U.S. RADIOACTIVE WASTE INVENTORY AND CHARACTERISTICS RELATED TO POTENTIAL FUTURE NUCLEAR ENERGY SYSTEMS

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
Used Fuel Disposition
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SUMMARY

Purpose

In February, 2011 the Blue Ribbon Commission (BRC) on America's Nuclear Future requested the Department of Energy (DOE) to provide a white paper summarizing the quantities and characteristics of potential waste generated by various nuclear fuel cycles. The BRC request expressed interest in two classes of radioactive wastes:

- Existing waste that are or might be destined for a civilian deep geologic repository or equivalent.
- Potential future waste, generated by alternative nuclear fuel cycles (e.g. wastes from reprocessing, mixed-oxide (MOX) fuel fabrication, and advanced reactors such as Sodium Fast Reactors (SFR).

This paper summarizes three existing reports prepared for the DOE's Office of Nuclear Energy (DOE-NE), Fuel Cycle Technology (FCT) program: "Fuel Cycle Potential Waste For Disposition" Rev 3 [Ref 1], "LLW Disposition, Quantity and Inventory" Rev 1 [Ref 2] and "LLW Inventory from MOX Fuel Fabrication" Rev 0 [Ref 3] and includes waste form loadings (Table S-3).

The evaluations presented are subject to several sources of uncertainty including: historical accuracy, the timeline for implementation of future fuel cycles, technology advancement during the implementation period and changing environmental regulations which may place a greater (or lesser) demand on waste capture and treatment processes. Differing assumptions are used throughout this study to aide in developing a range of quantities and characteristics for waste requiring disposal. The use of a single scenario or data point from a single example is discouraged.

Used Spent Nuclear Fuel and DOE High Level Waste Inventory

The current inventory of used/spent nuclear fuel (USNF) through 2010 is shown in Table S-1. Projected inventories of USNF from four potential future nuclear generation scenarios have been evaluated:

- Scenario 1 assumes no replacement of existing nuclear generation reactors.

- Scenario 2 assumes the amount of current nuclear generation is maintained at the current levels (100 GWe/yr) with new reactors replacing the existing reactors as the existing reactors are decommissioned.

- Scenario 3 assumes the amount of nuclear generation will increase to 200 GWe/yr from 2020 to 2060, and remain at 200 GWe/yr until the end of the century.

- Scenario 4 assumes the amount of nuclear generation will increase to 400 GWe/yr from 2020 to 2060, and remain at 400 GWe/yr until the end of the century.

These scenarios indicate the total potential USNF inventory could range from about 140,000 MT to nearly 700,000 MT by the end the century assuming the nuclear power generation quadruples rapidly.

Table S-1 Commercial USNF Estimated Discharges through 2010.

Total Number of Assemblies[b]			Total Initial Uranium (MTU)[a]			Average Enrichment		Average Burnup (MWd/MTU)[c]		Average Age (Yr)		Total Radioactivity (Ci)	
PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300	14.9	15.4	16 billion	7 billion
a the estimated fuel discharged has been rounded to the nearest 100 MTU, totals may not appear to sum correctly													
b the number of assemblies has been rounded to the nearest 200, totals may not appear to sum correctly													
c the burn-up has been rounded to the next 100 MWd/MT													

Table S-2 Projected Total Number of High Level Waste Canisters

	HLW Canisters[1] Best Estimate	Potential HLW Canister Range
West Valley	275	NA[2]
Hanford[4]	10,713	9,746-12,100
INL (Calcine)	3,328	1,190-11,200
INL (Electro-chemical processing)	102	82-135
SRS[5]	7,560	7,560-9,450
Total	21,980	18,900-33,200[3]
1. With the exception of Hanford all HLW canisters are 2 feet $\times$ 10 feet, Hanford HLW canisters are 2 feet $\times$ 14.76 feet		
2. All the West Valley HLW canisters currently exist		
3. Rounded to nearest 100 canister		
4. Reference 24		
5 Reference 10		

Light Water Reactor Used Spent Nuclear Fuel Reprocessing Wastes

Two aqueous reprocessing technologies for light water reactor (LWR) USNF are evaluated. The Co-Extraction process represents the next generation of commercial reprocessing methods which co-recover uranium and plutonium while generating off-gas wastes, metal hardware, and a single fission product waste. New extraction represents an advanced process which recovers uranium and TRU (transuranic) elements (Np, Pu, Am, Cm) while generating similar waste streams. Table S-3 compares the volume of waste from these processes to volume of a bare USNF fuel assembly. These results indicate that there maybe very little benefit in terms of volume reduction associated with recycling LWR fuel. However it must be recognized that the characteristics of the waste differ significantly from bare fuel and from themselves in terms of radionuclide content and decay heat production. These characteristics in addition to volume have a significant role on the management of these materials. In addition, future research and development on advanced waste forms could lead to higher waste loading densities and a reduction in the volume of wastes that would be generated from recycling.

Sodium Fast Reactor Used Spent Nuclear Fuel Reprocessing Wastes

An electro-chemical reprocessing technology for advanced burner reactor (ABR), specifically SFR, USNF is evaluated. Inventories for off-gas wastes, metal and fission product wastes from electro-chemical reprocessing are provided.

Secondary Waste

Secondary waste inventory from geologic disposal of USNF is provided in the report (Section 2.4). Secondary waste inventories for aqueous reprocessing of LWR USNF (Section 4.2.2 and 4.3.2), electro-chemical reprocessing of SFR USNF (Section 6.3.3) and fabrication of MOX fuel (Section 5.2) from plutonium obtained from the reprocessing of LWR USNF are also provided.

Table S-3 Estimated Volume of Waste Forms from Recycling Including Waste Loading Assumptions

	Co-Extraction	New Extraction	Waste Form and Waste Loading Assumptions
I from Off Gas Recovery (m^3/MT)	0.0054	0.0054	Grouted silver mordenite Iodine is loaded to 170 mg per gram of mordenite which is then 25 wt% of the final grouted waste form
Tritium from Off Gas Recovery (m^3/MT)	0.0025	0.0025	Recovered tritiated water is solidified as 30 wt% tritiated water in grout
C-14 from Off Gas Recovery (m^3/MT)	0.0042	0.0042	Converted to a carbonate and solidified as 30 wt% carbonate in grout
Kr from Off Gas Recovery (m^3/MT)	0.12	0.12	Separated by cryogenic methods and stored in high pressure type A gas cylinders. The Gas is approximately 98% Kr and 2% Xe.
Hulls and Hardware (m^3/MT)	0.075	0.075	Decontaminated, compacted and placed inside a canister
Fission Product (m^3/MT)	0.18[a]	0.17[b]	Borosilicate glass. To avoid multi-phase glass formation a decay heat limit of 14,000 watts per canister has been established. Specific radionuclides limits are determined based on the isotopic composition of the waste which varies by separation methodology. [a] For the Co-extraction glass in this example, fission product waste loading is approximately 10.3 wt %. [b] For the New Extraction glass in this example, fission product waste loading is approximately 15.3 wt %.
Total (m^3/MT)	**0.39**	**0.38**	
Total Volume Change relative to PWR Assembly	91%	88%	
Total Volume Change relative to PWR Assembly, Fission Product Waste Only	42%	39%	

Revision History

Revision 1:

1) revised the number of DOE HLW canisters in Tables S-2 and 3-4 to reflect the latest system planning bases from DOE-EM for Hanford and Savannah River. Corrected a footnote error in Table S-2.

2) corrected the row descriptor for the last row of Table S-3 and 7-1 and corrected a rounding error on the same row of Table S-3.

3) corrected the column alignments in Table 4-2 and 4-5 and added footnotes to these tables

4) deleted the figures related to secondary waste - mixed GTCC . Text was added to provide the same information.

5) in a few places the text was clarified.

Revision 2:

1) waste loading assumption information was added to Tables S-3, 4-1, 4-2, 4-4, 4-5, 6-2, 6-4 and 7-1.
2) All units converted to SI.

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ACRONYMS

ABR	Advanced Burner Reactor
BRC	Blue Ribbon Commission
BWR	Boiling Water Reactor
COEX™	a recycling method trademarked by AREVA which co-recovers Uranium and Plutonium
CR	Conversion Ratio
DOE	Department of Energy
DOE-EM	Department of Energy Environmental Management
DOE-NE	Department of Energy Office of Nuclear Energy
EAS	Engineering Alternative Studies
EIS	Environmental Impact Statement
FCR&D	Fuel Cycle Research and Development
FCT	Fuel Cycle Technology
GNEP	Global Nuclear Energy Partnership
GTCC	Greater Than Class C
GWd	Giga-Watt days (thermal)
GWd/MT	Giga-Watt days (thermal) per Metric Ton
GWe	Giga-Watts electric production
GWSB	Glass Waste Storage Building
HIP	Hot Isostatic Pressing
HLW	High Level Waste
HM	Heavy Metal
HS	Hanford Site
INL	Idaho National Laboratory
LLW	Low Level Waste
LWR	Light Water Reactor
MOX	Mixed Oxide
MT	Metric Ton
MTHM	Metric Tons of Heavy Metal
MTU	Metric Tons of Uranium
MWd	Mega-Watt days (thermal)
NNPP	Naval Nuclear Propulsion Program
NRC	Nuclear Regulatory Commission

NUEX	New-Extraction
NWPA	Nuclear Waste Policy Act
OCRWM	Office of Civilian Radioactive Waste Management
ORIGEN	Oak Ridge Isotope Generation and Depletion Code
PEIS	Programmatic Environmental Impact Statement
Pu	Plutonium
PWR	Pressurized Water Reactor
SFR	Sodium Fast Reactor
SRS	Savannah River Site
TALSPEAK	Trivalent Actinide Lanthanide Separation by Phosphorus-based Aqueous Komplexes
Tc	Technetium
TRU	Transuranic
TRUEX	Transuranic Extraction
U	Uranium
UDS	Undissolved Solids
UOX	Uranium Oxide Fuel
UREX	Uranium Extraction
USNF	Used/Spent Nuclear Fuel
WAC	Waste Acceptance Criteria
WIPP	Waste Isolation Pilot Plant
W/MT	Watts per Metric Ton
WVDP	West Valley Demonstration Project

Used Fuel Disposition
U.S. Radioactive Waste Inventory and Characteristics
Related to Potential Future Nuclear Energy Systems

1.   Introduction

In February, 2011 the Blue Ribbon Commission (BRC) on America's Nuclear Future requested the Department of Energy (DOE) to provide a white paper summarizing the quantities and characteristics of potential waste generated by various nuclear fuel cycles. The BRC request expressed interest in two classes of radioactive wastes:
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This paper summarizes three existing reports prepared for the DOE's Office of Nuclear Energy (DOE-NE), Fuel Cycle technology (FCT) program: "Fuel Cycle Potential Waste For Disposition" Rev 3 [Ref 1], "LLW Disposition, Quantity and Inventory" Rev 1 [Ref 2] and "LLW Inventory from MOX Fuel Fabrication" Rev 0 [Ref 3] and includes waste form loadings. Additional details on the methods used in generating these waste estimates as well as results not provided in this summary can be obtained from the references listed above.

The evaluations presented are subject to several sources of uncertainty including: historical accuracy, the timeline for implementation of future fuel cycles, technology advancement during the implementation period and changing environmental regulations which may place a greater (or lesser) demand on waste capture and treatment processes. Differing assumptions are used throughout this study to aide in developing a range of quantities and characteristics for waste requiring disposal. The use of a single scenario or data point from a single example presented is discouraged.

2.   Commercial USNF from the Once-Through Fuel Cycle

Commercial nuclear power plants have operated in the United States since about 1960. There are currently 104 operating nuclear power plants. Used/Spent Nuclear Fuel (USNF) from these operating plants is currently stored on site in pools or dry storage casks with disposal in a geologic repository envisioned in a once-thru fuel cycle. In addition, USNF from 14 shutdown reactors is currently stored on the reactor sites. The General Electric facility at Morris, Illinois is currently the only USNF licensed storage facility in operation that is not collocated at an active or former reactor site.

2.1   Current Inventory of Commercial USNF

Reference 1 summarizes the current inventory data collected in support of the Office of Civilian Radioactive Waste Management's (OCRWM) efforts for licensing the Yucca Mountain Repository [Ref 4]. Information collected from RW-859 forms is available on an assembly basis for USNF discharges from 1968 through 2002. Data are also available that was collected to support RW activities on a batch bases for fuel discharges from 1968 through April 2005 [Ref 5].

The specific USNF data available from these sources are:
- Reactor type (Pressurized Water Reactor (PWR) or Boiling Water Reactors (BWR).
- Number of assemblies.
- Burn-up by assembly or batch.
- Date of discharge.
- Initial uranium loading.
- Initial enrichment.

To develop an inventory estimate through 2010, fuel discharge predictions developed for the Nuclear Energy Institute in 2005 were used to estimate the number of assemblies and metric tons of uranium [Ref 6]. To estimate the average

enrichment and burn-up through 2010, projections made by utilities as part of the RW-859 surveys were used. These projections are documented in OCRWM's "Calculation Method for the Projection of Future Spent Fuel Discharges", February 2002 [Ref 7]. These projections identified a burn-up increase of 2.38% per year for BWR fuel and 1.11% per year for PWR fuel through 2010. The enrichment increases at the same rate as burn-up. Comparison of these projections made in 1998 to actual data collected through 2004 show very good agreement (PWR - actual 46,950 MWd/MTU vs. projected 46,922 MWd/MTU; BWR - actual 43,447 MWd/MTU vs. 42,787 projected MWd/MTU). Table 2-1 provides an estimate of the commercial USNF discharged through 2010.

Table 2-1 Commercial USNF Estimated Discharges through 2010.

Total Number of Assemblies[b]			Total Initial Uranium (MTU)[a]			Average Enrichment		Average Burnup (MWd/MTU)[c]		Average Age (Yr)		Total Radioactivity (Ci)	
PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300	14.9	15.4	16 billion	7 billion
a the estimated fuel discharged has been rounded to the nearest 100 MTU, totals may not appear to sum correctly													
b the number of assemblies has been rounded to the nearest 200, totals may not appear to sum correctly													
c the burn-up has been rounded to the next 100 MWd/MT													

2.2   Characteristics of the Current Inventory

The current inventory has an average burn-up of approximately 39.6 giga-watt days per metric ton (GWd/MT) for PWRs and 33.3 GWd/MT for BWRs. Figure 2-1 provides a distribution of this estimated inventory as a function of burn-up. Nearly 100% of the fuel currently being discharged exceeds the "high burn-up" threshold of 45 GWd/MT defined by the Nuclear Regulatory Commission (NRC). The maximum burn-up from the current reactor fleet is nearing 60 GWd/MT which is limited by both the 5% U-235 licensing basis for the current enrichment and fuel fabrication plants, and the reactor licensing basis to 62.5 GWd/MT.
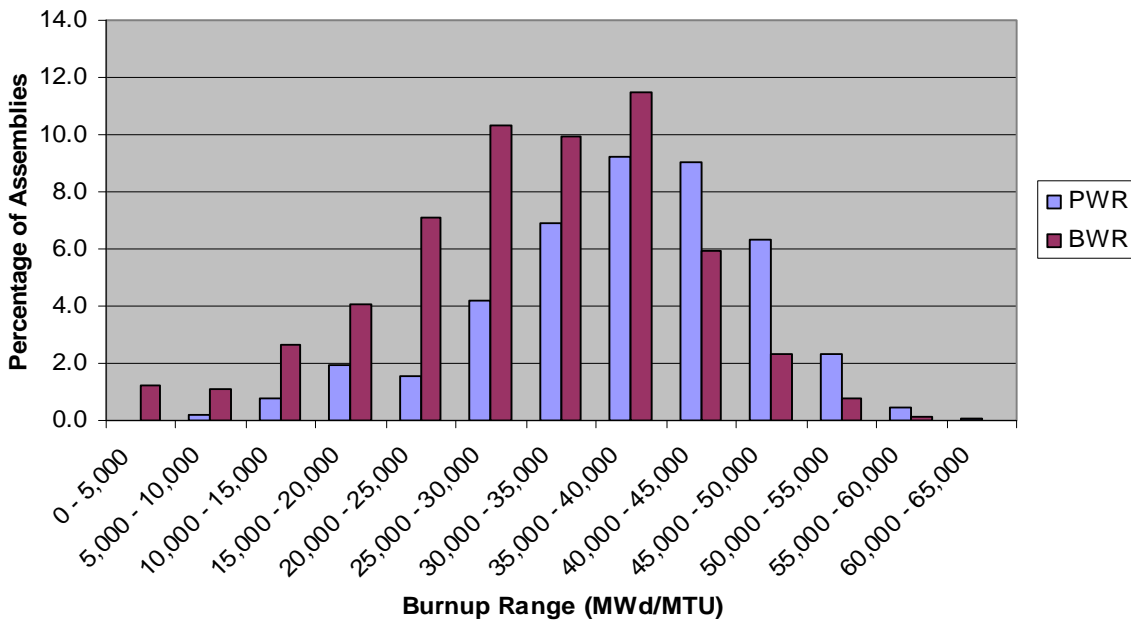
Figure 2-1 Percentage of Assemblies per Burn-up Range Current Inventory

Reference 1 developed a radionuclide inventory for USNF compositions for a representative range of compositions in the current and projected future inventory. The radionuclide decay algorithms and isotopic parameters from Oak Ridge Isotope Generation and Depletion (ORIGEN) code version 2.2 were used in estimating the decay heat.

