aspects for storage of fissile material, have increased since the early 2000s and may further increase. The lower operational costs in the CFTC studies indicate the magnitude of the stand-alone versus collocated assumption.

E3-6.8 Selected Values for Pu and Pu-U Storage Unit Costs

There is good consistency of the OECD-NEA cost range of \$1.5–2.9/g(Pu)·yr in 2007 dollars for plutonium oxide storage and the \$2.2/g(Pu)·yr in 2007 dollars for plutonium/plutonium oxide storage of the SRS Consolidated Storage Facility design study. The CSF costs are consistent with those for the APSF and reasons have been given that account for the lower cost values shown by Mayak and CPP-651.

Accordingly, a cost value of \$2,200/kg(Pu)·yr in 2007 dollars for commercial storage of pure plutonium oxide is adopted. A 1-year period is taken as the minimum for cost purposes, and costs for longer periods are estimated using the product of the storage time and yearly rate. For stored Pu-U oxide and using the factor 1.2 from Table E3-4, the nominal storage cost for a Rokkasho-type mixed oxide (Pu-U oxide) product for is estimated as \$2,600/kg(Pu)·yr. The nominal storage period is taken as 2 years, which gives a storage cost of \$5,200/kg(Pu) for MOX (Pu-U mixed oxide). Note that the unit cost is based on the plutonium mass rather than the combined Pu and U mass.

E3-6.9 Selected Values for Pu-Minor Actinide-Uranium Costs

Using a factor 2 multiplier derived from Table E3-4, a value of \$4,400/kg(Pu-MA)·yr in 2007 dollars for storage of mixed plutonium, transuranic minor actinides, and uranium is adopted. A 1-year period is taken as the minimum for cost purposes, and costs for longer periods are estimated using the product of the storage time and yearly rate. The nominal storage period is taken as 4 years, which gives a nominal storage cost of \$17,600/kg(Pu-MA) for MOX (Pu-MA-U mixed oxide). Note that the unit cost is based on the sum of the plutonium and minor actinide mass rather than the combined Pu, MA, and U mass.

The reference costs are approximately 5 times those used in the CFTC U/TRU vault studies, which ranged from \$700 to \$1000/kg(TRU)·yr. The primary difference between the estimates is that the CFTC assumes an integrated storage facility versus the reference cost is based on a stand-alone storage facility. Estimates for an integrated storage facility are estimated at one fifth of the reference costs.

E3-7. DATA LIMITATIONS

The information obtained for these facilities is at a very high level. Details were considered proprietary and not available as of this writing. The technology readiness level (TRL) was considered to be “viable” or “commercial” for the facilities APSF and CSF because of the existence of operating facilities. Mayak, NDA Sellafield, and AREVA Cap La Hague are existing operating facilities.

OECD-NEA is using commercial industry data, which may be more broadly based from worldwide nuclear operators. Also, OECD-NEA does comment on the wide spread of costs. But design requirements, particularly relating to safeguards, physical security, and non-proliferation aspects for storage of fissile material, have increased since the early 2000s and are likely to continue. The nominal cost value for storage of PuO_2 is therefore judged to lie above the minimum value established by OECD-NEA in 1994 for stores constructed and operating in the 1980–1990s period. It is equated to the undiscounted unit storage cost from CSF data developed by SRS and is less than the OECD-NEA upper range value.

E3-7.1 Mayak

Russian craft wages are considerably less than in the U.S.A., although productivity is historically much lower than in the U.S.A.

E3-7.2 APSF

The overall technology incorporated into this facility is not considered "new technology"; some specially engineered equipment is included. Although a portion of the facility is used for plutonium oxide firing and packaging, it is classified as a storage facility. The relatively small storage capacity of this concept would tend to drive up the per-MTHM-cost.

E3-7.3 CFTC Studies

The CFTC study technology incorporated into this facility is not considered "new technology"; some specially engineered equipment is included. The facility is dedicated to the storage mission. The O&M costs appear low compared to other studies and most likely reflects the difference between a stand alone facility and one co-located with a reprocessing facility.

E3-7.4 CPP-651

Because of its hands-on operation, it does not compare well with the above two examples. The secure nature of this facility limits the amount of information available on storage capacity and operating costs. No attempt was made to adjust the capital cost to represent a totally remote-operated facility.

E3-7.5 CSF

This "APSF-style" facility was the subject of an unpublished study by SRS in 2001. The estimate was considered rough order of magnitude.

E3-7.6 P&RS

The construction cost value does not include money value year and the facility has smaller scope than the SRS studies. P&RS stores pure commercial product in contrast with SRS studies for defense fissile materials.

E3-8. COST SUMMARIES

The module cost information is summarized in the What-It-Takes (WIT) cost summary in Table E3-7. The summary shows the reference capital cost basis (constant year U.S. dollars), the reference basis cost contingency (if known), the cost analyst's judgment of the potential upsides (low end of cost range) and downsides (high end of cost range) based on references and qualitative factors and selected nominal costs (judgment of the expected costs based on the references, contingency factors, upsides, and downsides). These costs are subject to change and are updated as additional reference information is collected and evaluated, and as a result of sensitivity and uncertainty analysis. Refer to Section 2.6 in the main section of this report for additional details on the cost estimation approach used to construct the WIT table.

The following cost values are proposed, which are summarized in Table E3-7:

1. The reference cost for plutonium oxide storage is derived as \$2,200/(kg(Pu)·yr) in 2007 dollars for buffer storage using a 50 MT(Pu) capacity store of recycled pure plutonium oxide product based on U.S. security requirements. For cost purposes, a minimum 1-year storage is assumed. The cost of

storage is estimated as the product of storage time and unit storage cost. The latter covers both capital and operating costs. A nominal period of 2 years storage is expected for a well-developed conventional PUREX fuel cycle.

2. The reference cost for plutonium-uranium mixed oxide storage is estimated as \$2,600/(kg(Pu)·yr) in 2007 dollars for buffer storage using a 50 MT(Pu) capacity store of recycled mixed plutonium-uranium oxide product based on U.S. security requirements. For cost purposes, a minimum 1-year storage is assumed. The cost of storage is estimated as the product of storage time and unit storage cost. The latter covers both capital and operating costs. A nominal period of 2 years storage is expected for a well-developed fuel cycle. Storage of plutonium as the mixed oxide with uranium is the expected approach for most recent and near future PUREX reprocessing plants.
3. A nominal cost value of \$4,400/(kg(Pu-MA)·yr) is estimated corresponding to buffer storage of the proposed PuMAOx(33%wt.)-UO_3 advanced fuel cycle fissile product material using a 50 MT(Pu+MA) capacity store based on U.S. security requirements. For cost purposes, a minimum 1-year storage is assumed. The cost of storage is estimated as the product of storage time and unit storage cost. The latter covers both capital and operating costs. A nominal period of 4 years storage is expected for a well-developed fuel cycle. This corresponds to a doubled storage cost per transuranic actinide mass and a doubled storage period compared to the reference pure plutonium oxide commercial buffer storage. Storage of plutonium as the mixed oxide containing minor actinides and uranium is one of the main approaches [homogeneous recycle of PuMAOx(33%wt.)-UO_3 material proposed under the AFCI/FCRD programs] for the advanced fuel cycle.
4. The high-range estimate of \$6,000/(kg(Pu–MA)·yr), corresponding to buffer storage of proposed PuMAOx (33%)-UO_3 material, reflects a greater number of canisters due to lower material loadings and higher decay heat and radiation levels, over the nominal cost estimate. The low range cost value is assessed as \$3,300/(kg(Pu-MA)·yr). It might correspond to the cost value for long (e.g., 50–100 years) storage of Pu-MAO_2-UO_3 inventories where operating costs might possibly be reduced as loading/unloading operations become a less-significant part of overall costs.
5. Higher cost values/(kg(Cm)·yr) for curium storage due to exceptionally high decay heat and need for dilution in engineered glass and/or very small canisters.

Table E3-7. Cost summary What-It-Takes (WIT) table of selected values for storage of mixed recycled plutonium, minor actinides, and uranium (**including escalation to 2020$).**

	$30,000/kg(Pu-MA)[1]				
Reference Unit Capital Cost based on 50 MT(TRU) Capacity	Reference Cost Contingency (+/- %)	Low Cost $/(kg(Pu-MA)·year)	Mode Cost $/(kg(Pu-MA)·year	Mean Cost $/(kg(Pu-MA)·year	High Cost $/(kg(Pu-MA)·year
From 2009 & 2015 CBAs Stand-Alone Collocated[3].	(± 25%)	$3,300 $660	$4,400 $880		$6,000 $1200
Mod. E3-1a 2020 $ "stand-alone" Mod. E3-1b "co-loc"(2020$)		$3937 $715	$5249 $953	$5448 $989	$7158 $1300
	$18,000/kg(Pu)[2]				
		Low Cost $/(kg(Pu)·year)	Mode Cost $/(kg(Pu)·year)	Mean Cost $/(kg(Pu-MA)·year	High Cost $/(kg(Pu)·year)
From 2009 CBA & 2015 CBAs Stand-Alone Collocated[3].		$2,000 $400	$2,600 $520		$3,300 $660
Mod. E3-2a 2020 $ "stand-alone" Mod. E3-2b "co-loc"(2020$)		$2386 $433	$3102 $563	$3142 $570	$3937 $715
		Economy-of-scale for larger capacity facility, increased Pu/MA loading of canisters, or shared safeguards and security functions	For other periods use product of unit storage cost and storage time; minimum period of 1 year.		Increased regulatory/safety requirements, ROM estimate as partial basis

1. For storage of PuO_2-MAO_2-UO_3 in remote handling facility (see Table E3.4; Store cost factor = 2). 50-year facility life.
2. For storage of mixed PuO_2-UO_3 in remote handling facility (see Table E3.4; Store cost factor = 1.2). 50-year facility life.
3. Co-located facility costs based on 1/5 of the stand-alone facility costs.

[A factor of 19.3% was used to escalate from 2009$ for stand-alone facilities. A factor of 8.3% was used to escalate from 2015$ for co-located facilities]

The triangular distributions based on the costs in the WIT Table E3-7 are shown in Figures E3-9 and E3-10. Figure E3-9 gives the estimated cost frequency distribution for the advanced fuel cycle product from, for example, UREX+ processing (i.e., mixed plutonium, minor actinide, and uranium oxide material). The costs for the similar component metallic product from molten salt electrochemical processing are less well known, and until further work is performed may be taken as similar to the UREX+ oxide product.