Figure 2-2 provides the decay heat projections for 60 GWd/MT PWR maximum burn-up. The figure provides the total decay heat and isotopic groups with similar isotopic parameters that make up the total. The total decay heat for 1 year cooled fuel from the average burn of 40 GWd/MT is 10,500 watt per metric ton (W/MT) or somewhat less than the 14,000 W/MT from the maximum burn-up (60 GWd/MT) fuel.
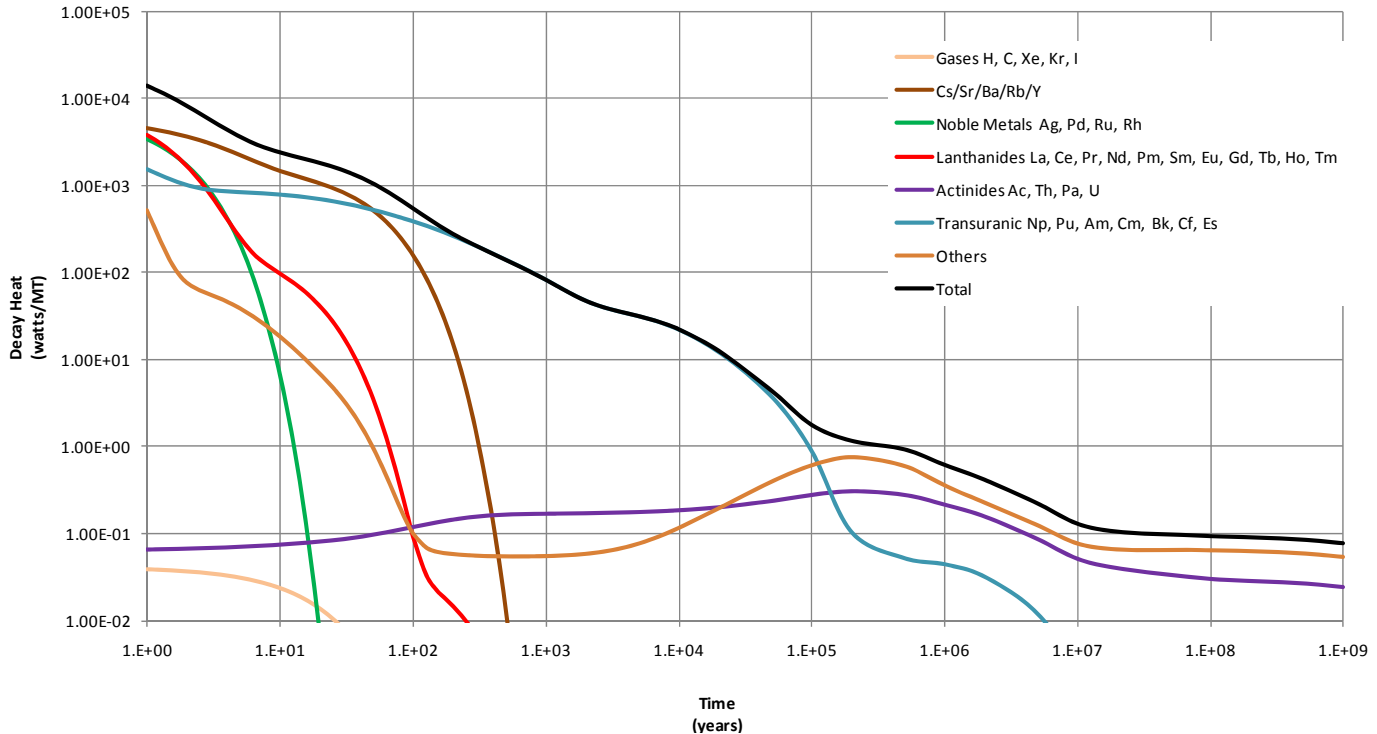
Figure 2-2 PWR 60 GWd/MT Used Fuel Decay Heat

2.3   Future Commercial USNF Inventory Projections

Four scenarios were used to evaluate the projected increases in the commercial Light Water Reactor (LWR) USNF inventory [Ref 1]. The four scenarios were selected from those previously evaluated by DOE in the Global Nuclear Energy Partnership (GNEP) Programmatic Environmental Impact Statement (PEIS) with only slight modification for the operational date of the first new reactors (2020 vs. 2015) and the end of the construction period (2060 vs. 2060 to 2070). Use of these scenarios does not constitute an endorsement; the scenarios were selected to provide a wide range of LWR fuel inventory for use in future analysis.

- Scenario 1 assumes no replacement of existing nuclear generation reactors.
- Scenario 2 assumes the amount of current nuclear generation is maintained at the current levels (100 GWe/yr) with new reactors replacing the existing reactors as the existing reactors are decommissioned.
- Scenario 3 assumes the amount of nuclear generation will increase to 200 GWe/yr from 2020 to 2060, and remain at 200 GWe/yr until the end of the century.
- Scenario 4 assumes the amount of nuclear generation will increase to 400 GWe/yr from 2020 to 2060, and remain at 400 GWe/yr until the end of the century.

Assuming an electrical demand growth of 0.7% annually, scenario 3 results in a decrease in the nuclear share of electricity production from about 20% to about 15% in 2030, while scenario 4 results in an increase in the nuclear share to about 30% by 2030.

Figure 2-3 projects the mass (MT of uranium) cumulatively to the end of the century. The inventory can range from about 140,000 MT assuming no replacement reactors are constructed, to nearly 700,000 MT assuming nuclear power generation quadruples between 2020 and 2060.
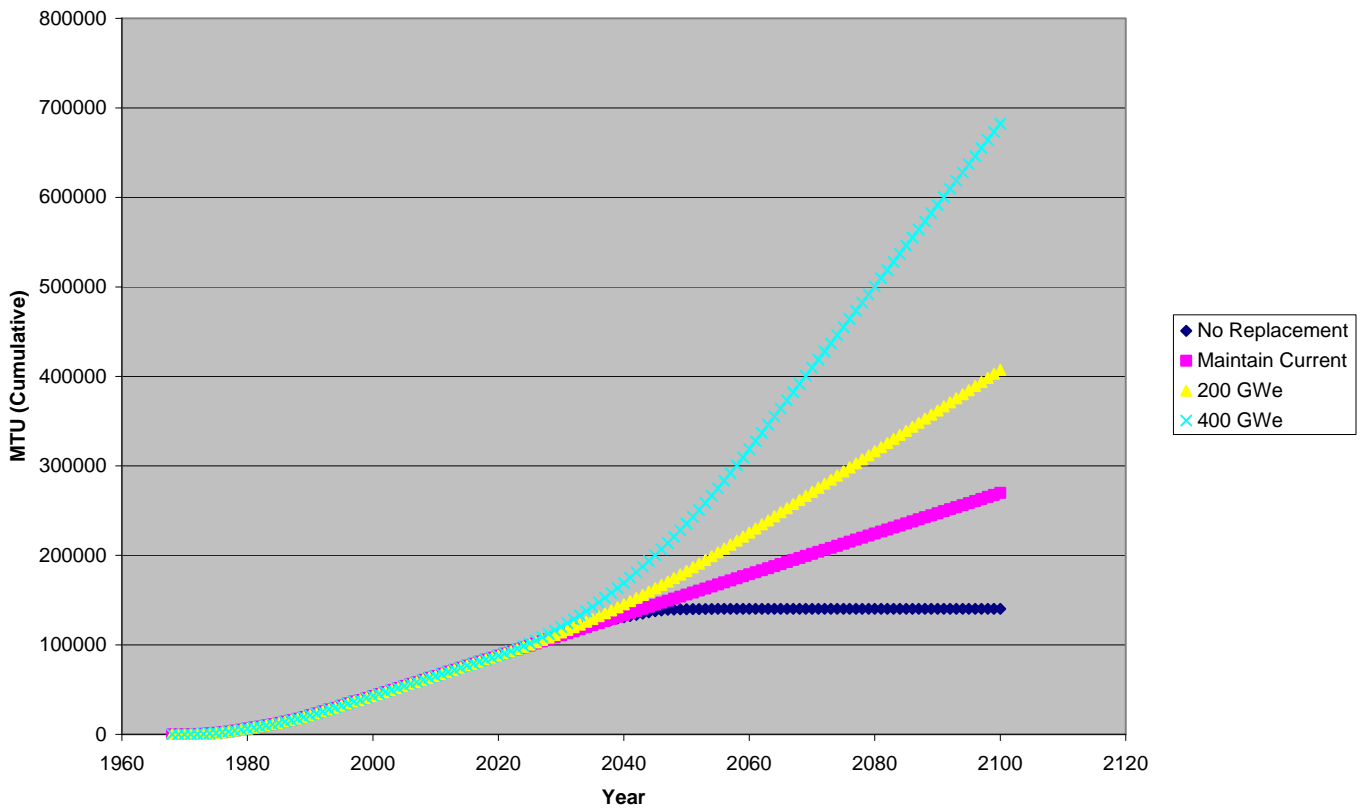
Figure 2-3 Cumulative USNF Mass Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr Cases.

The enrichment and burn-up in these scenarios is expected to continue to increase to the current United States enrichment plant limit of 5% U-235, which corresponds to a maximum burn-up slightly less than 60 GWd/MT.

As an example of an alternative fuel cycle from continuous improvement of the current United States reactor fleet, Figure 2-4 provides decay heat information for 100 GWd/MT LWR uranium oxide (UOX) used fuel. This burn-up requires an enrichment of approximately 8.3% U-235. The total decay heat increases to about 19,500 W/MT after one year of cooling.
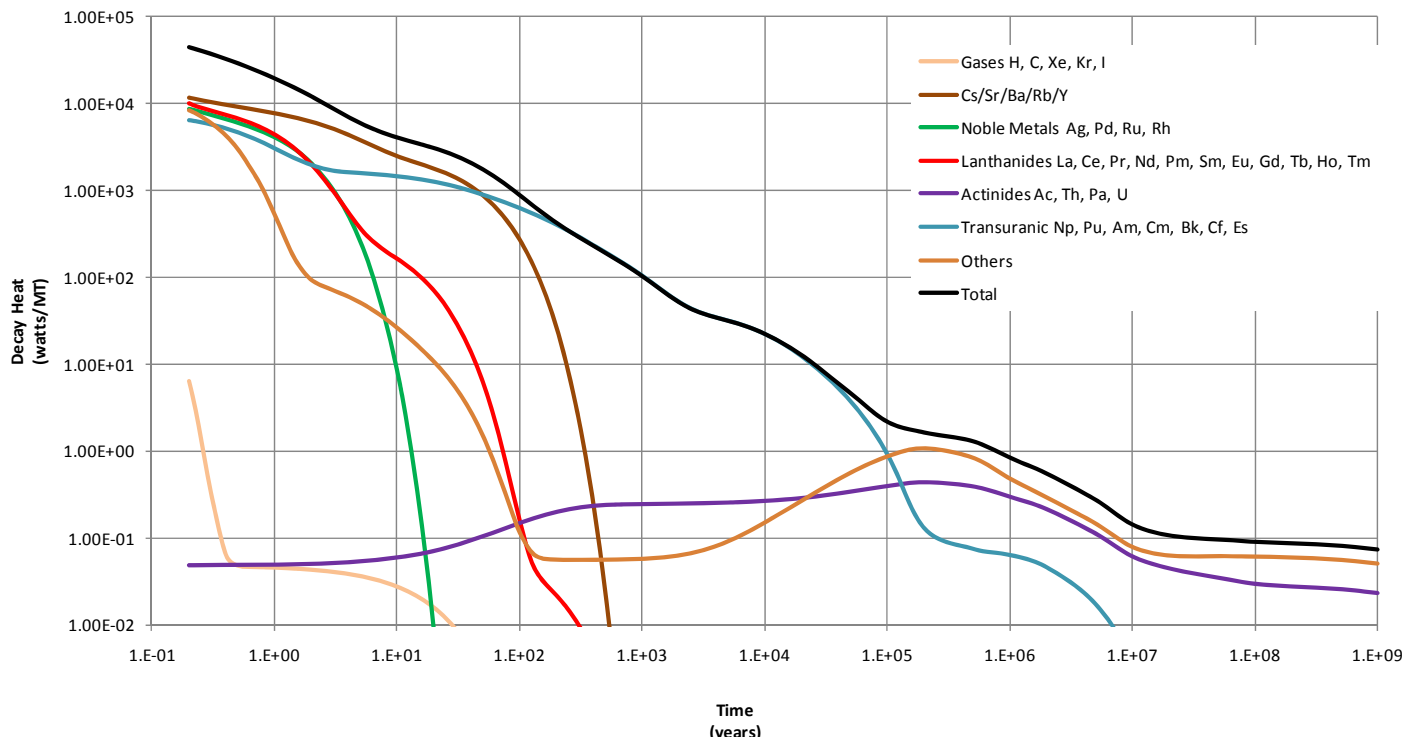
Figure 2-4 PWR 100 GWd/MT Used Fuel Decay Heat

2.4 Secondary Wastes From Repository Operations

Secondary wastes associated with the once-through fuel cycle are those generated by the handling and emplacement activities involved in the disposal of USNF at a geologic repository. Low Level Waste (LLW) estimates for repository operations are provided in the *Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (hereinafter referred to as the Supplemental (Environmental Impact Statement) EIS [Ref 8]. These waste estimates are based on the disposal of 11,000 waste packages in the proposed Yucca Mountain repository containing no more than a total of 70,000 MT of heavy metal (MTHM) of USNF and high level waste (HLW). Sources of secondary waste from repository operations include:

- Cask, facility and equipment decontamination activities.
- Pool system skimming and filtration operations.
- Used dual purpose canisters.
- Tooling and clothing.
- Facility ventilation filtration.
- Chemical sumps.
- Carrier and transporter washings.

All of the radioactive waste streams from repository operations are classified as Class A, B or C LLW. No greater than Class C (GTCC) or mixed wastes are anticipated from repository operations.

The LLW estimates assume that most of the USNF and HLW (90%) is received in disposable canisters that will not require opening at the repository but will be placed directly into a waste package for disposal in the repository. The estimate assumes the remaining 10% of the USNF will arrive in dual purpose canisters that will require opening and repackaging. Appendix A, Section A.2 of the Draft Supplemental EIS provides a sensitivity analysis that evaluates the impacts of receiving only 75% of commercial USNF in disposable canisters. To accommodate the additional USNF that

would require repackaging at the repository, the analysis considers the construction and operation of an additional Wet Handling Facility and the elimination of one of three Canister Receipt and Closure Facilities to repackage the remaining 75% of the USNF not considered in the sensitivity analysis. If none of the USNF is received in disposal containers, five additional Wet Handling Facilities would be needed and the remaining two Canister Receipt and Closure Facilities can be eliminated.