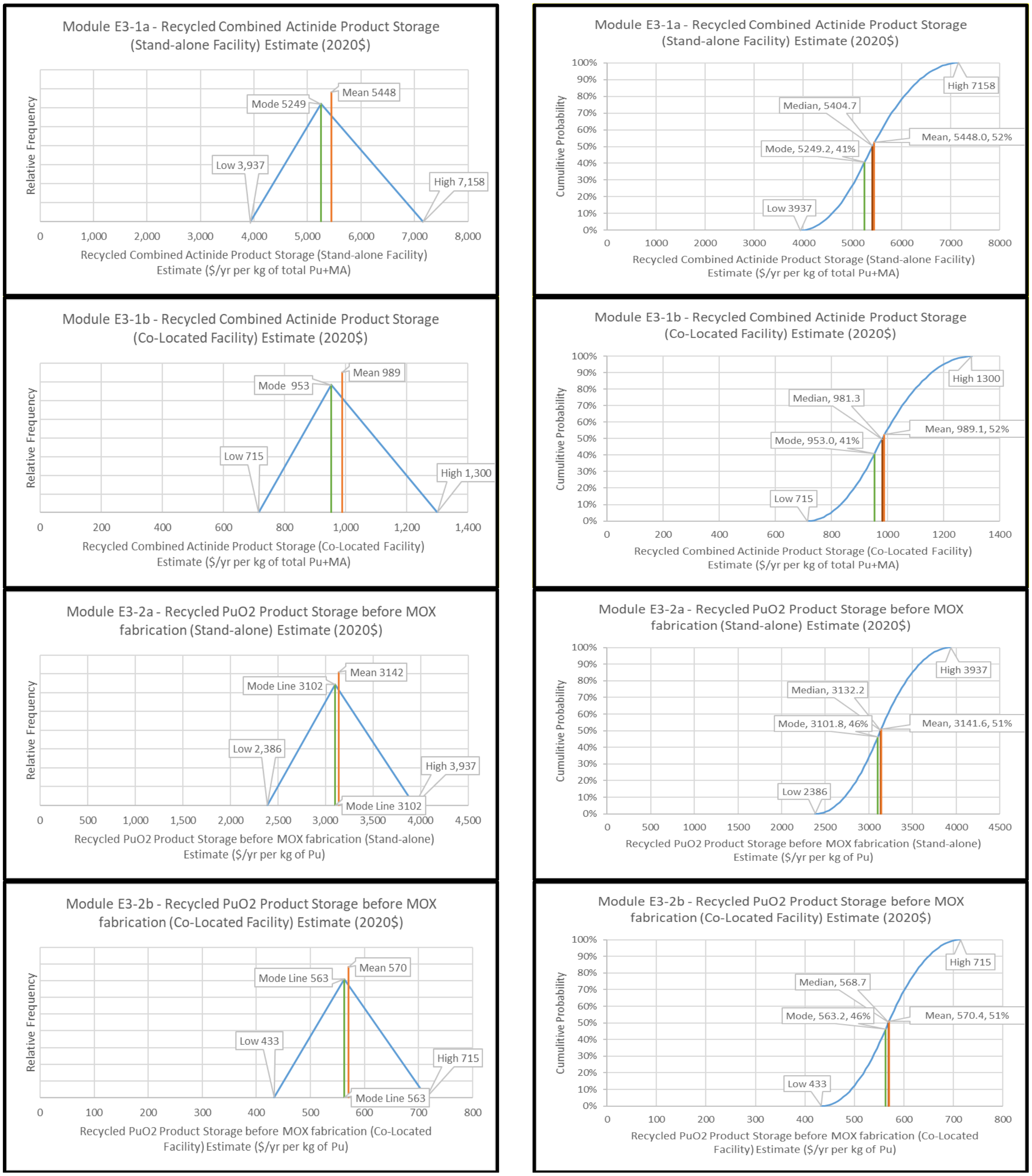

Figure E3-9. Module E3-1 recycled combined actinide storage (Pu-U) cost frequency distributions.

Figure E3-9 gives the estimated cost frequency distribution for the latest generation PUREX product from, for example, reprocessing at Rokkasho, Japan (i.e., mixed plutonium) and uranium oxide material.

E3-9. SENSITIVITY AND UNCERTAINTY ANALYSIS

None performed to date.

E3-10. REFERENCES

Boore W. B., Westinghouse Savannah River Company, Fax to R. Adams, INEEL, August 18, 2004, “Addresses estimated capitalized costs for a proposed Actinide Packaging and Storage Facility (APSF), and an unpublished Savannah River Site study for the consolidated storage facility (CSF) with surveillance capability, a 10,000-canister storage facility, design and rough order of magnitude costs completed in 2001.” (This document has a restricted distribution, may be proprietary, or both; and is not publicly releasable.)

Cabinet Office, “The Road to 2010 – Addressing the Nuclear Question in the Twenty First Century,” Command 7675, p. 28, The Stationery Office, U.K., July 2009.

DOE-STD-3013-2004, DOE Standard, “Stabilization, Packaging, and Storage of Plutonium-Bearing Materials,” April 2004.

Forsberg C. W., “What is Plutonium Stabilization, and What is Safe Storage of Plutonium?,” Oak Ridge National Laboratory, Report Ornl/M-4322, June 1995.

Hansard, (Parliamentary Record: “Sellafield: Security,” Column 1176W, March 31, 2009), http://www.publications.parliament.uk/pa/cm200809/cmhansrd/cm090331/text/90331w0033.htm, Web page accessed July 2009.

Hudlow S.L., 2004, “Welding Plutonium Storage Containers,” Westinghouse Savannah River Company, Report, http://www.osti.gov/bridge/product.biblio.jsp?query_id=1&page=0&osti_id=823032, Web page accessed July 2009.

NTI (Nuclear Threat Initiative), “Russia: Mayak Fissile Material Facilities, Fissile Material Storage Facility,” 2009, http://ww.nti.org/db/nisprofs/russia/fissmat/pumayak/fismatst.htm#storfac, Web page accessed July 2009.

Jones, Robert, “U/TRU Oxide Packaging, Storage and Transportation Alternatives,” UTR-G-EAS-G-00002, Rev. B. February 2007.

OECD-NEA, 1994, “The Economics of the Nuclear Fuel Cycle,” Section 4.3.2.6, Plutonium Storage, 1994.

ReFalo L. A. and Wong J. W., 1998, “Automation in the Actinide Packaging and Storage Facility,” Westinghouse Savannah River Company, Report WSRC-MS-98-00558, http://sti.srs.gov/fulltext/ms9800558/ms9800558.html, Web page accessed July 2009.

Richardson W., Letter to J. T. Conway, DNFSB, July 2, 1999, “Effect on Material Stabilization Activities of Delays in Construction of the Actinide Packaging and Storage Facility at SRS,” https://www.hss.doe.gov/deprep/1999/tb99102a.pdf, Web page accessed July 2009.

U.S. Army Corps of Engineers, 2007, Civil Works Construction Cost Index System (CWCCIS), Using CWBS Feature Code – 07 Power Plant, Appendix A, EM 1110-2-1304, Appendix Revised March 31, 2009.

E3-11. REFERENCES

Bae, Eugene, “The Design and Evaluation of an International Plutonium Storage System,” PhD Thesis, Dept. of Nuclear Engineering, M.I.T., 2001, http://hdl.handle.net/1721.1/8683, Web page accessed July 2009.

Bunn, M. et al., “The Economics of Reprocessing vs. Direct Disposal of Spent Nuclear Fuel,” Cambridge, Massachusetts, Project on Managing the Atom, Harvard University, DE-FG26-99FT4028, December 2003, pp. 19, 55–57, http://http://bcsia.ksg.harvard.edu/.

Clarke, John, International Nuclear Services, “Reprocessing and Recycle in Support of Sustainable Development and UK Experience with Thorp,” GNR2, June 12–14, 2007, http://www.gnr2.org/html/presentations.html, Web page accessed July 2009.

DOE/EIS-0229, “Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement,” November 1996.

DOE/EIS 0236, “Final Programmatic Environmental Impact Statement for Stockpile Stewardship and Management,” September 1996.

DOE/EIS-0240, “Disposition of Surplus Highly Enriched Uranium,” June 1996.

DOE/EIS-0283, “Supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement,” November 1999.

Forwood M., “The Legacy of Reprocessing in the United Kingdom,” Research Report No.5, International Panel on Fissile Material, www.fissilematerials.org, July 2009.

IAEA, “Safe Handling and Storage of Plutonium,” Safety Reports Series No. 9, STI/PUB/1061, September 1998.

Michaels, G. E., and T. D. Welch, Evaluation of Disposition Options for Reprocessed Uranium, ORNL/TM-12326, February 1993, pp. 27–47, 89–109.

Nuclear Energy Agency, *The Economics of the Nuclear Fuel Cycle*, 1994, pp. 11, 27, 37–38, 50, 77–80, http://www.nea.fr/html/ndd/reports/efc/, Web page accessed January 24, 2006.

Rothman A. B. and Liu Y. Y., “Review of DOE Criteria for Safe Storage of Pu Metals and Oxides,” Argonne National Laboratory, Report ANL/ET/CP-95417, April 1998.

Savannah River Site, 2007, “Nuclear Materials Disposition, Pu Storage Documents,” PBS 0011C, Department of Energy, http://professionals.pr.doe.gov/srs/smopustorage.html, Web page accessed July 2009.

Savannah River Site, “Technical Safety Requirements,” K Material Storage Facility, WSRC-TS-96-20, Rev. 18, November 21, 2004. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

U.S. General Accounting Office, *Weapons of Mass Destruction: Effort to Reduce Russian Arsenals May Cost More, Achieve Less Than Planned*, GAO/NSIAD-99-76, April 1999.

Washington Savannah River Company, “Engineering Alternative Studies for Separations Summary Report,” EAS-G-ESR-G-00049, June 2007. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

Washington Savannah River Company, “Follow-on Engineering Alternative Studies Summary Report,” GNEP-CFTC-PMO-MI-DV-2008-00087, March 2008. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)Westinghouse Savannah River Company, "K-Area Material Storage Facility, Documented Safety Analysis," WSRC-SA-2002-00005, Rev. 1, June 2004. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

Westinghouse Savannah River Company, "Safety Analysis - 200 Area Savannah River Site, Bldg. 235-F," WSRC-RP-89-575, Rev. 3, January 2003. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

Westinghouse Savannah River Company, "Technical Safety Requirements, Savannah River Site, Bldg. 235-F," WSRC-TS-97-3, Rev. 7, November 4, 2004. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

Westinghouse Savannah River Company, "Limited Extent Surveillance (Addendum to the 235-F Safety Analysis Report)," WSRC-RP-89-575, Rev. 0, April 2004. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

Westinghouse Savannah River Company, "Justification for Continued Operation, Savannah River Site, Upgraded Interim Control Posture for Bldg. 235-F," WSRC-RP-2004-00432, Rev. 0, June 2004. (This document has a restricted distribution, may be proprietary, or both: and is not publicly releasable.)

Wilkey D., Wood W., and Guenther C., "Long-Term Plutonium Storage: Design Concepts," 35th Annual Meeting of INMM, Naples, FL, LANL Report LA-UR-94-2390, July 1994.

Module E4

Managed Decay Storage

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E4 REVISION LOG

Rev.	Date	Affected Pages	Revision Description
	2009	All	Version of AFC-CBR in which Module first appeared: 2009 as Module E-4.
	2009	All	Latest version of module in which new technical data was used to establish unit cost ranges: 2009
			New technical/cost data which has recently become available and will benefit next revision: None identified
	2021	All	Re-formatted module consistent with revised approach to release of the AFC-CBR and escalated cost estimates from year of technical basis to escalated year 2020. Cost estimates are in US dollars ($) of year 2020.

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Module E4
Managed Decay Storage

E4-MD. SHORT DESCRIPTION OF METHODOLOGY USED FOR ESTABLISHMENT OF MOST RECENT COST BASIS AND UNDERLYING RATIONALE

- **Constant $ base year 2020 for this FY21 update.**
- **Nature of this FY21 Module update from previous AFC-CBRs:** Escalation only from last time values underwent technical assessment (2009 AFC-CBR)
- **Estimating Methodology for latest (2009 AFC-CBR) technical update from which this FY21 update was escalated:** Bottom-up pre-conceptual estimate for large aqueous reprocessing plant based on UREX technology where select fission products are separated and temporarily stored for decay storage. Some data from reprocessing plants which store vitrified waste was also available.

E4-1. BASIC INFORMATION

Module E4 provides the cost for decayed storage of immobilized, heat generating, mixed cesium-strontium-barium-rubidium (CsSr-BaRb) waste arising from advanced fuel cycles. The removal of these short and medium-term heat emitters from used fuel enables a substantial (1–2 orders of magnitude) increase in repository loading compared to direct disposal of spent nuclear fuel (SNF) (Wigeland et al. 2006). The period of decay storage has a nominal value of 300 years and may allow subsequent shallow disposal as low-level waste (LLW). After 100 years of storage, 3 half-lives for CsSr, the decay heat reduces to several hundred watts per canister and actinide decay power starts to predominate.

Advanced fuel processing flowsheets (e.g., UREX+) may separate a near pure aqueous CsSr-Rb-Ba nitrate or carboxylate (salt-free) stream and use electroreduction /refining using molten chloride salts to extract CsSr halides using zeolite ion exchange/occlusion. The often lower selectivity of molten salt processing may decrease CsSr loadings, decrease specific decay power, and increase waste quantities as well as decay periods for compliance with Class C waste criteria, and affect subsequent waste management. CsSr heat generating streams may be immobilized using a wide variety of waste forms and processes including vitrification and ceramic sintering. Depending on design, many repositories have good capacities for vitrified high-level waste (HLW) at this stage though further decay, increasingly effective removal of actinides and CsSr and/or disposal in heat dissipating salt repositories may significantly increase repository capacity. However, this study is restricted to performing sensitivity analysis for technology and cost of decay storage of immobilized CsSr waste form arising from advanced fuel cycles. Unit costs of CsSr waste storage are estimated based on storage technology, waste form properties, and canister size.

This module includes an evaluation of immobilization processes and corresponding waste form properties for storage of the separated CsSr fission product waste fraction derived from advanced fuel separation processes for treatment of light-water reactor (LWR) SNF. Waste form properties determine acceptable canister dimensions, features, numbers, and store design, which enable prediction of capital and operational costs. Various storage designs (e.g., ponds, standalone casks, housing arrays, dry vault storage [passively or forced convection cooled], and modification of existing facilities) to provide dry storage are available for storage of solid heat generating wastes such as spent fuel and vitrified HLW. Cost data from existing heat-generating waste storage studies is used to determine unit costs for 300-year CsSr storage. Three centuries provide 10 half-lives of decay and is the nominal value for the CsSr waste to comply with Class C waste criteria for proposed shallow disposal as low-level waste (LLW). Data is taken from a variety of sources including U.S., UK, French, and Japanese vitrified HLW, SNF, and Advanced Fuel Cycle Initiative (AFCI) studies. These provide various costs and waste forms (e.g., aluminosilicate, bentonite, borosilicate glass, synroc, and SNF) for a variety of specific decay powers (W/L) and store designs, so that it is important to reconcile scope and consistency of data.

E4-2. FUNCTIONAL AND OPERATIONAL DESCRIPTION

E4-2.1 Waste Forms, Canister Sizes, and Storage of Heat Generating Wastes

Solidification of HLW, the main existing heat generating waste, serves the two main purposes of immobilization of waste for storage, transport, and emplacement in the repository and long-term fixation of radionuclides after repository closure and permanent disposal. Interim storage is normally required to allow further decay of the major heat emitting nuclides, and therefore reduce the early thermal loading of the repository. Thermal, mechanical and chemical stability of the waste form is required including effects of irradiation and leaching. For separated CsSr wastes, the main objective is that of immobilization while a secondary one is longer-term stability (e.g., to prevent the leaching of long-lived isotopes such as Cs-135 and stable toxic species such as Ba [Alvarez 2007]).

Various waste forms and corresponding waste immobilization processes are known for HLW. Calcines are products or intermediates obtained by partial or complete removal of volatile components of the waste, such as water and nitrates, at temperatures of 400–900°C. This creates a mixture of oxides in particulate form and with a specific surface of 0.1–5 m^2/g. Calcine powders may not be very stable because of the chemical properties of some constituents, large surface area, low thermal conductivity, and friable nature of the solids creating fine dust. Depending on calcination temperature, calcines may have residual water and nitrate content. Calcine powder may be pressed or solidified within cements or concentrated solutions grouted. If waste temperatures, radiation, or canister corrosion effects are sufficient to release water, NOx or hydrogen, then canister venting, inspection, and off-gas treatment may be needed (Streatfield et al. 2006).

For passive long-term decay storage, higher process temperature, refractory, near-inert waste forms, such as glasses and ceramics are preferred. For heat generating wastes, waste form dimensions may need optimization to limit center line temperatures to acceptable values. The more important immobilization alternatives for HLW are calcine, ceramics, glass, glass ceramics, and cement (Benedict et al. 1981), see Table E4- 1.

Table E4- 1. Immobilization waste form options.

Alternative	Calcine	Glass	Ceramic
Basic	Fluidized bed (particulates) Pot (cake)	Borosilicate (cylinder) Phosphate (cylinder)	Aluminosilicate Bentonite
Advanced	Supercalcine (additives, high T)	Borosilicate glass ceramic (cylinder)	Synroc (multi-phase ceramic) – Hi T, HIP
Composite	Multibarrier (e.g., pyroC, SiC in metal matrix)	Vitromet (glass/ glass ceramic in metal matrix)	Glass ceramic (e.g., puck crystallized glasses and sodalite in glass matrix)
Cement-vented Canister	Low T encapsulation of concentrated solution or calcined particles.		Higher temperature specialized cements (e.g., supercalcines).

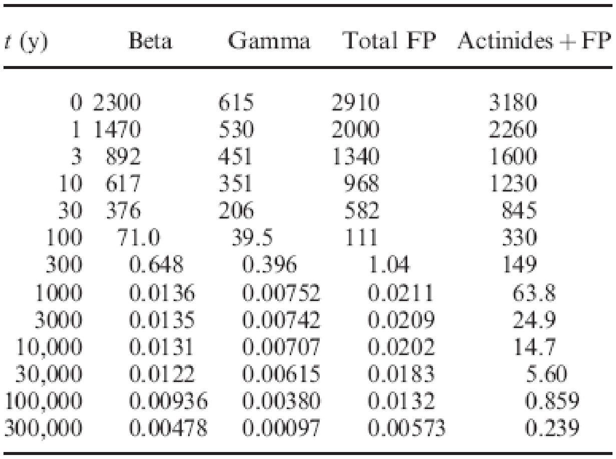
There is generally an increased processing cost for refractory, inert waste forms. This may take place through need for high-temperature operation, corrosive conditions limiting equipment lifetimes, volatilization of selected fission products requiring complex off-gas systems and waste recycle. Synrocs often require small batch operations using hot isostatic pressing (HIPing) at high temperature and with relatively long process cycles.

Waste forms may incorporate differing waste concentrations to meet waste performance and economic goals. Waste packaging and transportation costs are significant so that reduction of package number is desirable. However, thermal limitations apply to transportation, disposal and the waste form itself, and excessive fission product (FP) concentrations reduce the chemical performance of waste and may cause excessive internal temperatures. Composition limitations are typical for glasses where either phase separation or lack of glass forming occurs. For borosilicate glasses, for example, the FP oxide limit is normally considered to be around 20–25% by weight (plus process additives) using existing hot-wall melters. Higher concentration may create a distinct yellow crystalline phase formed of alkaline and alkaline earth molybdates. This readily soluble phase contains Cs-137 and Sr-90. Glasses can be formulated to incorporate most fission product and actinide oxides with good stability. Devitrification occurs above the glass transition temperature, for example at elevated temperatures of ~500°C for phosphate and ~600°C for present borosilicate glasses. Some processes employ controlled crystallization to glass ceramics to create known waste form properties. Other waste form composition/temperature limitations may arise from a wide variety of limits such as melting, volatization, and recrystallization into new phases, chemical reaction, canister pressurization, etc. Composition is not a direct limitation for calcines, which are amorphous, but high-heat load may cause further chemical decomposition, canister corrosion, and pressurization. Grouts are normally used to immobilize mineral ion exchange (IX) materials used for treatment of LLW and (in Europe) for intermediate level wastes (ILW) liquid wastes, but have also been used for encapsulation of low specific decay power raffinates from specialized recycling operations.

Industrial practice for HLW vitrification tends to use large canisters for low specific decay heat defense wastes (e.g., 0.61 m o.d., 3 m height and 2.1 t filled mass) for Savannah River Defense Waste Processing Facility (DWPF), and small canisters for high-specific decay power LWR wastes (e.g., 0.43 m o.d., 1.3 m height and 0.5 t filled mass) for Ateliers Vitrification La Hague (AVH) (IAEA 1992). Even smaller diameter canisters, o.d. 0.3 m, have also been used at PAMELA (Germany) and WIP (India). In general terms, canisters with diameters less than 0.2–0.15 m diameter are not favored industrially due to difficulties of filling with molten glass due to bridging, potentially more thimble tubes due to retention of moderate l/d ratios for canister cooling, increased pressure drops with high-air velocities, and possible limited cost reduction of storage with decreasing thimble tube diameter. KfK Germany developed a process where HLW phosphate glass beads were cast and then embedded in molten metal in a canister. Such an approach or other internal features for heat conduction may be especially useful for CsSr vitrification of short-cooled, high-loaded fission products.

For LWR fuel of typical burn-up of 40 GW(t).d/t(iHM), the decay powers are given in the Table E4-2, (Bergelson et al. 2005).

Table E4- 2. Decay heat power of FPs and transuranics (Pu, Am, Cm) during long-term storage, W/t(iHM).



t (y)	Beta	Gamma	Total FP	Actinides + FP
0	2300	615	2910	3180
1	1470	530	2000	2260
3	892	451	1340	1600
10	617	351	968	1230
30	376	206	582	845
100	71.0	39.5	111	330
300	0.648	0.396	1.04	149
1000	0.0136	0.00752	0.0211	63.8
3000	0.0135	0.00742	0.0209	24.9
10,000	0.0131	0.00707	0.0202	14.7
30,000	0.0122	0.00615	0.0183	5.60
100,000	0.00936	0.00380	0.0132	0.859
300,000	0.00478	0.00097	0.00573	0.239

Data in Table E4- 2 shows both total FP and total FP with transuranic (Pu, Am, Cm) contributions to decay power. Often the Pu contribution is omitted as vitrified HLW includes minor actinides, but not Pu. The total FP decay power reduces by 40% in the period 10-year cooled to 30-year cooled, which indicates the major contributions of Cs-137 ($t_{½}$ = 30y) and Sr-90 ($t_{½}$ = 28y) to decay power in this period.

Heat generation in immobilized HLW and CsSr waste causes the waste form to be at elevated temperature for more than 100 years. With some simplifications, the maximum temperature difference between the centerline and surface of a long cylindrical waste form is given by:

$$\Delta T_{max} = q\, r^2 / 4\, \kappa \qquad (1)$$

Where q is the power density, W/m^3, r is the radius of the cylinder, and κ is the waste form thermal conductivity, W/(m°C). The surface temperature is given by the storage conditions including canister wall and waste surface/canister interfacial properties. This enables scaling of canister radius against heat loadings from existing commercial practice (IAEA 1992). Representative values for conductivity of waste forms are given in Table E4-3 (Benedict et al. 1981).

Table E4- 3. Thermal conductivity ranges for various HLW forms in temperature range 100–500°C.

Waste Form	Thermal Conductivity, κ W/(m°C)
Particulate calcine	0.2–0.3
Phosphate glass	0.8–1.2
Borosilicate glass	0.9–1.3
Borosilicate glass ceramic	1.5–2.0
Particulate calcine or glass beads in metal matrix (e.g., vitromet)	~10

Waste form conductivity clearly has a major influence on centerline (peak) temperature and corresponding canister dimension (radius), see Figure E4-1, (Benedict et al. 1981).