Figure 2-5 shows the volume of LLW estimated from repository operations based on the fraction of USNF received in directly disposable containers.
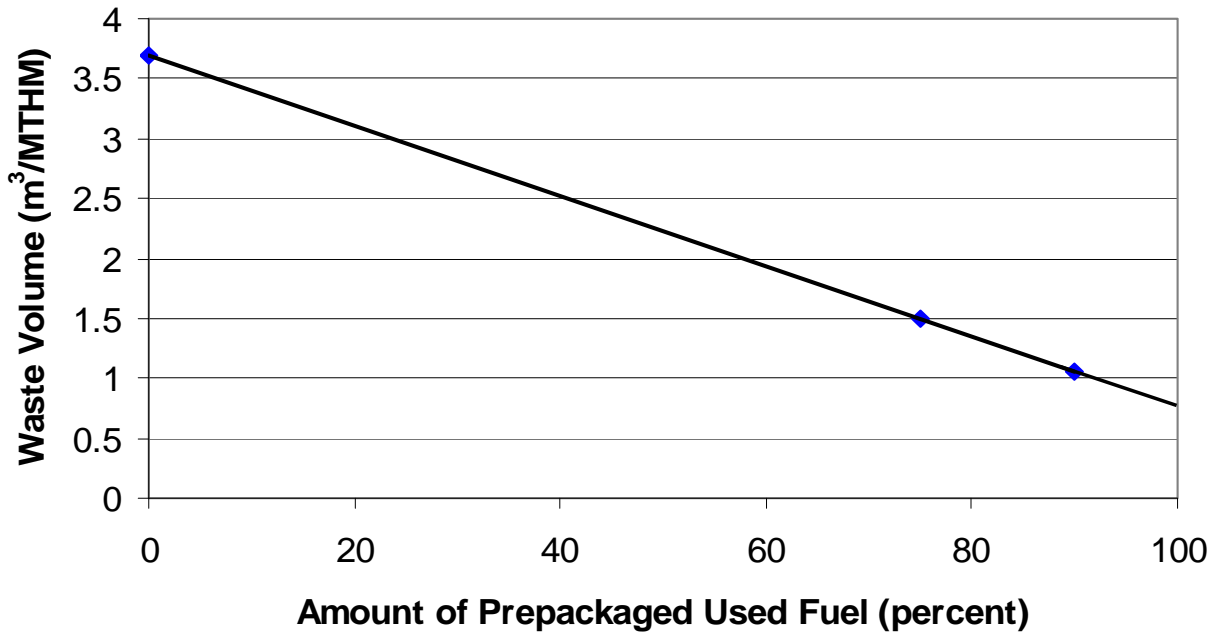
Figure 2-5 LLW Volume from Repository Operations

3.   DOE Materials for Disposition

Since the inception of nuclear reactors, the DOE and its predecessor agencies operated or sponsored a variety of research, test, training, and other experimental reactors both domestically and overseas. The Naval Nuclear Propulsion Program (NNPP) has generated USNF from operation of nuclear powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored training ship reactor plants, early development of commercial nuclear power, and conduct of irradiation test programs.

Aqueous reprocessing of DOE USNF has occurred at the Hanford Site, the Idaho National Laboratory (INL), and the Savannah River Site (SRS). SRS is actively vitrifying the legacy HLW into borosilicate glass and the Hanford Site has a similar facility under construction. The INL has calcined HLW and is pursuing the use of electro-chemical processing to treat 60 MTHM of sodium bonded USNF. DOE is also responsible for clean-up of the commercial USNF reprocessing site at West Valley, New York.

The waste requiring disposition from these DOE activities are fairly well understood and documented. This section summarizes these wastes.

3.1   DOE Fuels

DOE USNF is primarily generated by DOE production reactors, demonstration commercial power reactors, and domestic and foreign research and training reactors. DOE USNF includes some commercial USNF not in the possession of NRC licensed commercial utilities such as Shippingport, Peach Bottom, and Fort St. Vrain, which is stored within the DOE

complex. This USNF was generated for commercial power demonstration purposes or as part of research projects. Also, the Three Mile Island Unit 2 USNF debris is stored at the INL.

3.1.1 DOE Fuel Inventory

DOE USNF comes from a wide range of reactor types, such as light- and heavy-water-moderated reactors, graphite-moderated reactors, and unmoderated (fast) reactors, with various cladding materials and enrichments, varying from depleted uranium to over 93% enriched U-235. Many of these reactors, now decommissioned, had unique design features, such as core configuration, fuel element and assembly geometry, moderator and coolant materials, operational characteristics, and neutron spatial and spectral properties. There is a large diversity of reactor and fuel designs. In addition, there is a relatively large number (over 200,000) of fuel pieces (assemblies, rods, rod segments, or canistered debris, etc.) which range from a large number of pieces for some reactors (N Reactor) to a few individual pieces for other unique reactors (Chicago Pile-5 converter cylinders).

The majority of DOE USNF (about 2,500 MTHM) has been irradiated and is in storage. DOE continues to operate several research reactors and will be receiving USNF from universities and the foreign research reactor return program. This amount of projected material is relatively small (about 50 MTHM) and there is some uncertainty in the final quantity to be received.

Although DOE USNF is stored throughout the United States at numerous facilities, a decision was made in 1995 to consolidate DOE USNF at three existing DOE sites; the Hanford Site in Washington, the INL in Idaho, and the Savannah River Site in South Carolina. The vast majority of DOE USNFs are currently stored at these three sites. The storage configurations vary for each of the sites and include both dry and wet storage. On a MTHM basis, a large portion of the USNF (about 2,100 MTHM) is contained in about 400 sealed canisters almost all of which is N Reactor fuel in dry storage at the Hanford site. The remaining 400MT will result in an additional 3,000 canisters.

3.1.2 DOE Fuel Characteristics

Process knowledge and the best available information regarding fuel fabrication, operations, and storage for DOE USNF is used to develop a conservative source-term estimate. The DOE USNF characterization process relies on pre-calculated lookup tables that provide radionuclide inventories for selected representative USNF at a range of decay times. These results are used as templates that are scaled to estimate radionuclide inventories for other similar fuels. Pre-calculated radionuclide inventories are extracted from the appropriate template at the desired decay period and then scaled to account for differences in fuel mass and burn-up for individual items in the inventory. Table 3-1 lists the projected radionuclide inventory of DOE USNF for the nominal and bounding cases as of 2010. The nominal case is the expected or average inventory. The bounding case represents the highest burn-up assembly or accounts for uncertainties if fuel burn-up is not known. Reference 1 Appendix D, Table D-2 provides a more detailed isotopic inventory of DOE USNF for the nominal and bounding cases as of 2010.

Table 3-1 DOE SNF Radionuclide Inventory Range in 2010

Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
1.91×10^8	3.48×10^8

3.2 Naval Fuel

The NNPP has generated USNF from operation of nuclear powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored training ship reactor plants, early development of commercial nuclear power, and conduct of irradiation test programs. The source of naval USNF information for this report is the Yucca Mountain Repository License Application.

3.2.1 Naval Fuel Inventory

Naval USNF consists of solid metal and metallic components that are nonflammable, highly corrosion-resistant, and neither pyrophoric, explosive, combustible, chemically reactive, nor subject to gas generation by chemical reaction or off-

gassing. Approximately 27 MTHM of Naval USNF currently exists with a projected inventory of less than 65 MTHM in 2035.

USNF from the NNPP is temporarily stored at the INL. To accommodate different naval fuel assembly designs, naval USNF is loaded in either a naval short USNF canister (66 inches nominal outside diameter by 185.5 inches in length) or a naval long USNF canister (210.5 inches long). Both were sized to fit within the proposed design for the Yucca Mountain repository waste package.

Approximately 400 naval USNF canisters (310 long and 90 short) are currently planned to be packaged and temporarily stored pending shipment. The NNPP is responsible for preparing and loading naval USNF canisters and began canister loading operations in 2002. As of February 2010, 27 naval USNF canisters have been loaded and are being temporarily stored at INL.

3.2.2   Naval Fuel Characteristics

Table 3-2 provides the radionuclide inventory for a representative naval USNF canister. A period of five years after reactor shutdown was selected for use in the repository source term analyses. Reference 1 Appendix E, Table E-1 provides an isotopic inventory per canister.

Table 3-2 Naval Fuel Radionuclide Inventory per Canister (5 years after discharge)

— Inventory Per Canister (Ci/Canister)	Total Inventory [1] (Ci)
1.45×10^6	~580×10^6

[1] 400 canisters times 1.45×10^6 Ci/canister

3.3   DOE High Level Waste Treatment

High-level radioactive waste is the highly radioactive material resulting from the reprocessing of USNF. Following aqueous reprocessing, HLW is in a liquid form and initially stored in underground metal storage tanks. Long term storage of HLW requires stabilization of the wastes into a form that will not react, nor degrade, for an extended period of time. Hanford and SRS have continued to store the HLW generated at those sites in the liquid form prior to final treatment. At INL, the liquid waste was solidified for interim storage by calcination.

3.3.1   DOE HLW Current Inventory

A commercial fuel reprocessing plant located at West Valley, New York operated from 1966 through 1972 and reprocessed approximately 640 MT of USNF to recover the unused uranium. Of the USNF reprocessed at West Valley, about 260 MT was commercial used fuel and about 380 MT was DOE N Reactor used fuel. During operations about 2,500 m^3 of liquid HLW was generated [Ref 9]. The liquid HLW was vitrified between 1996 and 2001 producing 275 HLW canisters that are stored at West Valley [Ref 30].

The INL reprocessed USNF from naval propulsion reactors, test reactors, and research reactors to recover uranium and generated approximately 30,000 m^3 of liquid HLW. Between 1960 and 1997, the INL converted all of their liquid HLW into about 4,400 m^3 of a solid waste form called calcine (a granular solid with the consistency of powder laundry soap). These solids are stored retrievably on-site in stainless steel bins (like grain silos but smaller) within concrete vaults that are mostly underground.

The SRS has reprocessed defense reactor USNF and nuclear targets to recover valuable isotopes since 1954 producing more than 530,000 m^3 of liquid HLW. Through evaporation and vitrification of the waste, SRS has reduced this inventory to the current level of about 136,000 m^3 of liquid HLW [Ref 10]. SRS began vitrifying liquid HLW in 1996 and thru December, 2010 has produced 3,059 HLW canisters (2 feet × 10 feet) [Ref 11].

The Hanford Site reprocessed approximately 100,000 MT of defense reactor USNF between 1944 and 1988 generating approximately 2,000,000 m^3 of liquid HLW. Disposal to the ground and evaporation reduced the volume to about 220,000 m^3 of liquid HLW to recover the plutonium, uranium, and other elements for defense and other federal programs.

Construction of a vitrification facility is currently underway with startup scheduled for 2019. Table 3-3 summarizes the current HLW inventory.

Table 3-3 Current High Level Waste Inventory

	HLW Canisters[1]	Liquid HLW[2] (m^3)	Dry HLW[3] (m^3)
West Valley	275	N/A	N/A
Hanford	N/A	220,000	N/A
INL	N/A	N/A	4,400
SRS[4]	3,059	136,000	N/A
Total	3,334	256,000	4,400
1. Vitrified HLW in stainless steel canisters			
2. HLW stored in tanks			
3. Calcined HLW stored in bins.			
4. SRS canister and HLW Inventory as of December 31, 2010.			

The Hanford Site encapsulated cesium and strontium separated from the HLW between 1974 and 1985. Some of these capsules were leased to companies as radiation sources. After one of the capsules developed a microscopic leak, the capsules were recalled [Ref 31]. Hanford is storing 1,335 cesium capsules and 601 strontium capsules, approximately 109 million curies (as of 8/8/06) [Ref 16].

3.3.2   DOE HLW Projected Inventory

SRS currently has the only operating aqueous reprocessing facilityin the U.S, H Canyon. It is estimated that an additional 17,000 m^3 of liquid HLW may be generated with continued canyon operations (until approximately 2019).

At Hanford and SRS the liquid HLW currently in storage will be vitrified and placed in HLW canisters. Although the final waste form is known, the final number of HLW canisters cannot be definitely estimated because of potential changes in waste loading and processing.

At INL several options were considered for ultimate disposal of the calcine. Alternatives included direct disposal, vitrification, or hot isostatic pressing (HIP) to compress the calcine into a volume reduced monolithic waste form. A Record of Decision issued December 2009 stated that DOE will use the HIP technology to treat the calcine for final disposal in a geologic repository.

The INL is also pursuing an electro-chemical process for treating the 60 MTHM of sodium bonded USNF from INL, Hanford and Fermi. The process has been demonstrated and used to treat about 4 MTHM of sodium bonded USNF to date. The HLW generated from this process are converted into an iron based alloy and a glass bonded mineral [Ref 31].

Based on the proposed actions, Table 3-4 shows the projected number of HLW canisters that are estimated to be produced. The current best estimate and a potential range are provided [Ref 10, 12, 13].

The vitrified HLW at SRS is stored in below grade concrete vaults, called Glass Waste Storage Buildings (GWSB), containing support frames for vertical storage of 2,262 HLW canisters. SRS currently has two GWSB constructed and a third planned. The HLW canisters at West Valley are currently stored in a shielded cell in the former reprocessing plant. Alternate interim storage options are being considered at West Valley to allow decommissioning of the reprocessing facility. Hanford has its first canister storage facility and plans to construct more as needed when the WTP becomes operational.

Table 3-4 Projected Total Number of High Level Waste Canisters

	HLW Canisters[1] Best Estimate	Potential HLW Canister Range
West Valley	275	NA[2]
Hanford[4]	10,713	9,746-12,100
INL (Calcine)	3,328	1,190-11,200
INL (Electro-chemical processing)	102	82-135
SRS[5]	7,560	7,560-9,450
Total	21,980	18,900-33,200[3]

1. With the exception of Hanford all HLW canisters are 2 feet × 10 feet, Hanford HLW canisters are 2 feet × 14.76 feet
2. All the West Valley HLW canisters currently exist
3. Rounded to nearest 100 canister
4. Reference 24
5 Reference 10

3.3.3   DOE HLW Characteristics

Table 3-5 lists the total HLW radionuclide inventory for each of the generating sites decayed to 2017. Reference 1 Appendix F, Table F-1 provides an isotopic composition for these sites. This Hanford site data in these tables do not include the Cs/Sr capsules as these were not included in YMP license application pending a final disposal decision. This inventory was developed by OCRWM for the license application of Yucca Mountain based on estimates provided by the sites [Ref 14]. Although there may be some variation in the number of canisters produced for the sites that have not completed waste treatment, the total radioactivity will not change.

OCRWM used the "projected maximum" inventory on a per canister basis for the HLW curie content supplied by SRS. The use of the "projected maximum" on a per canister basis resulted in a conservative total curie content for SRS that is approximately twice the actual SRS tank farm inventory. The expected curie content of SRS HLW is about 418 million curies [Ref 15]. A detailed isotopic composition is presented in Reference 1 Appendix F, Table F-2.

SRS is also the only site continuing reprocessing and the DOE Environmental Management (DOE-EM) program periodically requires disposal of special isotopes via the reprocessing facility and the vitrification process. For example excess weapons plutonium has been disposed which results in the Pu concentration of some SRS canisters to be above the projected maximum inventory used in the licensing of Yucca Mountain. The potential for future DOE-EM special isotope disposal campaigns has not been assessed in this study.

The detailed isotopic inventory for treatment of sodium bonded used nuclear fuel is shown in Reference 1 Appendix F, Table F-3 [Ref 13].

Table 3-5 Total Radionuclide Inventory for each HLW Glass Site at 2017

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
Total	1.34×10^8	9.54×10^8	1.46×10^7	2.58×10^7	1.13×10^9

NOTE: [a]Radionuclide inventory for Idaho National Laboratory HLW canister is provided for year 2035.HS = Hanford Site; INL = Idaho National Laboratory; SRS = Savannah River Site; WVDP = West Valley Demonstration Project.

4.   Reprocessing Commercial LWR Fuel

Existing and potential future commercial LWR USNF reprocessing methods vary in process complexity and technical maturity. Generally the objective of additional complexity is to recover valuable material or lessen the potential environmental impact of the resulting waste disposition activities. To support future evaluations of potential environmental impact, Reference 1 investigated the types and characteristics of waste which would be generated by three aqueous reprocessing methods and one electro-chemical reprocessing method. These evaluations are subject to several sources of uncertainties including the timeline for implementation (at least 20 years), technical advances during the implementation period, and changing environmental regulations which can place greater (or lesser) demands on the waste capture and treatment processes. Differing assumptions are used throughout this study to expand (as a whole) the range of quantities and characteristics of the waste for disposition. The use of any single study point is discouraged.

The study documented in Reference 1, investigated the waste mass, volume and characteristics which would be generated by reprocessing 20, 40 and 60 GWd/MT PWR and 15, 30 and 50 GWd/MT BWR burn-ups over a broad range of potential ages (5, 30, 100 and 500 years). This broad range of potential ages allows evaluation of various strategic decisions (i.e., to process short or long cooled fuel).

The following sections discuss two examples for aqueous reprocessing methods and one electro-chemical example as might be envisioned for alternative fuel cycles. Use of a particular technology in these examples are not technically mandated (e.g., use of aqueous processing for recycling oxide fuel), other reprocessing methods could be used in the fuel cycle examined.