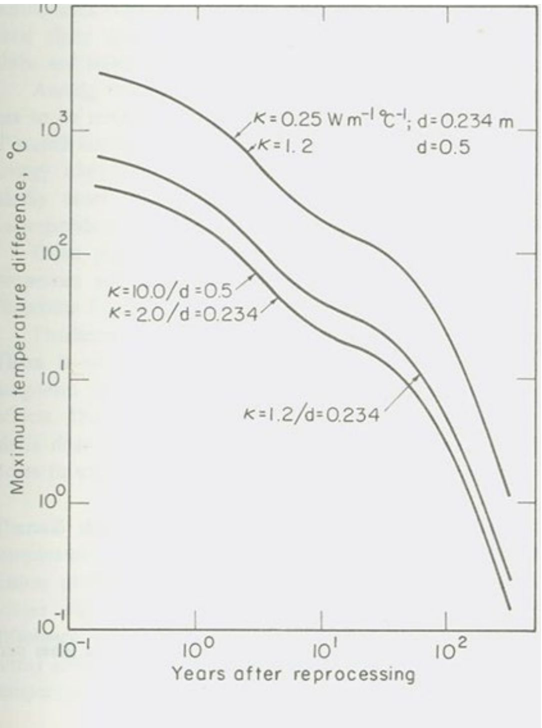


Figure E4-1. Maximum centerline temperature difference of waste form as a function of decay time.

Figure E4-1 calculated maximum temperature difference in a cylinder of solidified waste for different diameters and thermal conductivities as a function of time (years) after recycling. The heat generation rate is based on fission products and minor actinides incorporated into a waste form specific volume of 70 L/t(iHM). The originating SNF burnup is 30 GW(t).d/t(iHM) and recycling taking place at 150 days SNF decay.

At the assumed waste loading, Figure E4-1 indicates maximum temperature differences for glass ceramic waste with canister diameter of about 0.5 m of between the waste center line and surface of >1,000°C (≡ 100 W/L) and >100°C (≡ 10 W/L) for 1 year and 10 years decay after recycling, respectively.

In the present study, borosilicate glass is considered the reference HLW form. This is conservative since it is somewhat more restrictive thermally than some synrocs and other ceramics. For civil design, it is generally preferable to restrict natural convection cooling air discharge temperatures to around 150–200°C as concrete structural components are damaged by long-term contact with air at temperatures approaching 100°C. However, higher values can be engineered. Air cooling in forced convection stores would be less limiting, but for a long-term decay store, there is likely to be conservatism concerning highly rated systems and effects of cooling failure. This suggests that a maximum temperature difference, ΔT, between waste centerline and surface of around 300°C may be appropriate assuming a centerline maximum design temperature of ~500°C. For a canister with diameter 0.5 m containing borosilicate glass HLW, a maximum specific thermal power in the range 10–30 W/L appears suitable. Raising the glass transition temperature by ~200°C increases the maximum thermal power by about 60%.

E4-2.2 Potential Waste Forms for Immobilization of Short-Lived Fission Products

Of the fission products, the most troublesome Cs and Sr isotopes are Cs-137 ($t_{1/2}$=30.07 y: 0.66 MeV γ and 0.514 MeVmax β-) and Sr-90 ($t_{1/2}$=28.78 y: 0.546 MeVmax β-), so their activities remain a concern for ~300 years (i.e., ~10 half-lives). These two isotopes generate a major portion of the decay heat in spent nuclear fuel over the first 100 years of storage, but then are essentially stable. Removing Cs and Sr for decay storage will reduce the short-term heat load on a repository waste form.

Fission product oxide mass, excluding noble gases, is ~1 kg(FPOx)/GW(t).d, and so for a metric ton, t, of SNF at 40 GW(t).d/t(iHM), the mass of FP oxides is about 40 kg. Cs and Sr form about 10 atom % of the FPs, of which around half are the major heat emitting isotopes—Cs-137 and Sr-90. Total CsSr also form around 10% by weight of the FP oxides, that is 4 kg/t(iHM) of spent fuel. Total CsSr-Rb-Ba form around 15% by weight of the FP oxides, that is 6 kg/t(iHM) of spent fuel. From Table E4-2, where Cs-137 and Sr-90 are the only major FP isotopes with half lives between 10–50 years, the decay power of CsSr is seen as about 1 kWt(iHM) at 10 years ex-reactor. Alternatively the CsSr decay power can be expressed as ~¼ kW/kg(CsSr) at 10-years cooling. In engineering terms, the specific decay power of CsSrOx is about 10 times that of overall FPOx at 3–20 years cooling.

Recently, interest in separation of Cs and/or Sr during remediation of long-stored HLWs and for advanced fuel cycles has stimulated developments in waste forms tailored to CsSr immobilization. These include variants of waste forms for HLW and several new matrices (see Table E4-4), which is representative rather than complete. The various minerals formed have differing capacities for Cs and Sr.

Table E4-4. Representative CsSr-Rb-Ba waste forms.

Matrix	CsSr Composition	Composition Matrix, (% wt)	Process	Reference
Cement	4% Zeolite A, 5mEq/g	PFA, Ordinary Portland Cement	Grouting ambient T Maintain ≤95°C	El-Kamash et al. 2006
Alumino-silicate (Steam reform)	27%Cs / 8%Sr $SrCO_3$ $CsAlSi_2O_4$	Pollucite/hydroceramic Slawsonite	Steam reform CsSr-Ba with carbon & alumino-silicate clay at ~700°C	Ortega and McDeavitt, 2007 Law et al. 2007
Bentonite (alumino-silicate) Dry sinter	≤40% Cs loading	Celsian Pollucite Hydroxyl-apophyllite	Dry sintering bentonite clay containing Cs, Sr, Rb, Ba to 600–1,000°C	Kaminski and Merz, ANL.
Crystalline Silicotitanate, CST and niobate IX	Cs_2O ~ 20% wt	$Cs_2TiSi_6O_{15}$ •$3H_2O$ $Cs_3TiSi_3O_{95}$ •$3H_2O$ and Ti analogue of Pollucite $CsTiSi_2O_6$•$3H_2O$	Calcining CsSr soaked UOP CST IE-911 in air at 900–1,000°C	Balmer et al. 2000 Luca et al. 2006
Borosilicate Glass High mp glass	Cs_2O 13% wt and SrO 7% wt PNNL ~40%wt	Na_2O 10-20 B_2O_3 10-17 SiO_2 45-50 Al_2O_3 2-5 Ba,Pb,TiOx 4-6	Calcination and Melting High-melting glass	Aloy et al. 2007
Hexagonal Tungsten oxide Bronze (HTB)	Cs_2O ~12% wt or SrO ~5% wt	$Cs_{0.13}Mo_{0.03}W_{0.97}O_3$ $Sr_{0.05}Mo_{0.03}W_{0.97}O_3$	CsSr adsorbed hexagonal tungsten oxide bronze, Calcine 500–1,000°C in air.	Luca et al. 2006

Table E4-4. (continued).

Matrix	CsSr Composition	Composition Matrix, (% wt)	Process	Reference
Synroc-C Hydrous titanium oxide (HTO)	CsSr-Rb-Ba 12 %wt	Hollandite Rutile Titanates	Calcination 750°C & HIP 1,275°C, 30MPa, 1h	Carter et al. 2007
Cs/Sr Oxides	Cs_2O/SrO	Pure	Calcination	-
CsCl	100% Cs salt in capsule in pool	CsCl, 35kCi 190W	IX separation and Evap	Nat Acad Sci, 2003
SrF_2	Sr salt in capsule in pool	CsF_2, 33kCi 260W	IX separation and Evap	Nat Acad Sci, 2003

These potential CsSr waste storage forms evolved from upstream processing needs. Bentonite (including commercial UOP IE-911), and hydrous titanium oxide (HTO) are examples of IX materials used to selectively adsorb Cs, Sr, etc., from stored, complex chemistry salt HLWs to provide partitioning of waste for optimized waste management. These IX materials bearing low-medium Cs, Sr concentrations are heat treated by sintering, generally in the temperature range 500–1,000°C. This causes removal of water, recrystallization, denitration and additional phases, and ultimately removal of hydroxyl groups. By contrast, the advanced fuel cycle processes (e.g., UREX+ [Vandegrift et al. 2004]) create salt-free product streams of CsSr (e.g., nitrate and carboxylic acid based). These are not constrained by feed of mineral IX materials and can be used to form the complete range of waste forms from pure CsSr oxides/chlorides to glasses to freely tailored ceramics. The uses of zeolites (i.e., micro-porous crystalline solids with well-defined structures) generally contain silicon, aluminum, and oxygen in their framework and cations, and/or other molecules within their pores. For CsSr recovery in molten salt, electrochemical waste forms are more likely to be aluminosilicate ceramics or aluminosilicate glass ceramics.

High-level waste vitrification is well known as a complex technology with significant cost impact on existing PUREX commercial and defense recycling waste management. Advanced fuel processes generally partition FP and actinide species into more streams (e.g., seven for UREX+4). Some of these, (CsSr), have medium radioactive lifetimes and means have been sought to optimize the waste immobilization process to the waste lifetime, including storage requirements. For substantial masses, CsSr wastes need cooling for periods of 100–200 years. AFCI has examined use of a steam reforming process to fabricate alumino-silicate waste forms for CsSr storage (Law et al. 2006).

The UREX+ suite of processes has a separation segment, CCD-PEG in UREX+1a (Law et al. 2004) or FPEX in UREX+1b (Law et al. 2007), for recovery of CsSr-Rb-Ba from the raffinate of the UREX segment. Both of these technologies provide simultaneous solvent extraction of Cs and Sr together with the majority of Rb and Ba. With CCD-PEG, the CsSr by-product is stripped using an organic amine and carboxylic acid mixture whilst FPEX uses dilute nitric acid as strip. Steam reforming has been developed for stabilization of streams because it can produce a solid waste form while retaining the Cs and Sr in the solid, destroy the nitrates and organics present in these aqueous solutions, and convert the Cs and Sr into leach resistant aluminosilicate minerals. The waste form is intended to meet a 300 year, 10 half-life periods of storage prior to projected LLW disposal complying with Class C waste criteria.

A bench-scale steam reforming pilot plant has been operated at Idaho National Laboratory (INL) with several potential CsSr feed compositions and steam reformed product has been generated and analyzed (Law et al. 2006). A small, but representative fluidized-bed was used to conduct steam-reforming tests to produce mineralized granular product. Operating conditions of 700°C, ~3% H_2, ~4% CO, 70% CO_2, and

20% steam were used to decompose nitrates and organics. A starting bed of 100–300 micron aluminum oxide particles was used and Sagger clay slurried with the feed to produce pollucite and other alumino-silicate minerals. Excess clay was used to mineralize the cationic feed constituents. The clay particles are less than 10 µm to achieve a high-surface area for reaction. The final bed material in each run was generally a granular material much like the initial aluminum oxide starting bed with some additional smaller diameter solids. The bench-scale steam reformer tests successfully converted cesium/strontium strip products to a solid form without volatilizing the Cs. Results also indicate that with optimization of the steam reforming operating parameters, 100% mineralization is possible (Law et al. 2006). The bed waste product material may be compacted, for example, within canisters to form pucks, which may be loaded into an over-pack.