4.1   Baseline Waste Forms

The GNEP Integrated Waste Management Strategy Baseline Study summarized the state-of-the-art in stabilization concepts for byproduct and waste streams, and recommended a baseline of waste forms for the safe disposition of proposed waste streams from future fuel recycling processes [Ref 17]. This baseline has been adopted for this study as applicable to the specific reprocessing method.

4.1.1   Off-gas Waste Forms

Tritium (H-3) is not captured nor treated with current generation reprocessing methods (aqueous methods practiced commercially and electro-chemical methods practiced by INL). Tritium is currently released to the environment via atmospheric or waste water discharges. This release is assumed to be an unacceptable practice (inconsistent with 10 CFR20) in future domestic reprocessing applications. To prevent the aqueous phases from becoming contaminated with tritium, voloxidation (not a current commercial practice) is used to ensure tritium is released to the off-gas system where it is captured as tritiated water. The tritiated water is converted to a grout and allowed to cure in a 10 liter container, which is subsequently contained in a double steel box.

Iodine (I-129) is captured on silver mordenite. The mordenite is then grouted and allowed to cure in a 55 gallon drum.

Carbon (C-14) is converted to carbonate and grouted. The grout is cured in a 55 gallon drum.

Krypton (Kr-85) is separated from the other off-gas components (including xenon) by cryogenic methods and the Kr-85 is stored in high pressure type A gas cylinders.

There is considerable uncertainty in the need to capture and treat both the C-14 and Kr-85 released during reprocessing with many factors (e.g. reprocessing facility location and environmental regulations) influencing the final decision. To provide a comprehensive range of the waste quantities these waste forms were not assumed as part of the Co-Extraction processing method (to better reflect current practice) while they have been included in the other aqueous and electro-chemical alternatives (to provide a bounding inventory).

There is also considerable debate regarding the classification of these wastes. The Nuclear Waste Policy Act (NWPA) of 1982 defines HLW as the "highly radioactive materials from reprocessing including liquid waste …" and these wastes have been considered HLW in some studies. Other studies have relied upon definitions provided in the NRC regulations

which define HLW as the waste following the first aqueous extraction step in which case these wastes are not considered HLW since they are generated prior to this step. This study has included these wastes to provide a bounding inventory.

Reference 1 also investigates 4 alternative off-gas waste forms currently being investigated by the DOE. These include a single alternative for Kr capture and 3 alternatives for iodine capture and treatment.

4.1.2 Metal Waste Forms

Compacted hulls and hardware: After being separated from the fuel, the assembly hardware (principally iron and nickel-based alloys) and zirconium (mostly) and stainless-steel-based cladding are decontaminated, compacted and placed inside a HLW canister. Each canister is 2 feet in diameter by 10 feet tall and contains 3,600 kg of waste material. This form contains activated metals and transuranic elements to the point it will likely be a GTCC.

Metal alloy: In the electro-chemical process those elements which are more noble (as measured by electro-chemical potential) than uranium such as the hulls, hardware and noble metal fission products and Tc remain as metals. The metal waste is decontaminated by volatilizing any adhered salts, melted into cylinders and then stacked into a HLW canisters. Each canister is 2 feet in diameter by 10 feet tall and contains 3,600 kilogram of waste material. This waste form is generally considered as HLW due to the inclusion of fission products.

4.1.3 Principle Fission Product Waste Forms

Borosilicate Glass: In the aqueous processes most of the fission products are incorporated into a borosilicate glass. While this waste form is the accepted standard for reprocessing HLW disposal, the waste form is limited by a number of attributes which must be considered.

The limits to avoid the formation of multiphase glasses include:

- Maximum decay heat of 14,000 watts per 2 feet diameter canister to prevent the canister centerline temperature from reaching the transition temperature.
- Molybdenum trioxide solubility is limited to 2.5% by weight.
- Noble (Ag, Pd, Rh, Ru) metals are limited to 3% by weight.

The limit selected for any representative fuel allows the maximum waste loading and minimum projected waste volume, and mass. The glass is cast into a 2 feet diameter by 15 feet tall canister containing 2,900 kilograms of glass.

Glass Bonded Zeolite: The electro-chemical process purges excess salt and fission products by occluding onto zeolite. Additional zeolite is added to sequester the excess salt chloride and then bonded with 25wt% borosilicate glass. The glass bonded zeolite is sintered into a 2 feet diameter by 3 feet tall cylinder which is then stacked into 15 feet tall canister containing 2,900 kilograms of glass.

Lanthanide Glass: The electro-chemical process also separates the lanthanides which are converted to a lanthanide based glass. The glass is cast into a 6"diameter by 60" tall canister containing 500 kilograms of glass. The waste loading is 50% lanthanides. In some studies, this lanthanide waste is combined with the glass bonded zeolite with an increase in volume and decay heat.

Reference 1 also investigates 5 alternative waste forms to explore the uncertainties in fission product waste.

4.2 Commercial Reprocessing Methods

The Co-Extraction method represents the simplest and most technically mature aqueous reprocessing method evaluated. The process envisioned is similar to the current generation of deployed reprocessing technology (e.g., the Rokkasaho Reprocessing Facility) except that processing conditions are modified such that uranium and plutonium are recovered together (no pure plutonium separation). This represents the next generation of available reprocessing methods. The principle fission product wastes including the minor actinides are combined with the undissolved solids (UDS) and recovered technetium (Tc) into a single borosilicate glass wasteform. The single fission product/heat generating waste is a focus of the this reprocessing method, to limit both processing and waste handling.

The gaseous radionuclides I-129 and H-3 released during reprocessing are captured and converted to waste forms suitable for disposal while C-14 and Kr-85 are assumed to be released to the atmosphere.

While this process is similar in function to the industrial COEX™ process developed by AREVA, the two processes assume different processing conditions and steps and so the product and waste streams cannot be directly compared.

The U/Pu recovered is converted to a Mixed Oxide (MOX) fuel which is subsequently irradiated in a conventional LWR to about 50 GWd/MT (Section 5.0). This MOX fuel is not assumed to be reprocessed and is disposed after a single reactor pass. Development continues on reprocessing methods for MOX fuel.

4.2.1   Wastes from Co-Extraction Reprocessing

The potential waste from Co-Extraction reprocessing of USNF with a burn-up of 51 GWd/MT fuel is provided in Tables 4-1 and 4-2. The fuel was selected to correspond with the parameters associated with prior MOX fuel studies described in Section 5. The waste is also typical of fuel being discharged from commercial reactors. The waste listed represent those typical of current reprocessing technology except for the volatile radionuclides which as noted in Section 4.1 are not currently captured in international applications. Additional information in Reference 1 includes the waste estimates generated by reprocessing various fuel burn-ups and age at the time of reprocessing. Table 4-4 and 4-5 of Reference 1 provides estimates for alternative off-gas and fission product waste. For example, the iodine waste form mass and volume can be $\pm 10 \times$ of the values reported in Table 4-1 depending upon the waste form selected. Alternative fission product waste mass and volume indicate reductions ranging from $3 \times$ to $10 \times$.

Table 4-3 provides the mass, volume and containers of uranium recovered. This material is not generally considered a waste and is planned to be held as a resource for future reactor fuels. In France, AREVA has recently begun re-enrichment of this material to produce a blended fresh uranium/recycled uranium fuel for use in conventional LWR. The data on uranium recovery is provided here for completeness.

Table 4-1 Co-Extraction Fuel Reprocessing Off-gas Waste Summary Including Waste Loading Assumptions

	Captured Tritium Grouted					**Captured I on Silver Mordenite Grouted**			
	Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kilogram of cured grout					Containers: 55 gallon drum. Each drum contains 460 kilogram of cured grout(decay heat is <0.5 Watts/Canister)			
Burn-up (GWd/MT)	Mass (kilogram/ MT)	Volume (m³/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions	Mass (kilogram/MT)	Volume (m³/MT)	Containers per MT	Waste Loading Assumptions
51	2.10	2.55 E -3	0.09	0.18	Recovered tritiated water is solidified as 30 wt% tritiated water in grout	11.74	5.38E-3	0.03	Grouted Silver Mordenite Iodine is loaded to 170 mg per gram of mordenite which is then 25 wt% of the final grouted waste form

Table 4-2 Co-Extraction Fuel Reprocessing Metal and Fission Product Waste Summary Including Waste Loading Assumptions

	Compacted Metal			**Borosilicate Glass**			
	Containers: 2 ft. diameter x 10 ft. tall canisters. Each Canister Contains 3,600 kilogram. (decay heat is <0.5 Watts/Canister)			Containers: 2 ft. diameter x 15 ft. tall canisters. Each Canister Contains 2,900 kilogram.			
Burn-up (GWd/MT)	Mass (kilogram/ MT)	Volume (m³/MT)	Containers per MT	Mass (kilogram/MT)	Volume (m³/MT)	Containers per MT	Decay Heat (W/container)
51	300.5	7.42E-2	0.084	537.5	0.247	0.19	14,000[a]
a To avoid multi-phase glass formation a decay heat limit of 14,000 watts per canister has been established. Specific radionuclides limits are determined based on the isotopic composition of the waste which varies by separation methodology. Waste loading is approximately 10.3% fission products as limited by decay heat							

Table 4-3 Co-Extraction Fuel Reprocessing Recovered Uranium Summary

Burn-up (GWd/MT)	Recovered Uranium (U_2O_3) Containers: 55 gal Drum canisters. Each Canister Contains 400 kilogram. (decay heat is <0.5 Watts/Canister)		
	Mass (kilogram/MT)	Volume (m^3/MT)	Containers per MT
51	1,097	0.57	2.74

4.2.1.1   Co-Extraction Borosilicate Glass Characteristics

The isotopic composition for the principle heat generating waste from the Co-Extraction process, the borosilicate glass was decayed using the ORIGEN 2.2 methods and isotopic parameters.

Figure 4-1 provides the decay heat characteristics as a function of time for the Co-Extraction borosilicate glass.
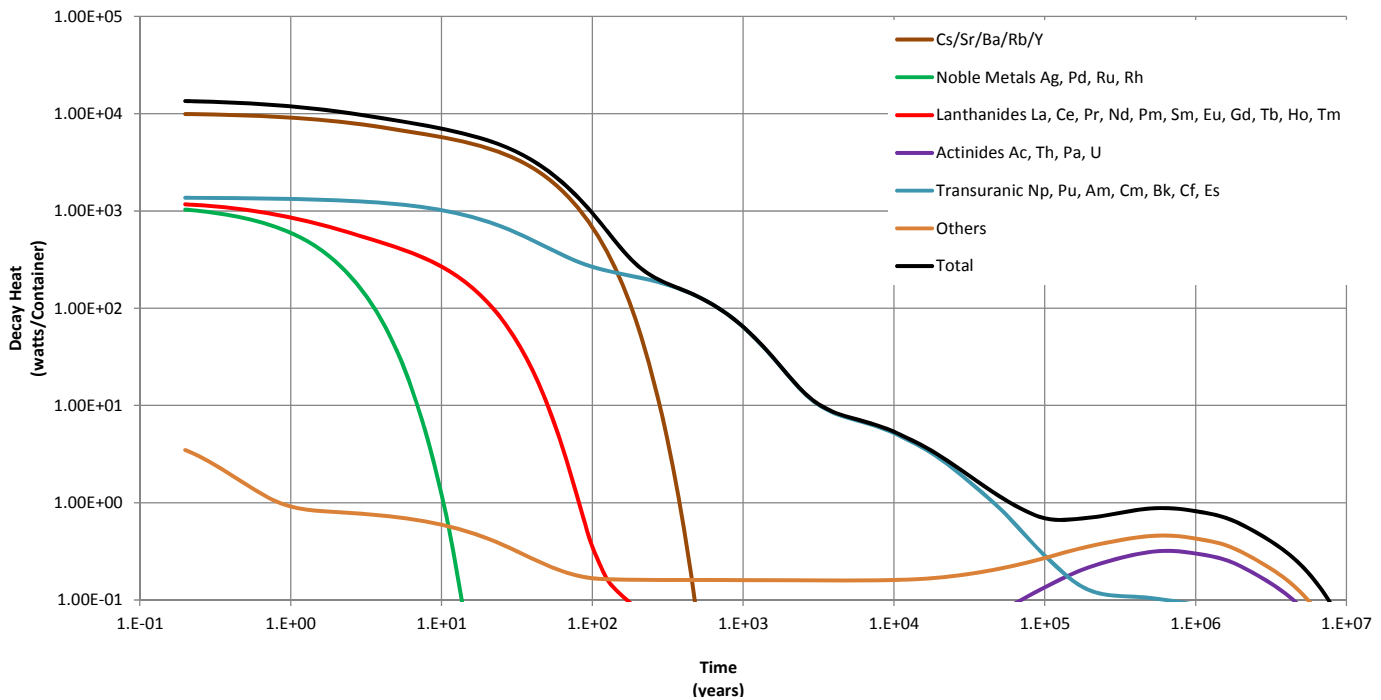
Figure 4-1 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 51 GWd/MT 5 Year Cooled PWR Fuel

4.2.2   Secondary Waste

Secondary wastes are those waste generated by the act of reprocessing. Secondary waste includes job control and maintenance materials contaminated with radioactivity. Estimates of the volume of secondary waste resulting from Co-Extraction reprocessing are available from several sources. AREVA has evaluated aqueous reprocessing facilities for construction in the United States and published waste generation estimates for a Co-Extraction facility with a capacity of 800 MTHM/year [Ref 18]. The Engineering Alternative Studies (EAS) for Separations provided a forecast of secondary waste generated from reprocessing facilities utilizing the UREX+1a process with capacities of 800 MTHM/year and 100 MTHM/year [Ref 19]. The method used to develop the EAS waste estimates allows manipulation of the data to estimate waste from other process variants (such as Co-Extraction) by the elimination (or duplication) of appropriate process functions. The EAS data is used in this manner to provide additional estimates of waste from a Co-Extraction reprocessing facility.

The volume of Class A, B and C LLW estimated by AREVA is significantly less than that estimated by using the EAS data. Waste estimates based on the EAS data for other reprocessing technologies (e.g. new extraction) are in closer agreement to the industry estimates (See Section 4.3.2). Because of the apparent agreement of the EAS data with other sources, the waste estimates for Class A, B and C LLW from Co-Extraction based on the EAS data are currently being used to provide a more conservative planning basis for other studies. Plans are being made to work with industry partners to understand and resolve the discrepancies in the Class A, B and C LLW estimates in order to provide more confidence in the estimates.

Estimates of GTCC waste based on the EAS data are in closer agreement with the AREVA data. Mixed wastes (waste defined as hazardous by the Resource Conservation and Recovery Act which also contain radioactive materials) are not estimated by AREVA, although AREVA has acknowledged the generation of mixed waste in their published data. The EAS data provides estimates for mixed waste; therefore, this data is used as a basis for mixed Class A, B and C LLW and mixed GTCC waste. This data was generated assuming today's commercial packaging (mainly compaction) practices. The study did not include future potential treatment alternatives.

Figures 4-2, 4-3 and 4-4 show the volume of Class A, B and C LLW, GTCC waste, mixed Class A, B and C LLW and mixed GTCC waste expected from a reprocessing facility using the Co-Extraction process. It should be noted that the data presented in the figures is based on specific facility capacities. For instance, 1,500 MTHM of USNF could be processed per year in a single facility with a capacity of 1,500 MTHM/year or in two separate facilities, each with a capacity of 750 MTHM/year. The waste volumes generated from these two scenarios will be different since the baseload maintenance activities for the facilities are similar but the unit volumes are reduced for the higher capacity facilities. Using Figure 4-2, a single facility (1,500 MTHM/year) will generate about 6 m^3 of Class A, B and C LLW per MTHM processed for an annual total of 9,000 m^3. The two facility Co-Extraction scenario (750 MTHM/year each) will generate about 9.7 m^3 of Class A, B and C LLW per MTHM processed for an annual total of 14,550 m^3 or 1.6 times the single facility scenario. A three facility Co-Extraction scenario (500 MTHM/year each) will generate about 12.8 m^3 of LLW per MTHM processed for an annual total of 19,200 m^3 or 2.1 times the single facility scenario. The volume of mixed GTCC waste is very low and estimated to be a constant 0.06 m^3/MTHM vs. capacity.