A collection of scoping studies, entitled Engineering Alternative Studies (EAS), related to a commercial scale UREX+ separations plant were commissioned by Department of Energy (DOE) and carried out by a multi-national laboratory team in 2006–2008. EAS investigated features of a canyon approach for a commercial plant, with expected throughput of ~3,000 t(HM)/year, with three solvent extraction lines. The Follow-on EAS (FOEAS) evaluation assumed a smaller plant throughput (~800 t(HM)/year UREX+) with re-examination of facility layout options, requirements, alternate flowsheets, etc. (Hebditch et al. 2007).

In the EAS, an engineering proposal and costing of the proposed storage of the UREX+ cesium-strontium (CsSr) waste stream was presented. The study was based on the UREX+1a process, throughput of 3,000 t(iHM)/year mixed LWR fuel of 60 GW(t).d/t and formed a variety of products and wastes, including an aluminosilicate mineral powder CsSr waste intended for a 300-year period of storage prior to projected LLW disposal. This study demonstrated reasonable feasibility, but was not an economic optimization and further studies were performed.

The FOEAS was based on the UREX+1b process, throughput of 800 t(iHM)/year mixed LWR fuel of 60 GW(t).d/t and formed a variety of products and wastes. For CsSr, three waste forms were examined conceptually (i.e., a sintered bentonite and two vitrified CsSr options with differing CsSr loadings). As for EAS, these would need nominal 300-years storage for compliance with Class C waste disposal. Other geological disposal scenarios may be feasible, but are not well defined yet and so are not considered here. This was a top-down assessment based on the above 3,000 t(HM)/year study with some variations to account for process changes and scale, etc. The use of sintered bentonite or vitrified CsSr wastes may possibly increase waste immobilization costs, but is expected to decrease overall waste storage costs life cycle costs (LCC) by increasing CsSr loading and canister diameter and by reducing waste volumes, total canisters, and required storage capacity. However, depending on design these may need periods of forced convection cooling and delayed potential for using passively cooled storage. If AFCI does call for CsSr separation, a vitrified CsSr waste form option is presently favored with a range of increased loading values being examined and this makes good use of state-of-the-art vitrification and waste storage technologies.

There was a third conceptual design, which was based on a high temperature, molten salt electrochemical process. This design was based on oxide fuel electrochemical reduction, uranium electrorefining, and transuranic product recovery by electrowinning. The design throughput is 300 t(HM)/year mixed LWR fuel of 60 GW(t).d/t and the process formed a variety of products and wastes including a glass-ceramic CsSr waste formed with zeolite used as an ion exchange material to recover CsSr from salt. The specific activity of this waste form is expected to be similar to or lower than the aluminosilicate waste and so costs for this case can be considered equivalent to EAS costs using a per kg(CsSr) basis. However, the electrometallurgical CsSr waste may have higher radioactive impurity levels (e.g., TRU), and require longer decay storage to comply with Class C limits or need disposal as GTCC.

In summary, the CsSr-Rb-Ba separated waste stream from UREX+ aqueous separations is salt free and can be decomposed thermally to the oxides and converted into a wide variety of waste forms and chemistries including particulate ceramics, cements, sintered ceramics, glass ceramic composites, and cast vitrified waste. The waste stream has few process additives so the CsSr waste form may be made as concentrated in CsSr as desired consistent with chemical, physical, and thermal waste-form properties. The Integrated Waste Management Strategy (IWMS) presently favors the CsSr vitrified waste option using existing waste storage and state-of-the-art vitrification technologies. Incorporation of 20% wt CsSrOx in borosilicate glass has been reported and fully active samples made (Aloy 2007). Pacific Northwest National Laboratory is understood to be investigating higher incorporations, ≥40% wt. CsSr-Rb-BaOx, which equates to ≥27% wt CsSrOx. CsSr concentrations in commercial LWR vitrified waste are around an order-of-magnitude lower than the latter number so that borosilicate glass (BSG) waste container diameters may be need to be reduced from 0.4 m to 0.13 m (i.e., by factor √10), or SNF decayed stored for >3 CsSr half-lives (i.e., ~100 years). Additionally, increasing the glass transition temperature by several hundred degrees centigrade may allow the canister diameter to be increased back to around 0.2 m with the same high CsSr incorporation. Glass formulations with higher devitrification temperatures generally require formulations with higher melting point as may be achieved by a cold crucible melter.

E4-2.3 Vitrification and Storage of LWR Oxide HLW

For LWR fuels, the main operating commercial separations plants in the world are UP2-800 and UP-3 at Cap La Hague and THORP at Sellafield (Rokkasho is believed to start full operation in 2019 and uses Japanese JCM vitrification technology). These French and UK plants have used French AVH vitrification technology for nearly 2 decades. Calcined fission product waste is mixed with glass frit in the ratio of around 1:3 by weight. The PUREX raffinate has low processing inerts and after calcination is mainly FP and minor actinide (MA) oxides with very low U, Pu content, and moderate corrosion product concentrations.

Vitrification of commercial and/or defense HLW has taken place at Cap La Hague and Marcoule in France, Sellafield in the UK, Tokai in Japan, Karlsruhe in Germany, Savannah River and West Valley in U.S., Tarapur in India, Russian Federation, etc. Almost all of these facilities use air-cooled vault storage systems where waste canisters are stored in cooled thimble tubes. Most use forced air convection, at least initially, whereas one uses natural convection with forced convection as standby during early operation (IAEA 1992).

France first performed vitrification operations in the 1970s in the Ateliers Vitrification Marcoule facility (known as the AVM) and then in the late 1980s in the R7 and T7 facilities of the La Hague plant (referred to as the AVH). French vitrification technology uses a rotary calciner feeding a metallic inductively heated melter vessel, which siphons batches of vitrified waste into HLW canisters. The Marcoule vitrified waste store used HLW canisters of dimensions, 0.5 m diameter and 1 m height, for lower burnup, lower decay power gas-cooled reactor wastes. For AVM, three casts of glass (120 kg each) totaling about 140 L are made into a single stainless steel canister. The vitrified waste store used thimble tubes (steel sleeves with base set into concrete) and stacked canisters, 10 high. The sleeves are 0.6 m diameter and 10 m height. The maximum output of the 10 canisters in a sleeve is 8 kW (i.e., 0.8 kW/canister) on average, but 1 kW peak value. Forced convection cooling air normally flows between sleeves and canisters at velocities of 10–15 m/s with filtration at outlet but can revert to natural convection without filtration for power failure conditions or after long storage, etc. Two vaults were built initially at Marcoule, one with 80 storage sleeves and the other with 60 storage sleeves. The maximum heat load of the whole store is 1 MW. One AVM single line plant was constructed in the 1970s at Marcoule, to provide vitrification of low burnup gas reactor fuel with vitrified waste power densities of ≤8 W/L.

The AVH stainless steel canisters are cylindrical with overall dimensions of around 0.42 m diameter (17 inches) and height 1.3 m (52 inches). The canisters have a top flange of reduced diameter with welded closure following filling with two pours from the melter. After pouring, the canisters contain about 400 kg (150 L) vitrified HLW and are around three-fourths filled. (In France, the residual space is filled with pucks of compacted leached fuel hulls.) Two AVH plants, designated R7 and T7 and each of three vitrification lines, were constructed in the late 1980s at La Hague, to provide vitrification of standard LWR fuel (33 GW(t).d/t) HLW after 4 years of cooling. Preliminary evaluations foresaw glass center line temperatures ≤650°C and power densities of ≤60 W/L, which implied a maximum canister heat load of 9 kW. Eventual design values were specified as 20 W/L and 3 kW, respectively.

The Sellafield waste vitrification plant was constructed with two AVH process lines, and first operated in the early 1990s. Its vitrified product store (VPS) accommodates up to 8,000 AVH canisters stacked 10 high (about 13 m). Each canister (400 kg waste) typically contains vitrified waste from the recycling of 8 t Magnox fuel or 2 t oxide fuel (Dobson and Phillips 2006). There are 800 stainless steel storage thimble tubes into which the canisters are stacked through top plugs and seals. Each storage tube is within a rectangular compartment to guide cooling air. Decay heat is removed by natural convection cooling of the exterior of the sealed storage tubes, and due to multiple barriers and compliance with glass centerline limits and civil structural limits, no filtration of the cooling air is required. The Sellafield borosilicate glass formulations have waste oxide incorporations in the range 20–30 wt% with glass transformation temperature of around 550°C; 500°C is taken as the glass center line temperature limit. VPS has capacity for vitrified HLW from 2 decades of THORP design throughput of 800 t(oxide SNF)/year, that is 16,000 t(oxide SNF) equivalent.

British Nuclear Fuels plc (BNFL 1991) commenced active commissioning of the third line at Sellafield's WVP in January 2002. The start of operation of the 320M UK pound (~2000 m.v.) ≡ U.S. $485M (2000 dollars) line enables BNFL to meet its commitment to speed up the conversion of liquid HLW to borosilicate glass blocks for longer term storage. The UK regulator requires year-on-year reductions in highly active liquid waste down to buffer stocks of 200 m^3 by 2015.

The operation of WVP has led to the production of over 4,000 containers of vitrified waste to-date, which are currently stored within the VPS at Sellafield. The VPS is deemed suitable for this interim storage requirement, subject to regular maintenance and refurbishment, for at least 100 years. A proportion of the vitrified HLW will be returned to overseas customers at the appropriate time as set out in the recycling contracts. The canisters of vitrified HLW are kept in a purpose-built store (VPS), which has passive cooling and a back-up forced cooling system.

The design and operation of HLW vitrification facilities has been well described for the major national nuclear programs (IAEA 1992). Following filling of stainless steel, cylindrical waste canisters with vitrified waste, various operations are used to prepare canisters for storage and ultimate disposal. Thermal conditioning of canisters to reduce heat shock and decrease glass cooling rate and fracture may be used. Tungsten Inert Gas (TIG), plasma torch, or upset-resistance welding is used to seal the canisters with lids. Canister welds are normally inspected optically or by helium leak testing. Canister dimensions, weight temperature and dose rate may be determined. The exteriors of canisters are often decontaminated using high-pressure water, sand slurries, dry blasting, or electrochemical decontamination. Waste canisters must be cooled in storage to minimize devitrification and maintain store integrity.

Currently, operating and planned interim stores use air cooling of canisters. Air cooling can be achieved by conduction, or natural or forced convection. For some high-specific decay power glasses, forced cooling is combined with natural convection cooling. Canister, waste, and store characteristics for various national facilities are given in Table E4-5.

Table E4-5. Canister and waste parameters for operating vitrified HLW stores.

Facility Cooling	Canister I.D./Height m/m	Glass Mass/ Volume, kg/L	Max Activity GBq α/β	Maximum Canister Power, W	Maximum Power W/kg
AVM France Forced/Natural	0.49 1.0	360 135	3.0×10^7 1.4×10^7	1,000	2.8
R7/T7 France Forced/Natural	0.42 1.34	400 150	1.4×10^5 2.8×10^7	2,980 peak 2,100 average	7.5
WVP–VPS UK Natural	0.42 1.34	400 150	-	*Estimated 2,000*	-
DWPF–U.S. Forced	0.59 3.0	670	-	-	-
TVF–Japan Forced	0.42 1.0	300 110	1.5×10^7 Combined	1,400	4.7

BNFL WVP with Lines 1 and 2 and VPS has dimensions 64 m long × 38 m wide × 40 m high, which gives footprint of 2,430m^2 (IAEA 1992). The capital cost is estimated as 250M Great Britain Pounds (GBP) (1990 dollars) ≡ \$446M (U.S. 1990 dollars[h]) ≡ 730M (U.S. 2008 dollars[i]). The two stores and access corridor have a footprint of around 25 m × 40 m = 1,000 m^2 (10^4 ft^2) or 40% of WVP footprint. A pro rata capital cost for the VPS is then \$292M (U.S. 2008 dollars) ≡ 146M GBP with a capacity corresponding to 16,000 t (LWR SNF). This corresponds to a facility square foot capital cost of \$29K. Commonly process areas have costs that are several times greater than waste storage areas. So, a value of \$150M (\$15K/ft^2) for the store may be appropriate here, and this is regarded as high although passive cooling favors lower long-term operational costs.