Figure 4-2 Class A, B and C LLW From A Co-Extraction Reprocessing Facility

Used Fuel Disposition U.S. Radioactive Waste Inventory and Characteristics
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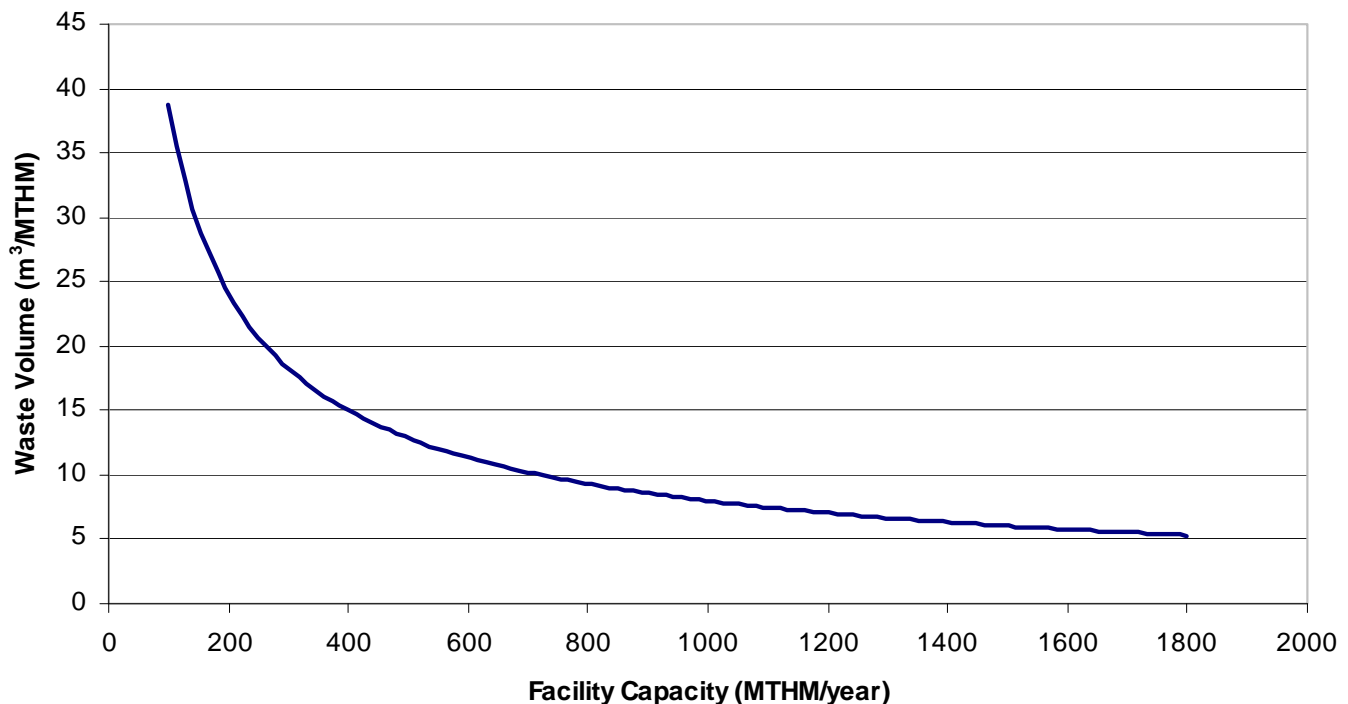16   May 2011
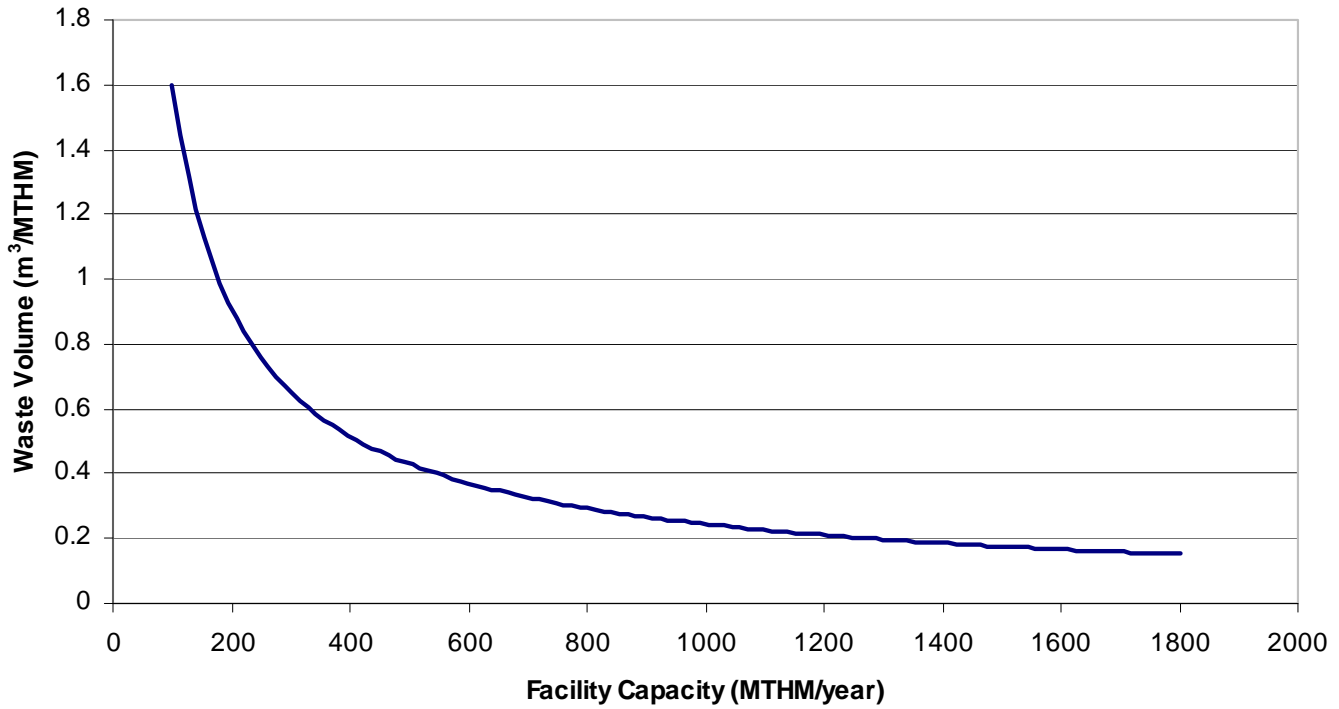
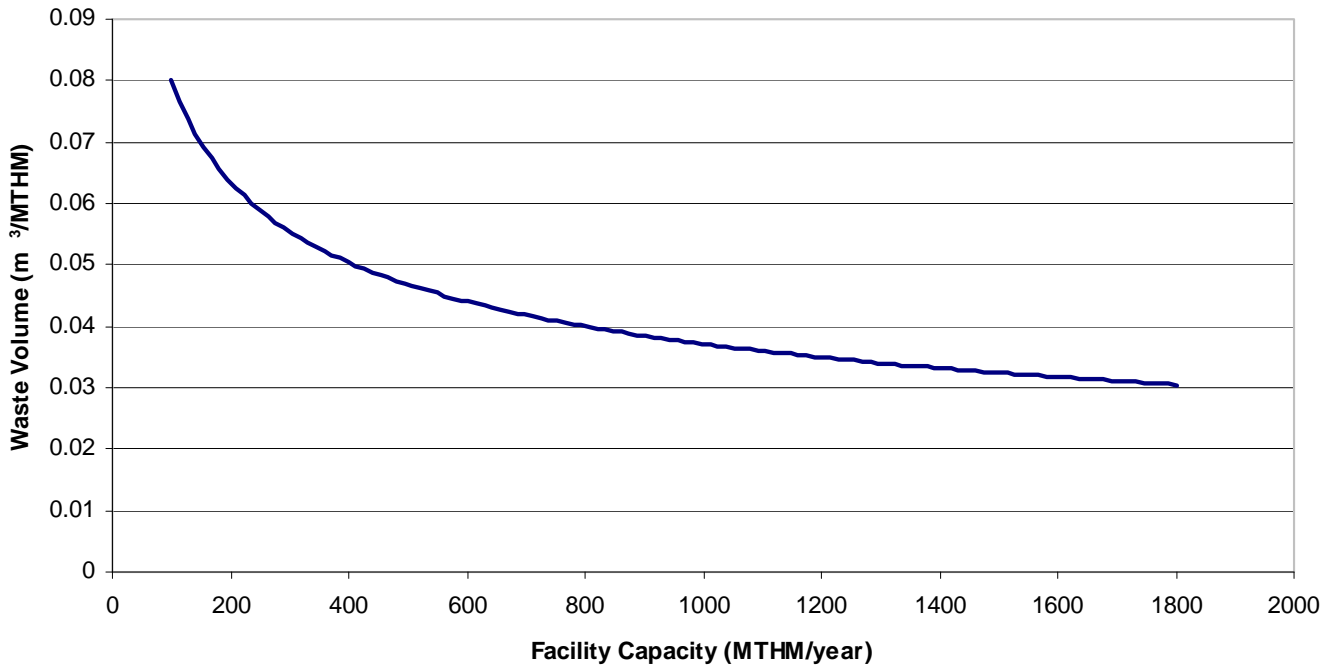Figure 4-3 Greater Than Class C Waste from a Co-Extraction Reprocessing Facility

Figure 4-4 Mixed Class A, B and C LLW From A Co-Extraction Reprocessing Facility

4.3   Advanced Aqueous Processes

Numerous advanced reprocessing techniques have been previously studied. Most of these involve recovery of transuranic (TRU) elements (e.g. Pu, Np, Am and Cm). None of these processes are practiced commercially today.

The Uranium Extraction (UREX) process was previously studied by DOE for TRU Recovery as well as separation of the waste stream into multiple components. Multiple waste forms were a focus of this concept to split the short term heat generating (Cs/Sr/Ba/Rb) waste from longer term fission product waste, and producing a more durable waste form for Tc by combining the UDS with a portion of the assembly hardware (Zr and stainless steel) in a metal alloy. The UREX process is discussed in Reference 1 but is not included in this summary since development efforts have been curtailed.

New extraction is an advanced aqueous process which recovers all of the TRU elements for re-use. The process envisioned includes Transuranic Extraction (TRUEX) and the Trivalent Actinide Lanthanide Separation by Phosphorus-based Aqueous Komplexes [*sic.*] (TALSPEAK) process for actinides separation from lanthanides. The principle fission product wastes are combined with the UDS and separated Tc into a single borosilicate glass wasteform (Cs and Sr are not separated as in the UREX process).

The principle gaseous radionuclides I-129, Kr-85, C-14 and H-3 released during reprocessing are captured and converted to waste forms suitable for disposal.

While this process is similar in function to the industrial process proposed by Energy *Solutions*, the two processes assume different processing methods and steps and so the product and waste streams cannot be directly compared.

The recovered TRU materials are assumed to be converted into sodium cooled fast reactor fuel which is subsequently recycled as discussed in Section 6.

4.3.1   Wastes from New Extraction Reprocessing

The potential waste from reprocessing the 50 GWd/MT LWR fuel cooled for 5 years are provided in Table 4-4, and 4-5. The fuel was selected to correspond with the parameters associated with prior SFR fuel studies described in Section 6. Additional information in Reference 1 includes the waste estimates generated by reprocessing various fuel burn-ups and age at the time of reprocessing.

Table 4-6 provides the mass, volume and containers of uranium recovered. This material is not generally considered a waste and is planned to be held as a resource for future reactor fuels. The data on uranium recovery is provided here for completeness.

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Table 4-4 New Extraction Fuel Reprocessing Off-gas Waste Summary Including Waste Loading Assumptions

	Captured Tritium Grouted				
	Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kilogram of cured grout				
Burn-up (GWd/MT)	Mass (kilogram/ MT)	Volume (m³/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions
50	2.10	2.55E-3	0.09	0.18	Recovered tritiated water is solidified as 30 wt% tritiated water in grout
	Captured Kr in High Pressure Cylinders				
	Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure.				
Burn-up (GWd/MT)	Mass (kilogram/ MT)	Volume (m³/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions
50	0.70	0.11	0.085	170	Separated by cryogenic methods and stored in high pressure type A gas cylinders. The Gas is approximately 98% Kr and 2% Xe.
	Captured I on Silver Mordenite Grouted				
	Containers: 55 gallon drum. Each drum contains 460 kilogram of cured grout (decay heat is <0.5 Watts/Canister)				
Burn-up (GWd/MT)	Mass (kilogram/ MT)	Volume (m³/MT)	Containers per MT		Waste Loading Assumptions
50	11.74	5.38E-3	0.03		Grouted Silver mordenite Iodine is loaded to 170mg per gram of mordenite which is then 25 wt% of the final grouted waste form
	Captured C-14 as Carbonate Grouted				
	Containers: 55 gallon drum. Each drum contains 460 kilogram of cured grout (decay heat is <0.5 Watts/Canister)				
Burn-up (GWd/MT)	Mass (kilogram/ MT)	Volume (m³/MT)	Containers per MT		Waste Loading Assumptions
50	9.41	4.25E-3	0.02		Converted to a carbonate and solidified as 30 wt% carbonate in grout

Table 4-5 New Extraction Fuel Reprocessing Metal and Fission Product Waste Summary Including Waste Loading Assumptions

	Compacted Metal			**Borosilicate Glass**			
	Containers: 2 ft. diameter x 10 ft. tall canisters. Each Canister Contains 3,600 kilogram. (decay heat is <0.5 Watts/Canister)			Containers: 2 ft. diameter x 15 ft. tall canisters. Each Canister Contains 2,900 kilogram.			
Burn-up (GWd/MT)	Mass (kilogram/MT)	Volume (m³/MT)	Containers per MT	Mass (kilogram/MT)	Volume (m³/MT)	Containers per MT	Decay Heat (W/container)
50	300.5	7.42E-2	0.084	309.2	0.142	0.11	14,000[a]
a To avoid multi-phase glass formation a decay heat limit of 14,000 watts per canister has been established. Specific radionuclides limits are determined based on the isotopic composition of the waste which varies by separation methodology. Waste loading is approximately 15.3% fission products as limited by decay heat							

Table 4-6 New Extraction Fuel Reprocessing Recovered Uranium Summary

	Recovered Uranium (U_2O_3)		
	Containers: 55 gal Drum canisters. Each Canister Contains 400 kilogram. (decay heat is <0.5 Watts/Canister)		
Burn-up (GWd/MT)	Mass (kilogram/MT)	Volume (m³/MT)	Containers per MT
50	1,094	0.57	2.74

4.3.1.1   New Extraction Borosilicate Glass Characteristics

The isotopic composition, for the principle heat generating waste from the new extraction process, the borosilicate glass was decayed using the ORIGEN 2.2 methods and isotopic parameters. Figure 4-5 provides the decay heat characteristics as a function of time for the new extraction process borosilicate glass.
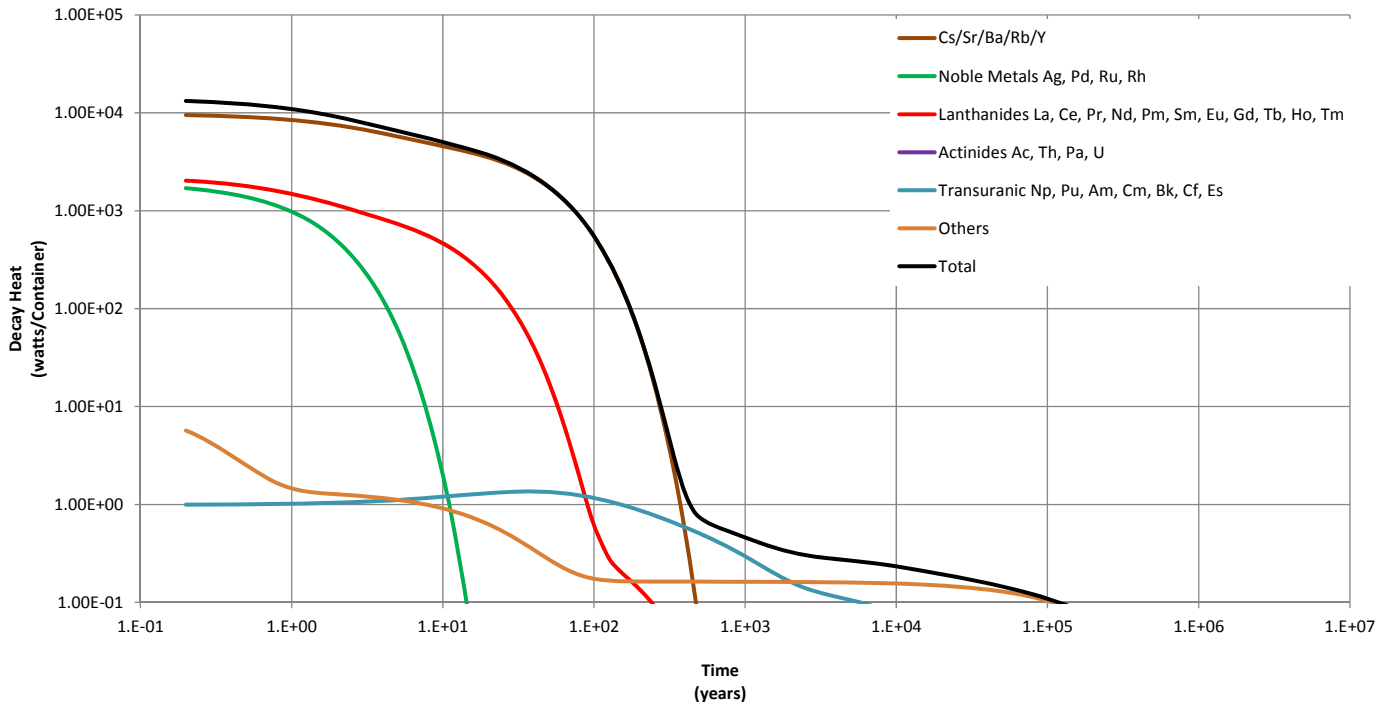

Figure 4-5 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 51 GWd/MT 5 Year Cooled PWR Fuel

4.3.2   Secondary Waste

Estimates of the volume of secondary waste resulting from new extraction reprocessing are available from several sources. Energy *Solutions* has evaluated aqueous reprocessing facilities for construction in the United States and published waste generation estimates for a new extraction facility with a capacity of 1,500 MTHM/year [Ref 20]. The EAS for Separations provided a forecast of secondary waste generated from reprocessing facilities utilizing the UREX+1a process with capacities of 800 MTHM/year and 100 MTHM/year [Ref 19]. The method used to develop the EAS waste estimates allows manipulation of the data to estimate waste from other process variants (such as New-Extraction) by the elimination (or duplication) of appropriate process functions. The EAS data is used in this manner to provide additional estimates of waste from a new extraction reprocessing facility.