E4-3. PICTURES AND DIAGRAMS

Figure E4-2 shows a cross-section of vitrified waste storage, showing how decay heat is removed by natural convection cooling of the exterior of the sealed storage tubes.

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[h] Measuring Worth - Exchange Rates Between the United States Dollar and Forty-one Currencies, http://www.measuringworth.com/datasets/exchangeglobal/result.php

[i] Money values derived using: U.S. Army Corps of Engineers, Civil Works Construction Cost Index System (CWCCIS), Using CWBS Feature Code – 07 Power plant, Appendix A, EM 1110-2-1304, Appendix Revised September 30, 2007.

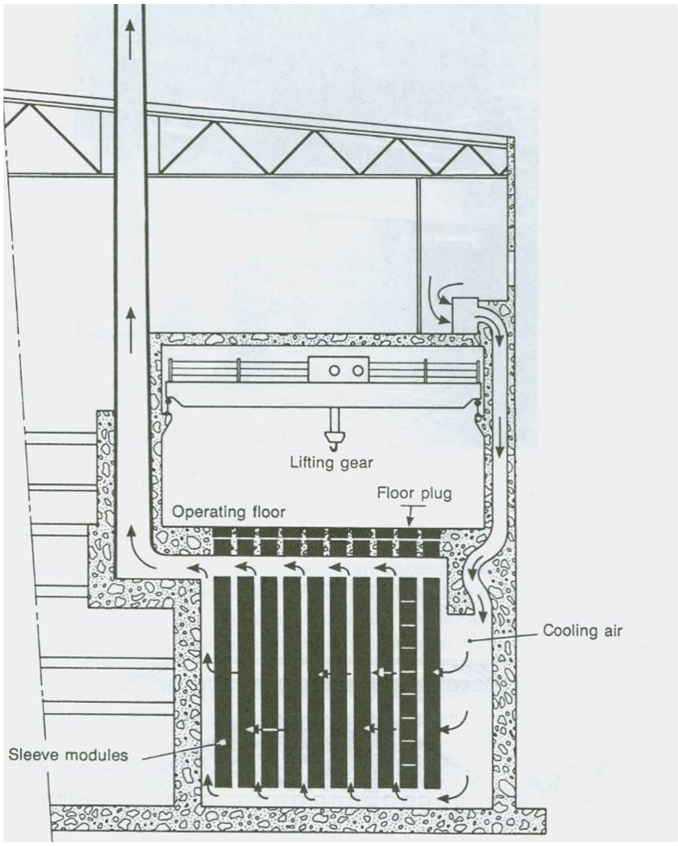


Figure E4-2. WVP VPS natural convection air-cooled storage system.

The canisters of vitrified HLW are kept in a purpose-built store (VPS) shown in Figure E4-3 has passive cooling and a back-up forced cooling system.

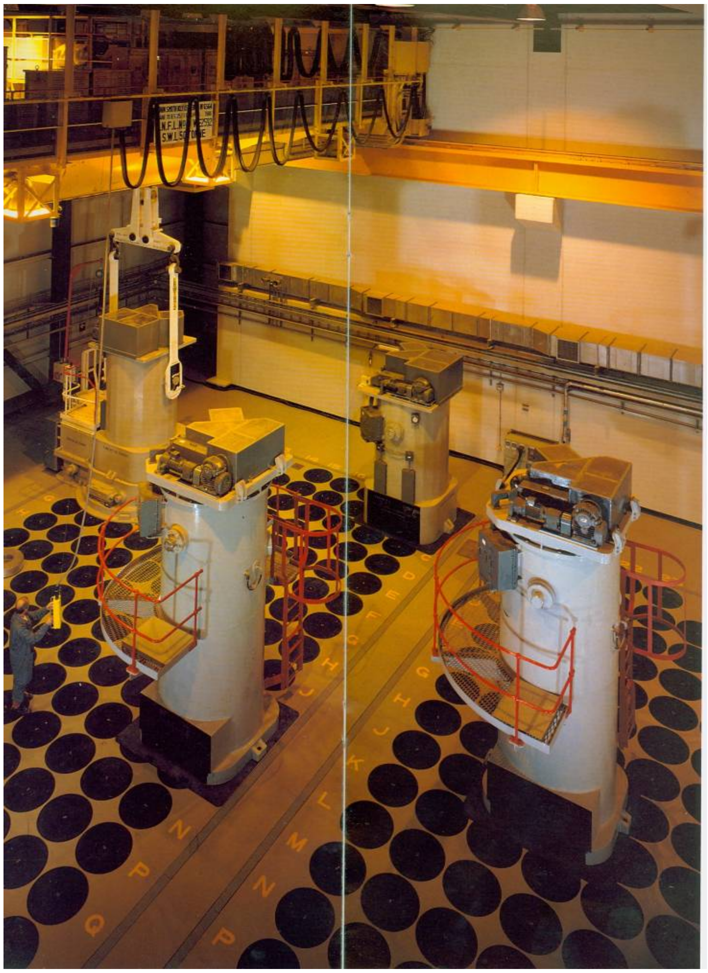


Figure E4-3. WVP VPS charge face showing waste product flasks over storage channels (BNFL 1991).

E4-4. MODULE INTERFACES

The wastes that would be stored in these facilities would be received from LLW-Greater-than-Class-C (GTCC) Conditioning and Packaging modules (Module G4). After the wastes have been decayed (e.g., 300 years) they would sent to Near Surface Disposal (Module J).

E4-5. SCALING CONSIDERATIONS

No scaling analysis has been completed for this module.

E4-6. COST BASIS, ASSUMPTIONS, AND DATA SOURCES

E4-6.1 Basis and Assumptions

Sound cost estimates require the examination of relevant design parameters, existing industrial practice, and the definition of a reference design concept for immobilization and storage of CsSr waste. The reference process need not be fully optimal, but should be broadly representative to provide a conservative basis for costing. The reference process adopted here for CsSr immobilization and storage is that of vitrification to form borosilicate glass of CsSr loading in the range 2–10% by weight depending on decay time before separation and immobilization. The shortest decay period is likely to be around 4 years and this requires use of canisters of diameter ~0.4 m, (e.g., AVH type), and low CsSr loadings. A modular vault dry storage system is selected consistent with worldwide practice for long storage of heat generating wastes. CsSr waste storage costs are then taken to be the same as HLW vitrified waste storage costs on a capital cost and yearly operational expense basis. Vitrified waste is planned to be stored 50–100 years before disposal whereas CsSr waste is planned to be stored 300 years prior disposal.

The current use of commercial MVDS facilities for storage of vitrified HLW and spent fuels for planned periods of ≤100 years gives confidence in this approach. However, the project and operational costs of these are often commercially protected. Estimates have been gathered where possible and values also taken from cost studies for planned facilities, which may not be constructed. Table E4-6 summarizes cost data given above.

Table E4-6. Summary cost data for managed decay storage.

Facility	Est. Capital Cost,[a] $M	Est. Operating Cost,[1] $M/yr	Comments
Vitrified Waste, MVDS			
WVP–VPS (1990)	150	4[b]	8,000 AVH canisters ≡ 16,000 t(SNF[c])
Hanford TWRS (1996)	106	4[b]	8,000 AVH canisters ≡ 16,000 t(SNF[c])
INEEL VWISF (2001)	126	4[b]	8,000 AVH canisters ≡ 16,000 t(SNF[c])
Oxide SNF, MVDS	Est. Life Cycle Cost		
Bunn et al. (2001)	$250M	3–4	1,000 t(SNF[c]) stored, LWR operation +40 years
Fairlie (2000)	~$200K/t(SNF[c])	-	

a. 2008 money values – Derived using US Army Corps of Engineers, Civil Works Construction Cost Index System (CWCCIS), Using CWBS Feature Code – 07 Power plant, Appendix A, EM 1110-2-1304, Appendix Revised September 30, 2007.

b. Operating cost based on team of 20 for 24-hour coverage at fully loaded cost of $200K/staff.

c. LWR oxide spent nuclear fuel.

The estimated capital costs of the vitrified waste stores are seen to be similar with possibly U.S. practice lower cost than UK.

Two sources provide similar values for annual operating costs for dry storage of vitrified waste and spent fuel. Due to the unusually long period of storage (300 years), the undiscounted total operating costs outweigh capital cost by around a factor of ~10.

Taking the operational expenses of a single module of vitrified CsSr waste air-cooled store, capacity of 64,000 kg(CsSr elemental) ≡ 16,000 t (oxide SNF), as $4M/year and the operational period as 300 years gives an operational expense of $1.2B. The capital cost of the store is estimated as $150M. The cost of decommissioning of the store is taken as 10% of capital cost plus 5 years operational cost.

CsSr Waste Storage Life Cycle Cost
= $4M × 300 + $150M + (5 × $4M + $15M) = $1,385M
CsSr Storage Unit Cost = 1,385M/64,000 = $21,600/kg(CsSr elemental)

The nominal cost of storage of CsSr is judged to be ~$22.5K/kg(CsSr elemental).

The low range cost is judged by halving the capital cost, discounting (starting at 100 years) the second and third centuries of operation to about 30% of their nominal value and halving decommissioning costs. This gives an unit value of around $10K/kg(CsSr elemental). Other opportunities include alternative glass formulations with high-temperature limits that enable increased CsSr concentrations and decreased numbers of waste canisters. Cold crucible induction melters are under development for HLW vitrification, which may supersede the two main industrial types, joule ceramic and hot wall induction, and provide higher temperature operation giving access to higher melting glass formulations with higher glass transition temperatures.

Risks for higher unit storage costs include requirements to increase operational team size, decrease the module storage capacity for each team, and/or increase of the required storage period to account for minor radionuclide decay. The high range cost is estimated as 60% higher than nominal (i.e., $35K/kg([s-Sr elemental]).

E4-6.2 Cost Correlations

Storage costs were correlated with the storage of spent oxide fuel and interim storage of vitrified waste.

E4-6.2.1 Storage of Spent Oxide Fuel Facility Costs

Further cost confirmation can be derived from data on storage of spent fuel. Storage of vitrified HLW is expected to not be dissimilar in cost to that of SNF since the thermal power and radiation level is similar though the mass of vitrified waste is about 80% less. Of the main alternatives of vault storage, casks, and housing arrays, the preferred option for long-term storage of large quantities of spent fuel and other heat generating wastes is dry vault storage. This concept for spent fuel was originated by Alstec (now Babcock International Group PLC) and deployed in the U.S. by Foster Wheeler (BNS 2009). There is considerable similarity between vault storage of vitrified waste and spent fuel.

Costs of dry cask interim storage of fuel are reported as about $250M for storage of 1,000 t (LWR SNF) during 40 years generation and for 40 years following this (Bunn et al. 2001). Casks are licensed by the Nuclear Regulatory Commission (NRC) for 20 year operational periods. Research and development (R&D) is underway to assess NRC's judgment is that storing fuel in dry casks would be safe for 100 years. This averages to $250K/t(iHM). Costs for dry vault storage were said to be similar. For shutdown reactors with all spent fuel in dry storage the annual costs were estimated as $3–4M, while for pool storage after reactor shutdown the annual costs were estimated as $9M (Bunn et al. 2001). Japanese and European SNF storage costs are in general higher than these costs.