Estimates of Class A, B and C LLW and GTCC waste based on the EAS data are in close agreement with the Energy *Solutions* data. Mixed wastes are not estimated by Energy *Solutions*, although Energy *Solutions* has acknowledged the generation of mixed waste in their published data. The EAS data provides estimates for mixed waste; therefore, this data is used as a basis for mixed Class A, B and C LLW and mixed GTCC waste. This data was generated assuming today's commercial packaging (mainly compaction) practices. The study did not include future potential treatment alternatives.

Figures 4-6, 4-7, and 4-8 show the volume of Class A, B and C LLW, GTCC waste, mixed Class A, B and C LLW and mixed GTCC waste expected from a reprocessing facility using the new extraction process. It should be noted that the data presented in the figures is based on specific facility capacities. For instance, 1,500 MTHM of USNF could be processed per year in a single facility with a capacity of 1,500 MTHM/year or in two separate facilities, each with a capacity of 750 MTHM/year. The waste volumes generated from these two scenarios will be different. Using Figure 4-7,

a single facility (1,500 MTHM/year) will generate about 7.6 m^3 of Class A, B and C LLW per MTHM processed for an annual total of 11,400 m^3. The two facility Co-Extraction scenario (750 MTHM/year each) will generate about 11.5 m^3 of Class A, B and C LLW per MTHM processed for an annual total of 17,250 m^3 or 1.5 times the single facility scenario. A three facility Co-Extraction scenario (500 MTHM/year each) will generate about 14.6 m^3 of LLW per MTHM processed for an annual total of 21,900 m^3 or 1.9 times the single facility scenario. The volume of mixed GTCC waste is very low and estimated to be a constant 0.06 m^3/MTHM vs. capacity.
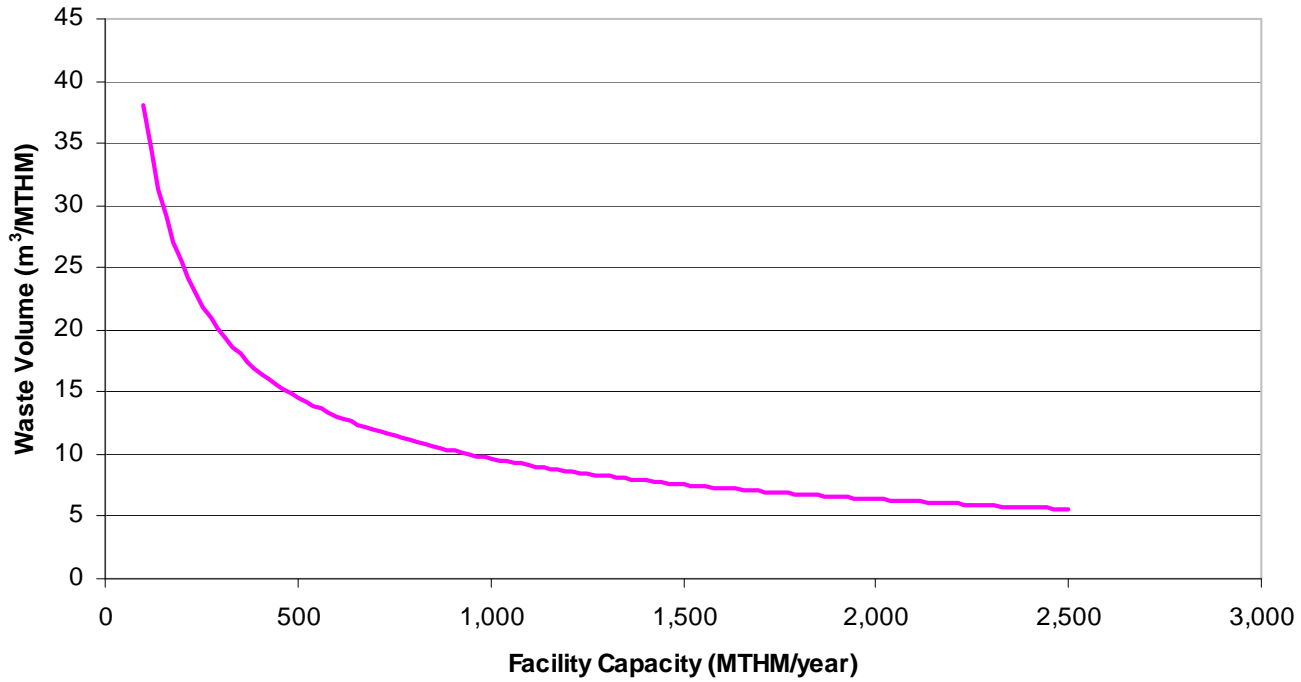

Figure 4-6 Class A, B and C LLW from a New Extraction Reprocessing Facility
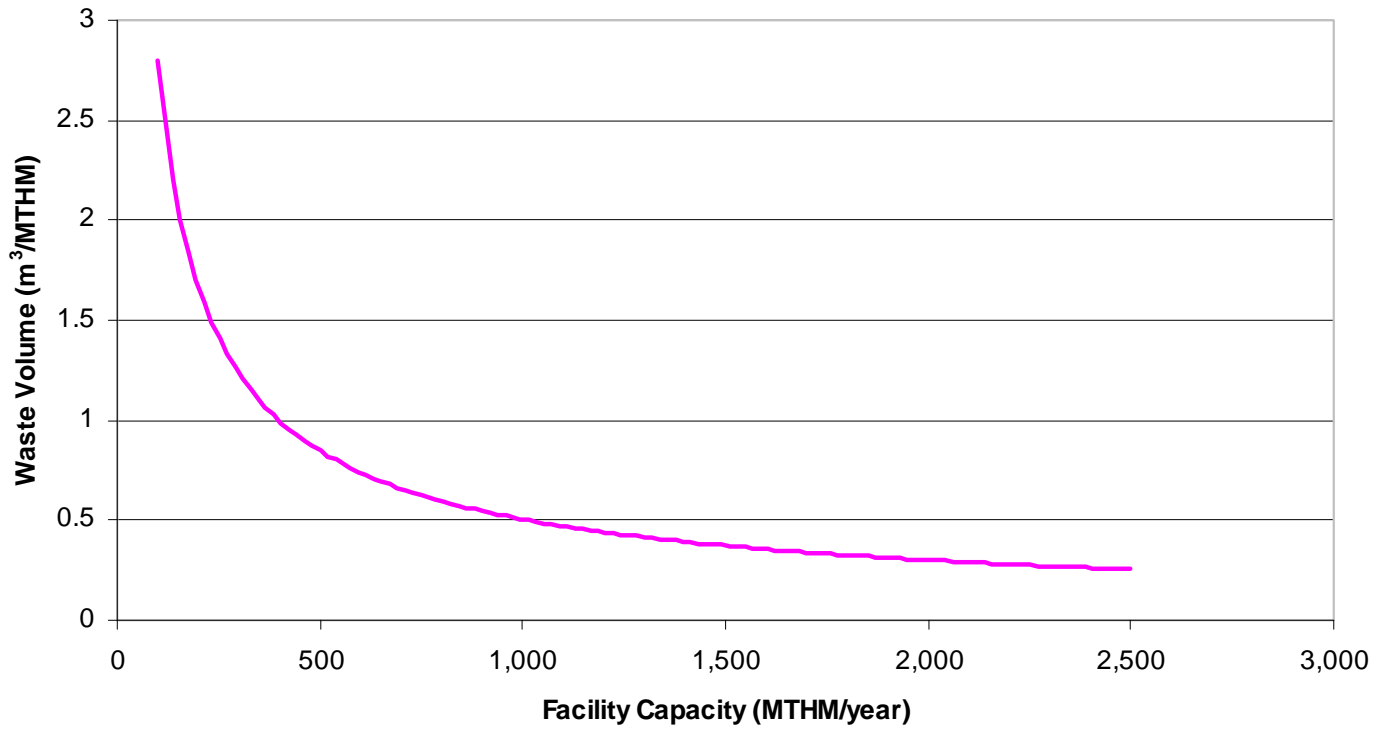
Figure 4-7 Greater Than Class C Waste from a New Extraction Reprocessing Facility
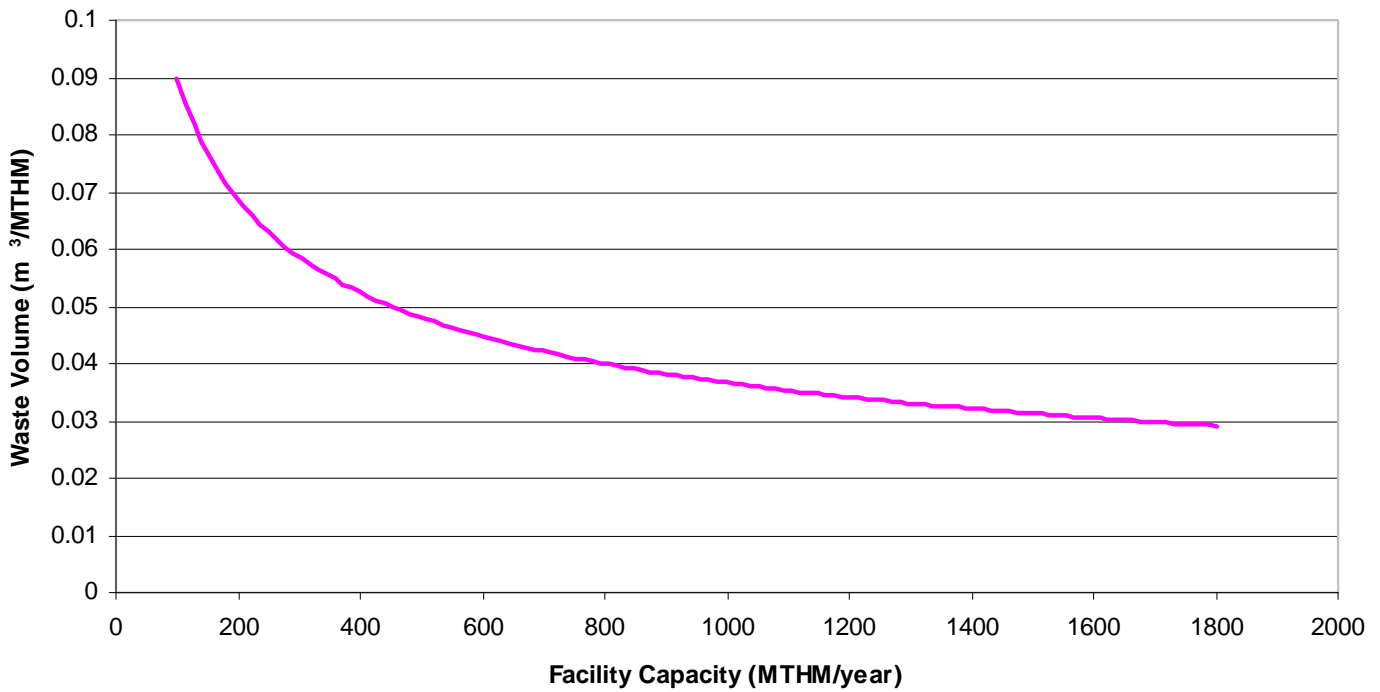
Figure 4-8 Mixed Class A, B and C LLW from a New Extraction Reprocessing Facility

5.   LWR MOX Fuel

The Fuel Cycle Research and Development Program has previously studied various MOX fuel alternatives [Ref 21]. Specifically they studied the scenario in which LWR UOX used nuclear fuel is burned to 51 GWd/MT and allowed to cool for 5 years post-irradiation and is then partitioned to separate the plutonium from the minor actinides, the other heavy metal nuclides, and the fission products. Because the Co-Extraction partitioning strategy is assumed, the spent fuel uranium in the LWR UOX USNF is assumed to be the uranium base of the MOX fuel (instead of natural or depleted uranium). It is assumed that 2 years are required to manufacture the recovered U/Pu into MOX fuel prior to introduction into the full MOX core. The delay time results in the build-up of Am-241 in the MOX fuel, which arises from the decay of Pu-241.

The full MOX fuel core is subsequently burned to an average value of 50 GWd/MT. The burn-up of the MOX core is limited to 50 GWd/MT because of a constraint on the plutonium content in the MOX fuel. Previous studies in Reference 22 have shown that plutonium content less than 12% (Pu in heavy metal) is necessary to ensure a negative void coefficient in a full MOX core; the specific value is actually plutonium isotopic vector dependent, but that dependence was not investigated in Reference 21.

5.1   Characteristics of typical LWR MOX fuel

The average plutonium enrichment is 10.74%. Therefore, each metric ton of LWR fuel which is reprocessed allows fabrication of 108.9 kilogram of MOX fuel.