The Modular Vault Dry Store (MVDS) system is used at the PAKS, Hungary (Figure E4-4) and at Fort St. Vrain (Figure E4-5) in the U.S. It is a passive system employing natural convection cooling. The Paks MVDS handles thermal power of up to 17 kWt of fuel when fully loaded compared with about 1.4 to 2 kWt for concrete casks. The PAKS MVDS uses a natural convection cooling system, as shown in Figure E4-6. Vaults have sufficient heat removal capacity to keep fuel-cladding temperatures of 5-year-cooled PWR assemblies below 200°C. The Fort St. Vrain MVDS was designed for a 40-year lifetime.

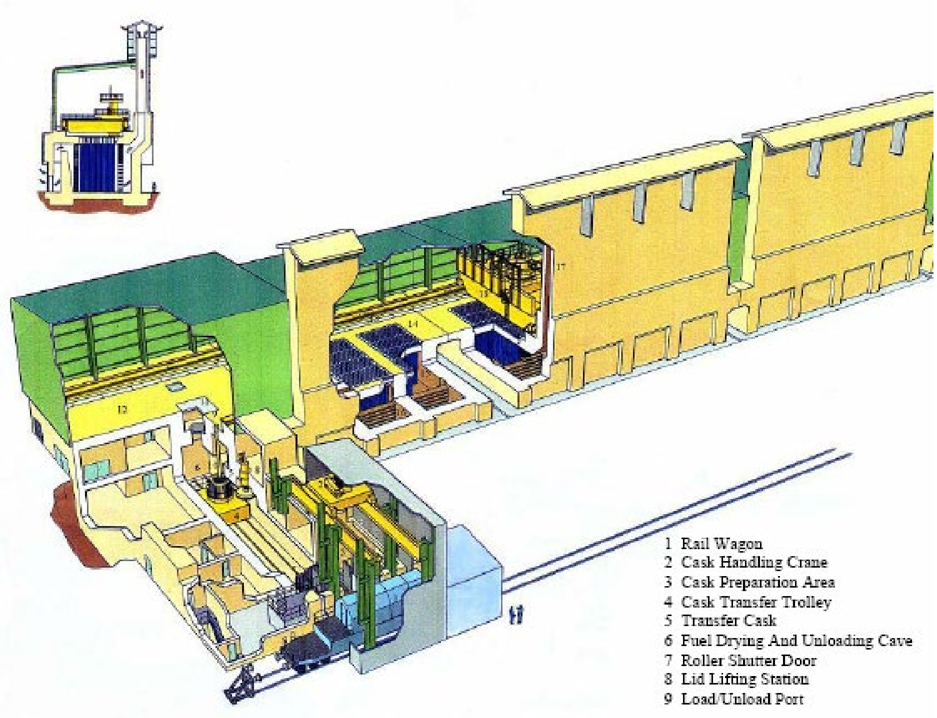
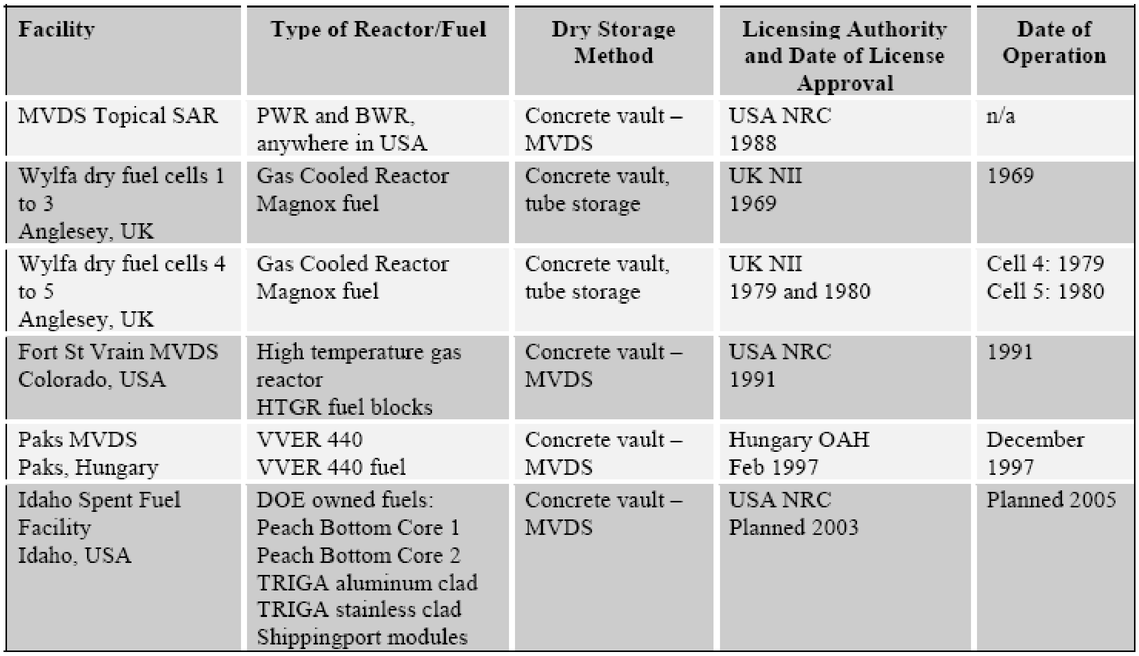


Figure E4-4. Schematic of modular vault dry storage of spent fuel at PAKS, Hungary (Ordogh et al. 2004).

Figure E4-5. Fort St. Vrain modular vault dry store (MVDS) showing charge face.

The status of MVDS facilities designed by Foster Wheeler and ALSTEC are shown in Table E4-7, below:

Table E4-7. Status of MVDS facilities by Foster Wheeler and ALSTEC (Roberts et al. 2003).



Facility	Type of Reactor/Fuel	Dry Storage Method	Licensing Authority and Date of License Approval	Date of Operation
MVDS Topical SAR	PWR and BWR, anywhere in USA	Concrete vault – MVDS	USA NRC 1988	n/a
Wylfa dry fuel cells 1 to 3 Anglesey, UK	Gas Cooled Reactor Magnox fuel	Concrete vault, tube storage	UK NII 1969	1969
Wylfa dry fuel cells 4 to 5 Anglesey, UK	Gas Cooled Reactor Magnox fuel	Concrete vault, tube storage	UK NII 1979 and 1980	Cell 4: 1979 Cell 5: 1980
Fort St Vrain MVDS Colorado, USA	High temperature gas reactor HTGR fuel blocks	Concrete vault – MVDS	USA NRC 1991	1991
Paks MVDS Paks, Hungary	VVER 440 VVER 440 fuel	Concrete vault – MVDS	Hungary OAH Feb 1997	December 1997
Idaho Spent Fuel Facility Idaho, USA	DOE owned fuels: Peach Bottom Core 1 Peach Bottom Core 2 TRIGA aluminum clad TRIGA stainless clad Shippingport modules	Concrete vault – MVDS	USA NRC Planned 2003	Planned 2005

Estimated dry storage costs from a variety of sources are given, see Table E4-8 (Fairlie 2000), that provide discussion of wide differences observed. A reasonable value of \$200K/t(iHM) is adopted with 2008 money value.

Table E4-8. Estimated dry storage costs for oxide spent fuels (Fairlie 2000).

STUDY*	LWR FUEL, \$K/t(iHM)
PAE-KfK (NuclearFuel 1993)	220[a]
OECD-NEA (1994)	225[b]
IAEA (1990)	82–165
Supko (1995)	50–100[c]
Wisconsin PS C (1994)	35–68[d]
Ontario Hydro, (Stevens-Guille, 1994; Nash, 1997)	15–20[e]

* Reported in Fairlie (2000)
a. Undiscounted
b. Levelised fuel cycle costs
c. Representative life cycle costs
d. Constant \$ analysis
e. Low burnup fuel. References given in Fairlie 2000.

Figure E4-6. Paks MVDS in 2000 showing natural convection cooling exhaust outlets (Ordogh et al. 2004).

German institutions give undiscounted estimates of about $225,000 per tonne of LWR fuel over indeterminate periods. Relatively expensive CASTOR spent fuel casks were used in these calculations. Estimated costs of U.S. and Canadian dry storage systems are significantly lower than European systems.

E4-6.2.2 Comparison with U.S. DOE HLW Vitrification and Interim Storage Costs

A valuable review and evaluation of interim storage facilities for application to Hanford Tank Waste Remediation System (TWRS) vitrified waste and Cs dry IX waste canisters was performed by Calmus (1996). He examined the four options of Standalone casks, Housing arrays, Vault storage, and Modification of existing major facilities to provide vault storage. The canister dimensions were vitrified waste: 0.61 m diameter × 3.0 m height (max 1 kW) or 0.61 m diameter × 4.57 m height (max 1 kW); and Cs IX waste: 0.33 m diameter × 1.37 m height (max 1.5 kW). The design requirements included air cooling. It was concluded that forced convection with high-efficiency particulate air (HEPA) filtration would generally be needed for reuse of existing facilities but natural convection with use of indirect cooling by means of loading waste canisters into storage sleeves (thimbles) was preferred for new facilities. Standalone casks and Housing arrays were found to be substantially more expensive.

Using passive above-ground air-cooled MVDS for vitrified waste canisters, Hanford TWRS estimated overall facility capital costs based on $37,000 (1996 mv) per storage tube of ~5 m tube height (Calmus 1996). A total of 2,000 tubes of ~5 m height then are required for 8,000 AVH type HLW canisters. This gives a capital cost of $74M (1996 dollars), which equates to around $106M (2008 dollars).

Idaho National Environmental and Engineering Laboratory (INEEL) reported a study of the Idaho Waste Vitrification Facilities Project—Vitrified Waste Interim Storage Facility (VWISF) (Aitken et al. 2001). Two scenarios were evaluated during this study. The first scenario includes individual storage tubes for the vitrified waste canisters (two canisters per tube) and a passive ventilation system. This option is called the "Hanford Option," because it is modeled after the Hanford vitrified waste storage design. The second scenario includes racks for holding the vitrified waste canisters and a mechanical ventilation system. The second option is labeled the "Savannah River Option," since it is modeled after the Savannah River Site's vitrified waste storage facility. The second option has lower total project cost, but higher life cycle cost. The costs interpolated here are for the first option. The major waste product resulting from the treatment process will be a vitrified waste glass. The glass will be placed in canisters approximately 15 ft (4.5 m) long × 2 ft (0.6 m) in diameter referred to as "Hanford Canisters." Under a “high” waste loading scenario in the IWVF melter, this will result in approximately 436 canisters of the treated SBW and 4,600 canisters of the treated calcine. Two canisters are stored in each Hanford thimble tube, which is equivalent to around 10 AVH type HLW canisters. This gives a capital cost of ~$95M (2001 dollars), which equates to around $126M (2008 dollars).

E4-7. DATA LIMITATIONS

The information obtained for these facilities is at a very high level. Additional details are not currently available.

E4-8. COST SUMMARIES

The module cost information is summarized in the What-It-Takes (WIT) cost summary in Table E3-7. The summary shows the reference capital cost basis (constant year U.S. dollars), the reference basis cost contingency (if known), the cost analyst’s judgment of the potential upsides (low end of cost range) and downsides (high end of cost range) based on references and qualitative factors, and selected nominal costs (judgment of the expected costs based on the references, contingency factors, upsides, and downsides). These costs are subject to change and are updated as additional reference information is collected and evaluated, and as a result of sensitivity and uncertainty analysis. Refer to Section 2.6 in the main section of this report for additional details on the cost estimation approach used to construct the WIT table.