The isotopic composition of discharged MOX fuel was obtained from the transmutation library maintained by the Systems Analysis Working Group. This discharge composition was decayed using the methods and isotopic parameters in ORIGEN 2.2. Figure 5-1 provides the decay heat of the MOX fuel as a function of time.
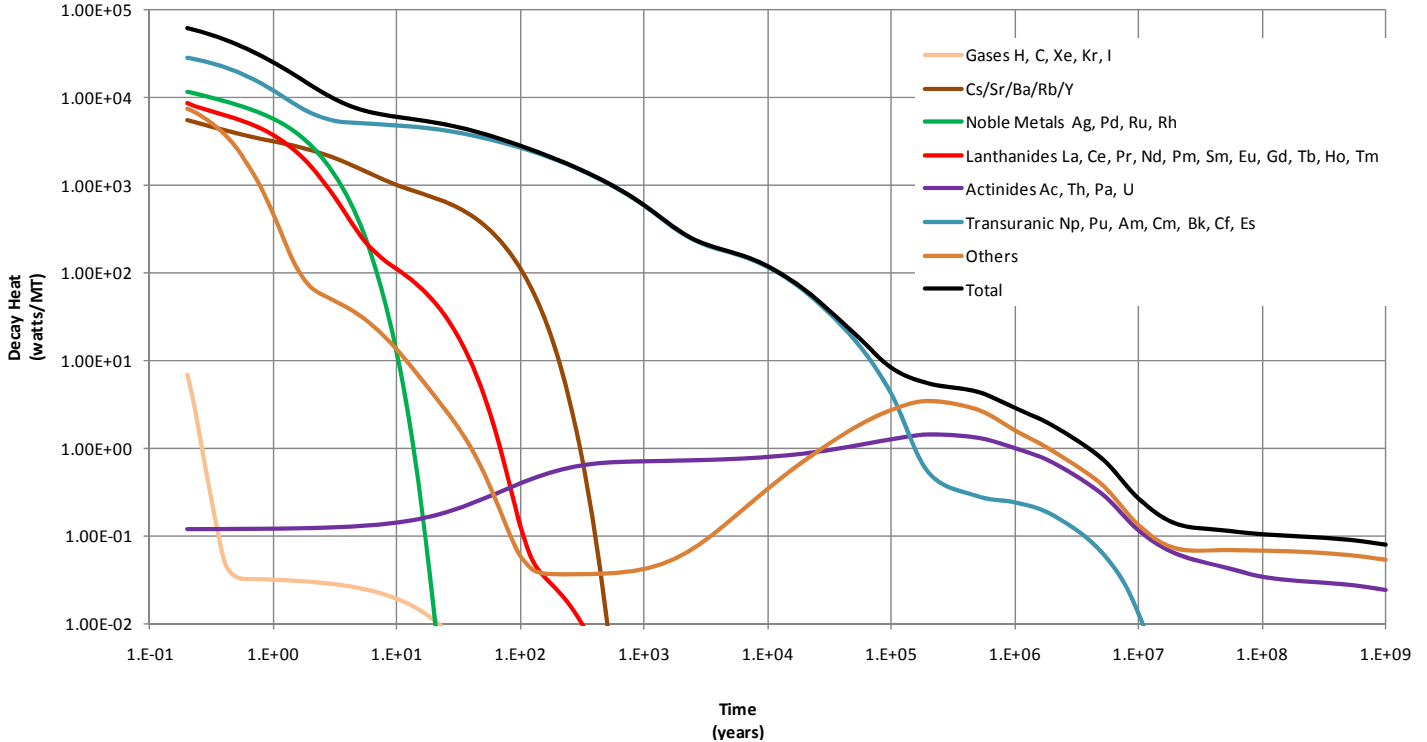
Figure 5-1 Mixed Oxide Fuel 50 GWd/MT Used Fuel Decay Heat

5.2   Wastes from LWR MOX Fuel Fabrication

Estimates of the volume of secondary waste resulting from the fabrication of MOX fuel from plutonium recovered from LWR USNF are based on the MOX Fuel Fabrication Facility under construction at the Savannah River Site [Ref 23]. The volume of Class A, B and C secondary waste from MOX fuel fabrication is dependent on the facility capacity. The volume of GTCC secondary waste is dependent on the facility capacity as well as the quantity and isotopic content of the plutonium used to fabricate the fuel which is in turn dependent on the burn-up and cooling time of the USNF from which it is obtained. This study assumed the waste is contaminated with the same quantity (mass) of Pu per unit waste volume as expected from the SRS MOX Fuel Fabrication Facility. The specific activity of Pu recovered from LWR USNF is much greater than the specific activity of the weapons grade Pu to be processed in the SRS MOX Fuel fabrication Facility. The increased specific activity will cause many of the waste streams to exceed the GTCC threshold for TRU content. The increased specific activity will also cause many of the resulting GTCC waste streams to exceed packaging limits for disposal (in the absence of a disposal facility for GTCC waste, this study assumed the Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria (WAC) as a reasonable basis for transportation and packaging of GTCC waste). The waste estimates are adjusted for the impact of the increased specific activity. Figure 5-2 shows the amount of Class A, B and C LLW estimated from MOX fuel fabrication relative to the plutonium throughput of the facility LLW. Figure 5-3 shows the amount of GTCC waste estimated from MOX fuel fabrication relative to the plutonium throughput of the facility for a range of USNF burn-ups and cooling times.

Figure 5-2 Annual LLW Volume Relative to Facility Capacity
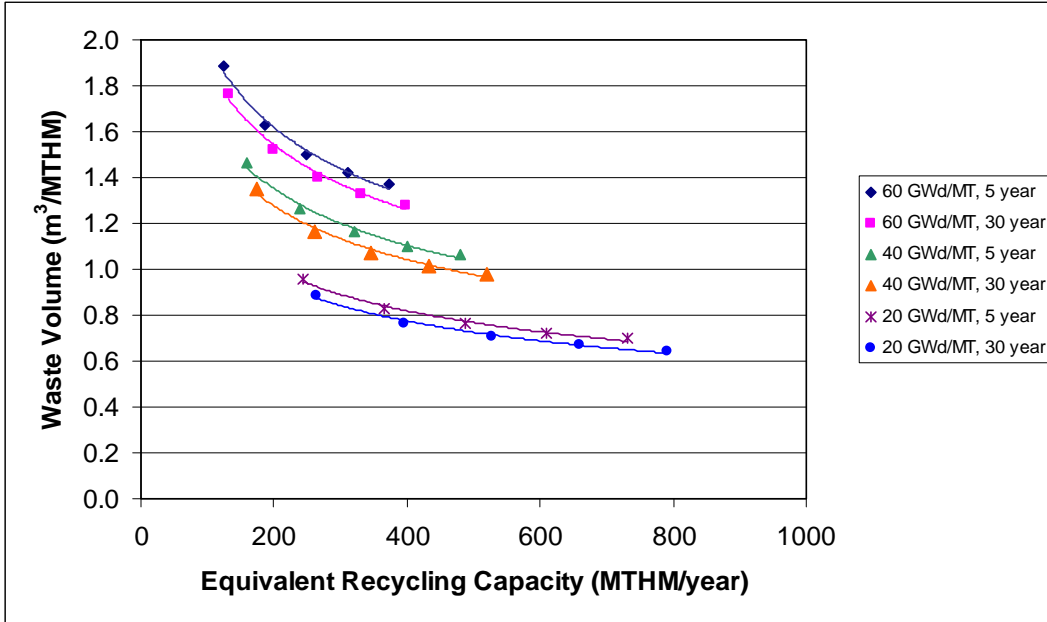
Figure 5-3 Annual Greater Than Class C Waste Volume Relative to Facility Capacity

6.   Advanced Fully Closed Fuel Cycle

A key attribute of the "fully closed" nuclear fuel cycle is that no USNF is disposed, only USNF reprocessing wastes are disposed.

6.1   Advanced Burner Reactor Fuels

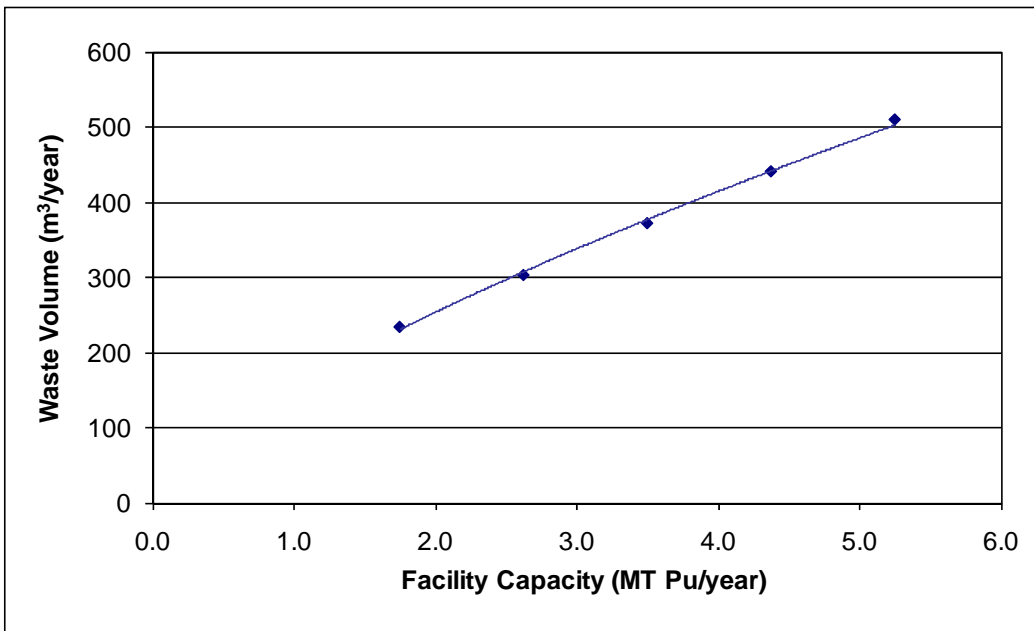Advanced Burner Reactor (ABR) systems have been previously studied with the majority of such studies utilizing fast spectrum reactors. These prior studies include numerous variations related to start-up core composition, equilibrium core design and operating parameters to provide various TRU Conversion Ratio (CR), fuel type (metal, oxide or carbon based) and reactor coolant (typically sodium, lead mixtures or gases).

Reference 1 investigated the waste generated by reprocessing oxide and metal fuel from reactors operated specified to result in TRU CRs of either 0.5 or 0.75 and found that the decay heat properties of the wastes were essentially the same for either fuel [Reference 1 Figure 6-2].

To provide an example of an alternative (to aqueous) reprocessing method, this study will assume the metal based sodium cooled reactor fuel with reactor operating parameters such that the CR is 0.75. The fuel will be reprocessed by an electro-chemical method. The metal fuel electro-chemical reprocessing method association is not technically mandated. The association is used in this summary to examine the wastes from an electro-chemical process.

6.1.1   Overall Mass Flows for a Closed Fuel Cycle

Advanced Burner Reactor core designs have been investigated and documented in References 24, 25 and 26. These studies include the basic design and operating parameters for a 1000 MWt SFR using U-TRU-Zr metal alloy fuel. Figure 6-1 provides the decay heat for both metal and oxide fuels which are all similar. The reactor parameters and the USNF isotopic data were combined to generate an overall reactor, fuel recycling, and fuel fabrication material balance for the reactor configuration. The material balances are documented in Reference 1 Appendix D Tables D-1 to D-4 and is summarized in Table 6-1. Since the reactors operate with a TRU conversion rate of less than 1.0, additional uranium and TRU must be supplied to the reactor system each year. The TRU source from the reference documents is LWR UOX fuel with a burn-up of 50 GWd/MT cooled for 5 years. Both the TRU quantity and quantity of LWR fuel which must be

reprocessed annually is provided in Table 6-1. The uranium source is the recovered uranium and additional depleted uranium.
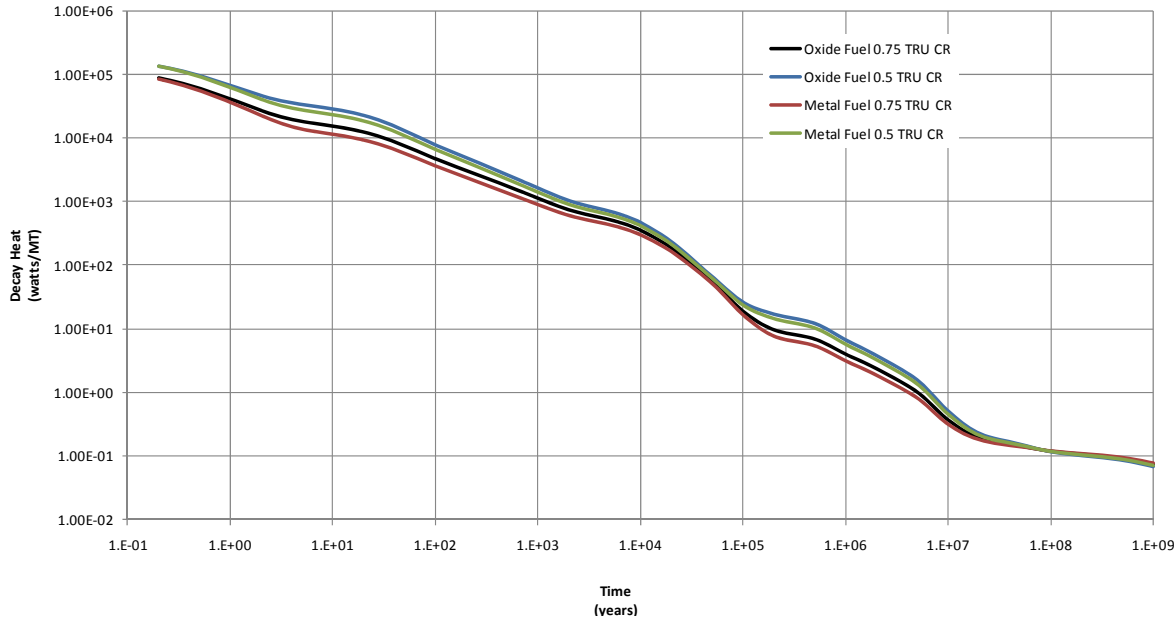
Figure 6-1 Sodium Fast Reactors Used Fuel Decay Heat

Table 6-1 Overall Reactor Material Balance Result

Reactor Parameters	Metal Fuel Core 1000 MWt CR = 0.75
Initial Core Charge (HM/TRU/Zr, MT)	14.07 / 2.98 / 1.57
Annual Fuel Requirements (HM/TRU/Zr,MT)	3.55 / 0.75 / 0.25
Annual LWR to Supply TRU (MT/yr)	5.78

6.2   Wastes from SFR Fuel Fabrication

At this time, secondary waste from the fabrication of SFR fuel using transuranic radionuclides from USNF has not been estimated by the Fuel Cycle Technologies program. It is anticipated that the waste volume would be on the order of that estimated for MOX fuel fabrication using plutonium recovered from LWR USNF (see Section 5.2); however, the waste volume from SFR fuel fabrication would be expected to be slightly higher though given the expected higher activity level of the feedstock containing additional radionuclides (e.g., minor actinides).

6.3   Reprocessing SFR Fuels

The electro-chemical process is a dry process using conductive molten salt baths to recover all the TRU elements. While electro-chemical recovery of uranium has been conducted at INL, the SFR fuel process envisioned is more complex and many aspects are under development and therefore uncertain. As envisioned the fission products are split between three waste streams. Elements which are more noble (as measured by electro-chemical potential) than uranium, such as fuel cladding and noble metal fission products, remain as metals and are incorporated into a metal alloy waste form. Elements less noble than uranium are converted to chloride salts. The lanthanide elements are recovered from the salt by electrolysis and converted to a lanthanide glass. Excess salt is purged; the chloride salt containing fission product chlorides is adsorbed by zeolite and bonded with glass to make the final waste form.

The principle gaseous radionuclides I-129, Kr-85, C-14 and H-3 released during reprocessing are captured and converted to a waste form suitable for disposal, although most of the I-129 in this process is not released to the gaseous phase but is converted to a molten salt and purged with the excess salt. In some studies, the lanthanide waste is not separated from the excess salt but is included in the glass bonded zeolite waste form which increases in decay heat, mass and volume. This change has not been assessed in reference 1. The volume of the glass bonded zeolite is also impacted by assumptions regarding the treatment of the chlorine released during U/TRU recovery. This study assumes the Cl is captured on steel wool as FeCl and included in the glass bonded zeolite waste form. The Cl waste stream accounts for about 45% of the mass and volume of the waste.

6.3.1 Electro-Chemical Reprocessing Waste

The potential waste from reprocessing the metal SFR fuels are provided in Table 6-2, 6-3 and 6-4.

Table 6-2 Sodium Fast Reactor Metal Fuel Reprocessing Off-gas Waste Summary Including Waste Loading Assumptions

Burn-up (GWd/MT)	Conversion Ratio	**Captured Tritium Grouted**				
		Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kilogram of cured grout				
		Mass (kilogram/MT)	Volume (m^3/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions
99.6	0.75	2.10	2.55E-3	0.09	0.65	Recovered tritiated water is solidified as 30 wt% tritiated water in grout
Burn-up (GWd/MT)	Conversion Ratio	**Captured Kr in High Pressure Cylinders**				
		Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure.				
		Volume (ft^3/MT)	Volume (m^3/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions
99.6	0.75	0.93	0.138	0.112	201	Separated by cryogenic methods and stored in high pressure type A gas cylinders. The Gas is approximately 98% Kr and 2% Xe.
Burn-up (GWd/MT)	Conversion Ratio	**Captured I on Silver Mordenite Grouted**				
		Containers: 55 gallon drum. Each drum contains 460 kilogram of cured grout				
		Mass (kilogram/MT)	Volume (m^3/MT)	Containers per MT		Waste Loading Assumptions
99.6	0.75	0.00	0.00	0.000		Grouted Silver Mordenite Iodine is loaded to 170mg per gram of mordenite which is then 25 wt% of the final grouted waste form
Burn-up (GWd/MT)	Conversion Ratio	**Captured C-14 as Carbonate Grouted**				
		Containers: 55 gallon drum. Each drum contains 460 kilogram of cured grout				
		Mass (kilogram/MT)	Volume (m^3/MT)	Containers per MT		Waste Loading Assumptions
99.6	0.75	19.62	8.79E-3	0.043		Converted to a carbonate and solidified as 30 wt% carbonate in grout

Table 6-3 Sodium Fast Reactor Metal Fuel Reprocessing Metal Waste Summary

Burn-up (GWd/MT)	Conversion Ratio	**Electro-chemical**			
		Metal Alloy			
		Containers: 2 ft. diameter x 10 ft. tall canisters. Each Canister Contains 3,600 kilogram.			
		Mass (kilogram/MT)	Volume (m^3/MT)	Containers per MT	Decay Heat (W/container)
99.6	0.75	4,403	1.09	1.22	3,838

Table 6-4 Sodium Fast Reactor Metal Fuel Reprocessing Fission Product Waste Summary Including Waste Loading Assumptions

| Burn-up (GWd/MT) | Conversion Ratio | Glass Bonded Zeolite Containers: 2 ft. diameter x 15 ft. tall canisters. Each Canister Contains 2,900 kilogram. | | | | |
		Mass (kilogram/MT)	Volume (m^3/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions
99.6	0.75	2,641	1.21	0.91	2900	75% Zeolite
Burn-up (GWd/MT)	Conversion Ratio	Lanthanide Glass Containers: 6in diameter x 60in tall canisters. Each Canister Contains 500 kilogram.				
		Mass (kilogram/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Waste Loading Assumptions
99.6	0.75	58.39	1.30E-2	0.12	21,175	50 wt% lanthanides

The waste unit quantities are reported per MT of fuel recycled. However the repository system analyst will likely need to know the total quantities of waste to be disposed. In order to determine the total quantities several additional parameters will need to be considered. These include the thermal efficiency and overall utility of the power plant if such studies are related to net power generation. The total quantities must also include the waste generated from reprocessing the LWR fuel as discussed in Section 4.3.1.