The following cost values are proposed, which are summarized in Table E4-9:

1. This module determined a cost range for 300-year decay storage of immobilized, heat-generating, mixed cesium-strontium (CsSr-BaRb) waste such as generated by the AFCI UREX+ based fuel cycle and a reference waste immobilization and storage process defined here. The period of storage may allow subsequent shallow disposal as LLW.
2. The life cycle cost estimates for storage of CsSr waste are $22.5K/kg (CsSr) nominal, $10K/kg (CsSr) low range, and $35K/kg (CsSr) high range. The estimates use 2008 money values and are based on mass of cesium and strontium in elemental form (i.e., do not include Rb, Ba, oxide, or other glass or ceramic forming elements).
3. Due to the uniquely long lifecycle period, ~350 years for construction through decommissioning, of the CsSr storage facility (normally nuclear facilities have a lifecycle of <100) the summed operational costs significantly exceed the initial capital cost (e.g., by one order of magnitude). Under these circumstances, discounting of all costs following 100 years of operation may be more appropriate for decision analysis and the low range value includes this effect to some degree.
4. The operational expenditure cost estimates are based on facility design that is operationally conservative (i.e., assumes full 24 hour, 365 day per year) manning for the full period of storage even when decay power has reduced substantially, and are technically conservative, appropriate to the long period of storage, through adoption of modular vault dry stores, passive cooling, multiple containment barriers, and refractory waste form (borosilicate glass) of relatively low specific decay power. New melter types and higher transition temperature glasses may enable higher CsSr-Rb-Ba loadings, which can reduce container numbers and storage requirements.

5. Due to the unusual cost structure, optimization of facility design in relation to progressive reduction of staffing consistent with decreasing hazards with eventual remote monitoring and control and periodic inspection may yield significant reduction in undiscounted cost. This is analogous to “Safestor” designs for radioactive decay of nuclear reactor structures prior to decommissioning. Further cost and design studies are needed to substantiate potential cost reductions. Table E3-7 is the cost summary table of selected values for managed decay storage.

Table E4-9. Cost summary table of selected values for managed decay storage (storage of Cs/Sr in remote handling facility) including escalation to year 20175$.

What-It-Takes (WIT) Table					
Reference Unit Capital Cost based on xx MT(TRU) Capacity	Reference Cost Contingency (+/- %)	Low Cost $/(kg(Cs/Sr))	Mode Cost $/(kg(Cs/Sr))	Mean Cost $/(kg(Cs/Sr))	High Cost $/(kg(Cs/Sr))
From 2009 CBA $/kg(kg(Cs/Sr)	(± 25%)	$10,000	$22,500		$35,000
Mod. E4 Escalated to 2020$ (factor of 19%)		$11,930	$26,843	$26,843	$41,755

The triangular distribution based on the costs in the WIT Table E4-9 is shown in Figure E4-7. Figure E3-9 gives the estimated cost frequency distribution for the advanced fuel cycle product from, for example, UREX+ processing, (i.e., mixed plutonium, minor actinide, and uranium oxide material). The costs for the similar component metallic product from molten salt electrochemical processing are less well known, and until further work is performed may be taken as similar to the UREX+ oxide product.

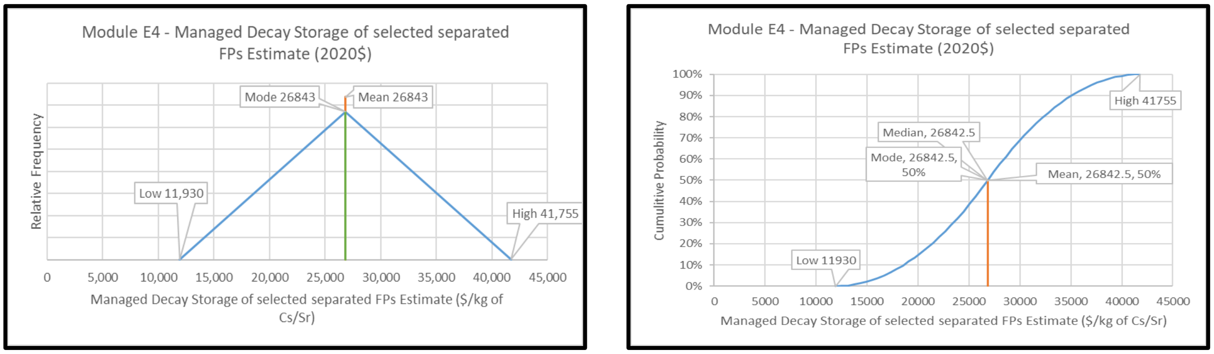


Figure E4-7. Probability Distributions for Unit Cost of Managed Decay Storage

E4-9. SENSITIVITY AND UNCERTAINTY ANALYSIS

None performed to date.

E4-10. REFERENCES

Aitken, S. et al., 2001, "Idaho Waste Vitrification Facilities Project Vitrified Waste Interim Storage Facility," INEEL/EXT-01-01217, September 2001.

Aloy, A.S. et al., 2007, "Development of Glass Matrix for Immobilization of CsSr Fraction after HLW Partitioning," *Proc. Global 2007 International Conference*, Boise, Idaho, U.S., pp. 1017–1021, 2007.

Alvarez, R., 2007, *Radioactive Wastes and the Global Nuclear Energy Partnership*, pp. 1–23, Institute for Policy Studies, Washington D.C., 2007.

Balmer, M. L. et al., 2000, *New Silicotitanate Waste Forms: Development and Characterization, Environmental Management Science Program*, Project Number: 60345, pp. 1.19–1.28, 2000.

Benedict, M. et al., 1981, *Nuclear Chemical Engineering*, 2nd Ed, pp. 580–590, McGraw-Hill, 1981.

Bergelson, B. R. et al., 2005, *Radiation Protection Dosimetry*, Vol. 115, No. 1-4, pp. 445–447, 2005.

BNFL, 1991, *The Vitrification Plant*, British Nuclear Fuels plc, 1991.

BNS, 2009, "Spent Fuel Dry Storage", BNS Nuclear Services, http://www.babcock.co.uk/opco/nuclear/activities.php?pageID=75118, Last accessed September 2, 2009.

Bunn, M., et al., 2001, "Interim Storage of Spent Nuclear Fuel, A Safe, Flexible and Cost-Effective Near-Term Approach to Spent Fuel Management," Harvard University and University of Tokyo, June 2001.

Calmus, R. B., 1996, *TWRS HLW Interim Storage Facility Search and Evaluation*, WHC-SD-WM-ES-374, Rev. 0, May 1996.

Carter, M. L., et al., 2007, "HIPed Tailored Ceramic Waste Forms for the Immobilization of Cs, Sr and Tc," *Proc. Global 2007 International Conference*, Boise, Idaho, U.S., pp. 1022–1028, September 2007.

Dobson, A. J. and C. Phillips, 2006, "High Level Waste Processing in the UK–Hard Won Experience that can benefit U.S. Nuclear Cleanup Work," *WM'06 Conference*, Tucson, Arizona, February 26–March 2, 2006.

El-Kamash, A. M., et al., 2006, *Immobilization of Cesium and Strontium radionuclides in zeolite-cement blends, J. Hazardous Materials*, Vol. 136, Issue 2, August 2006.

Fairlie, I., 2000, "Dry Storage of Nuclear Fuel: The Safer Alternative to Reprocessing," Response to Cogema Dossiers to the La Hague Public Inquiry, London, UK, May 2000.

Hebditch, D., et al., 2007, "Issues for Conceptual Design of AFCF and CFTC LWR Spent Fuel Separations influencing Next-Generation Aqueous Fuel Reprocessing," *Proc. Global 2007 International Conference*, Boise, Idaho, USA, September 2007.

IAEA, 1992, *Design and Operation of High Level Waste Vitrification and Storage Facilities*, Technical Report Series, No. 339, Vienna, 1992.

Kaminski M. D. and C. J. Merz, 2007, Personal communication, Argonne National Laboratory, Illinois, 2007.

Law, J. D., et al., 2004, "Development of a Cobalt Dicarbollide/Polyethylene Glycol Solvent Extraction Process for Separation of Cesium and Strontium to Support Advanced Aqueous Reprocessing," *Nuclear Technology*, 147(2), pp. 284–290, 2004.

Law, J. D., et al., 2006, “Development of Cesium and Strontium Separation and Immobilization Technologies in Support of an Advanced Nuclear Fuel Cycle,” *WM’06 Conference*, Tucson, Arizona, February 26–March 2, 2006.

Law, J. D., et al., 2007, “Flowsheet Testing of the Fission Product Extraction Process as Part of Advanced Aqueous Reprocessing,” INL/CON-07-12827, *PREPRINT 2007ANS National Meeting*, June 2007.

Luca, V., et al., 2006, “Tungsten bronze-based nuclear waste form ceramics. Part 1 Conversion of microporous tungstates to leach resistant ceramics,” *J. Nuclear Materials*, Vol. 358, pp. 139–150, 2006.

National Academy of Sciences, 2003, “Improving the Scientific Basis for Managing DOE’s Excess Nuclear Materials and Spent Nuclear Fuel,” Chapter 5, p. 52, 2003.

Ordogh, M., et al., 2004, “Operational Experiences and Future Plans for Hungarian Spent Fuel Storage,” *WM’04 Conference*, Paper WM-4232, Tucson, Arizona, USA, March 2004.

Ortega, L. H. and S. M. McDeavitt, 2007, “Precursors for the Immobilization of Radioactive Cesium and Strontium from Spent Nuclear Fuel,” Global 2007, pp. 1385–88, Boise, Idaho, U.S., 2007.

Roberts, R., et al., 2003, “The Idaho Spent Fuel Project Update–January 2003,” *WM’03 Conference*, Tucson, Arizona, February 23–27, 2003.

Streatfield, R. E. et al., 2006, “Gas Generation in Radioactive Wastes - MAGGAS Predictive Life Cycle Model,” *Waste Management WM’06 Conference*, Tucson, Arizona, Paper #6241, February 26–March 2, 2006.

Vandegrift, G. F. et al., 2004, “Designing and Demonstration of the UREX+ Process Using Spent Nuclear Fuel,” *ATALANTE 2004*, International Conference on Advances for Future Nuclear Fuel Cycles, Nimes, France, June 21–24, 2004.

E4-11. BIBILOGRAPHY

Bresee, J., et al., 2007, “Advanced Reactor, Fuel Cycle, and Energy Products,” *DOE Workshop for Universities*, Hilton Hotel, Gaithersburg, MD, March 20, 2007.

Gombert, D., et al., 2007, “Global Nuclear Energy Partnership Integrated Waste Management Strategy Technical and Policy Issues for Implementation,” INL/EXT-07-12620, May 2007.

Japan Nuclear Fuel Limited, 2008, Vitrified Waste Store, http://www.jnfl.co.jp/english/wastestorage.html, Web page last accessed September 2, 2009.

NuclearFuel, 1993, Direct Disposal Route Will Retain Cost Advantage. November 22 1993. p7.

OECD-NEA, 2006, “The Roles of Storage in the Management of Long-lived Radioactive Waste, Practices and Potentialities in OECD Countries,” pp. 1–64, NEA Report No. 6043, 2006.

Wigeland, R. A. et al., 2006, “Separations and Transmutation Criteria to Improve Utilization of a Geologic Repository,” *Nuclear Technology*, Vol. 154, pp. 95–106, April 2006.