6.3.2   Characteristics of the Heat Generating Wastes from SFR Processes

The isotopic composition, for the principle heat generating waste from the electro-chemical process, the glass bonded zeolite and the lanthanide glass was decayed using the ORIGEN 2.2 methods and isotopic parameters.

Figure 6-2 provides the decay heat characteristics as a function of time for the glass bonded zeolite, lanthanide glass and metal alloy waste form.
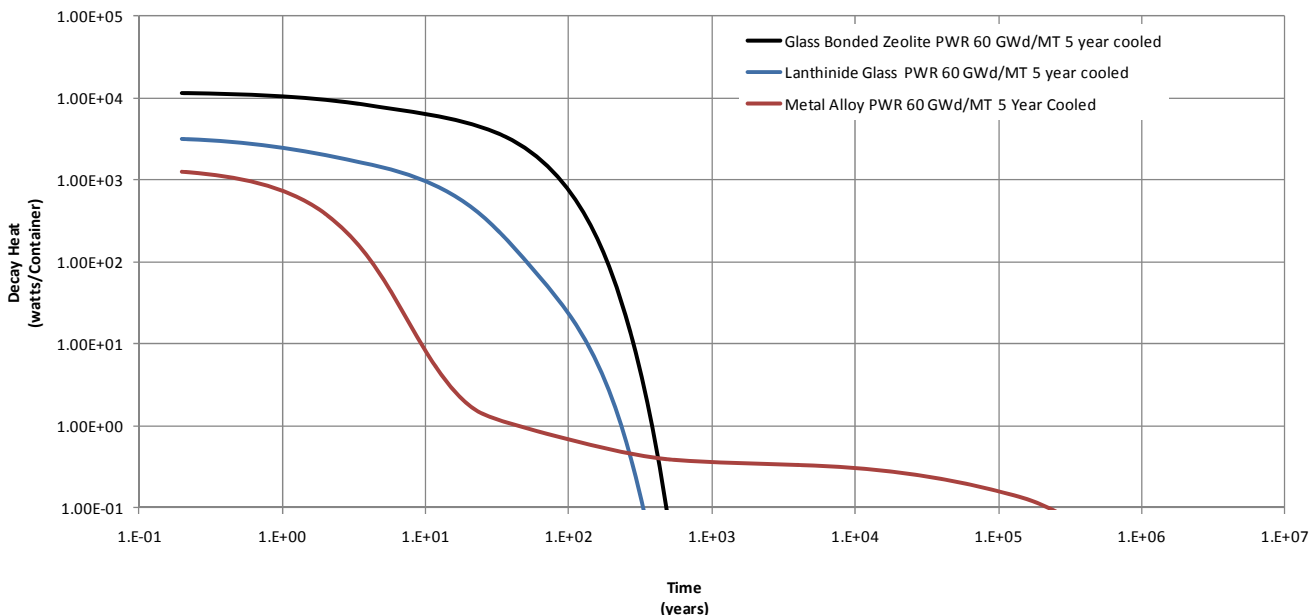

Figure 6-2 Electro-Chemical Process Waste Decay Heat Summary

6.3.3   Secondary Waste from SFR Fuel Reprocessing

Estimates of the volume of secondary waste resulting from electro-chemical reprocessing of LWR USNF was evaluated by the EAS for a reprocessing facility with a capacity of 300 MTHM/year.[Ref 27]. The estimates for LWR USNF reprocessing were adjusted to reflect reprocessing of SFR USNF.

The volume of Class A, B and C LLW from electro-chemical recycling of SFR USNF is expected to be similar to that generated from electro-chemical recycling of LWR USNF. Only a minor increase in waste volume driven by the design and configuration of the SFR fuel is expected. Increases in Class A, B and C LLW volume are driven by the increased amount of fuel receipts required for SFR USNF relative to LWR USNF for a given amount of heavy metal processed. The volume of GTCC waste, mixed Class A, B and C LLW and mixed GTCC waste from electro-chemical recycling of SFR USNF is not expected to be any different than that generated from electro-chemical recycling of LWR USNF. This data was generated assuming today's commercial packaging (mainly compaction) practices. The study did not include future potential treatment alternatives.

Table 6-5 provides a summary of the waste estimates for recycling of SFR USNF using an electro-chemical process for a facility capacity of 300 MTHM/year. Waste volume estimates are provided in terms of total annual waste generation (m^3/year) and a normalized value expressed as waste per MTHM processed.

Table 6-5 Annual Waste Volume for Electro-chemical Reprocessing of Sodium Fast Reactor Used/Spent Nuclear Fuel

Volume	Class A, B, and C LLW	GTCC Waste	Mixed Class A, B and C LLW	Mixed GTCC Waste
m^3/year	2,716	919	29	44
m^3/MTHM [1]	9.1	3.1	0.1	0.15

1.   Based on a facility capacity of 300 MTHM/year

7.   Comparison of LWR Recycle Waste Volumes with Bare Used Nuclear Fuel

This section provides a brief and simplistic comparison of the volume of nuclear waste generated from potential nuclear fuel cycles that would reprocess LWR fuel to the LWR fuel itself. A 17x17 PWR assembly is considered as the "basis" to which comparisons will be made. The dimensions are approximately 22cm x 22 cm x 4.07 m, giving a volume of 0.20 m^3. A 17x17 PWR fuel assembly contains approximately 0.46 MTHM, giving a "specific volume" of 0.43 m^3/MTHM. Table 7-1 shows the volume of each potential HLW form that would be generated from the recycling of PWR fuel with a burnup of 60 GWd/MT that has been cooled for 30 years after discharge from the reactor [Ref 1]. Note: the mass and volume of some waste forms in Table 7-1 differs from that reported in Tables 4-1 to 4-6 due to the use of different cooling periods.

These results indicate that there may be very little benefit in terms of volume reduction associated with recycling LWR fuel. While this may be true, it must be recognized that the characteristics of the wastes differ significantly from bare LWR fuel and from themselves in terms of radionuclide content and decay heat production.   These characteristics, in addition to volume, have a significant role in the management of these materials.

Table 7-1 Estimated Volume of Waste Forms from Recycling Including Waste Loading Assumptions

	Co-Extraction	New Extraction	Waste Form and Waste Loading Assumptions
I from Off Gas Recovery (m^3/MT)	0.0054	0.0054	Grouted silver mordenite Iodine is loaded to 170 mg per gram of mordenite which is then 25 wt% of the final grouted waste form
Tritium from Off Gas Recovery (m^3/MT)	0.0025	0.0025	Recovered tritiated water is solidified as 30 wt% tritiated water in grout
C-14 from Off Gas Recovery (m^3/MT)	0.0042	0.0042	Converted to a carbonate and solidified as 30 wt% carbonate in grout
Kr from Off Gas Recovery (m^3/MT)	0.12	0.12	Separated by cryogenic methods and stored in high pressure type A gas cylinders. The Gas is approximately 98% Kr and 2% Xe.
Hulls and Hardware (m^3/MT)	0.075	0.075	Decontaminated, compacted and placed inside a canister
Fission Product (m^3/MT)	0.18[a]	0.17[b]	Borosilicate glass. To avoid multi-phase glass formation a decay heat limit of 14,000 watts per canister has been established. Specific radionuclides limits are determined based on the isotopic composition of the waste which varies by separation methodology. [a] For the Co-extraction glass in this example, fission product waste loading is approximately 10.3 wt %. [b] For the New Extraction glass in this example, fission product waste loading is approximately 15.3 wt %.
Total (m^3/MT)	**0.39**	**0.38**	
Total Volume Change relative to PWR Assembly	91%	88%	
Total Volume Change relative to PWR Assembly, Fission Product Waste Only	42%	39%	

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While Table 7-1 provides estimated waste form volumes from a single recycling case, some general conclusions can be drawn from inventory estimates over a range of fuel burn-up and cooling time prior to recycling (see Ref 1, Tables 4-1, 4-2 and 4-3):

- The volume of the tritium, carbon-14, and hulls/hardware bearing waste forms is insensitive to fuel burn-up and cooling time;
- The volume of iodine bearing waste forms increases with burnup, but is insensitive to cooling time;
- The volume of krypton bearing waste forms increases with burnup and decreases with cooling time; and
- The volume of fission product bearing waste forms increases with burnup and decreases with initial cooling (generally less than 30 years). After the initial cooling period the mass and volume are constant based upon other waste loading limitations.

Only the heat generating waste forms require thermal management during storage, transportation, and disposal. The remaining waste streams (off-gas recovery, hulls and hardware) do not generate significant quantities of heat. In the case of the primary fission product waste forms:

- The waste loading of the heat-generating waste forms (and perhaps others) may be further constrained by thermal limits within a geologic disposal system. These limits will be specific to the geologic media and repository design concepts, and were not considered in estimating the volumes shown above.
- While the volume of the Co-Extraction and New Extraction fission product waste forms are virtually identical, the thermal output of the New Extraction waste form considered in this example (60GWd/MT 30 year old fuel) is almost a third of that of the Co-Extraction waste form at the time it is created due to the removal of americium and curium (Ref 1., Figures L-13 and L-14).
- This section considers only the material that is characterized as waste at the end of a single processing cycle. If the recovered uranium oxide were determined to be a waste the volume increase by about 0.55m^3/MT.

In addition to the primary waste streams, the recycling of used LWR fuel would also generate secondary waste from the act of reprocessing. Secondary wastes include Class A/B/C LLW, mixed Class A/b/C LLW, GTCC LLW, and mixed GTCC LLW. Generation rate estimates for these secondary wastes have been developed (based on current compaction practices) and are shown in Table 7.2 for an 800 MT/yr capacity aqueous separations plant.

Table 7-2 Secondary Waste from Recycling LWR USNF

	Co-Extraction	New Extraction
LLW (m^3/MT)	9.3	11.0
Mixed LLW (m^3/MT)	0.04	0.04
GTCC (m^3/MT)	0.29	0.60
Mixed GTCC (m^3/MT)	0.06	0.06

This section considers only the material that is characterized as waste at the end of a single processing cycle. If the recovered uranium oxide were determined to be a waste, the total volume would increase by about 0.55m^3/MT.

An overall summary of this brief and simplistic comparison of the volume of nuclear waste generated from potential nuclear fuel cycles that would reprocess LWR fuel is provided in Figure 7-1.

Future research and development on advanced waste forms could lead to higher waste loading densities and a reduction in the volume of wastes that would be generated from recycling. However, in the case of heat-generating waste forms, the benefit of increased waste loadings must be evaluated considering the effects of higher heat generation rates on thermal limits within geologic disposal systems.
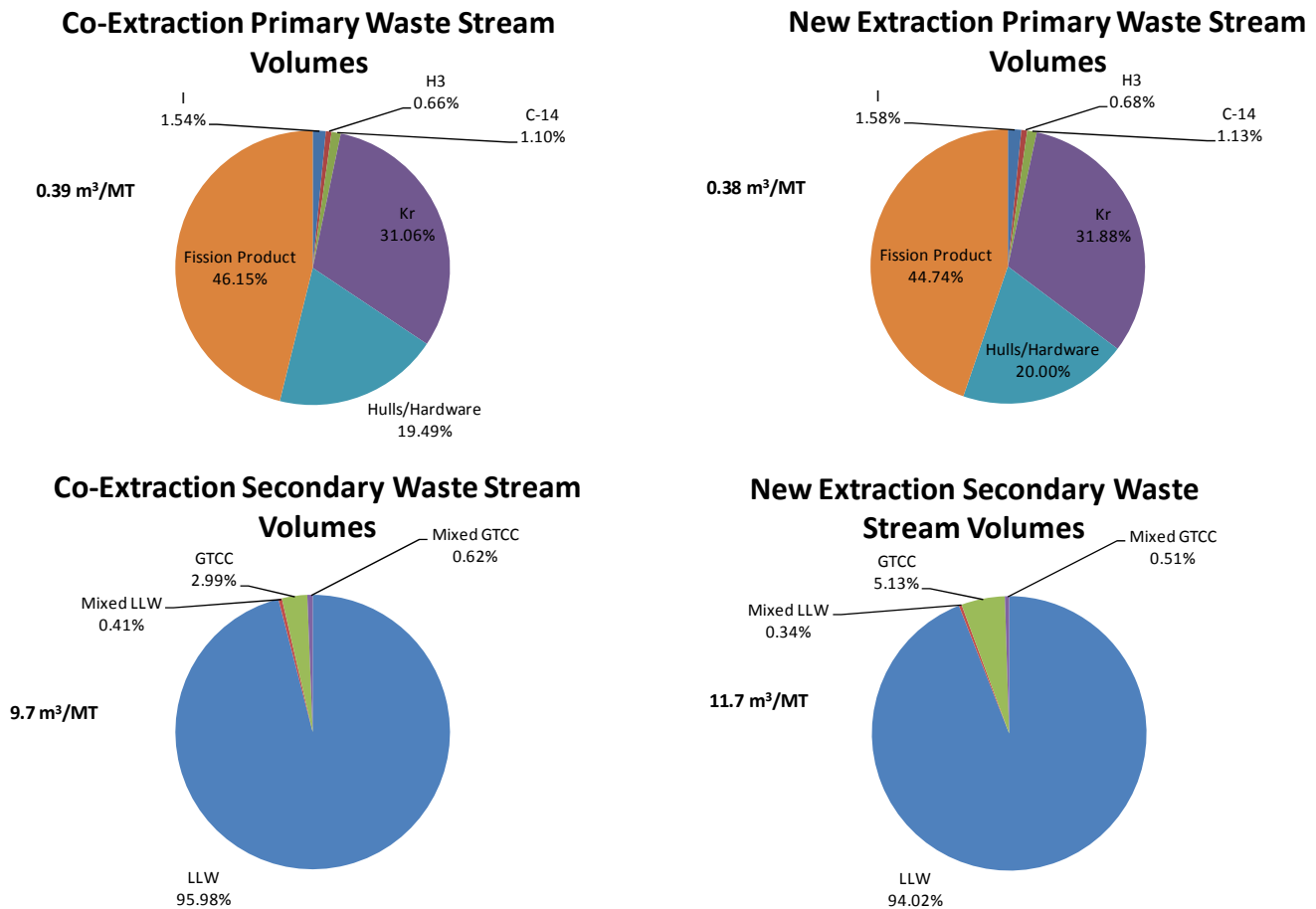
Co-Extraction Primary Waste Stream Volumes

0.39 m³/MT

- I 1.54%
- H3 0.66%
- C-14 1.10%
- Kr 31.06%
- Hulls/Hardware 19.49%
- Fission Product 46.15%

New Extraction Primary Waste Stream Volumes

0.38 m³/MT

- I 1.58%
- H3 0.68%
- C-14 1.13%
- Kr 31.88%
- Hulls/Hardware 20.00%
- Fission Product 44.74%

Co-Extraction Secondary Waste Stream Volumes

9.7 m³/MT

- Mixed GTCC 0.62%
- GTCC 2.99%
- Mixed LLW 0.41%
- LLW 95.98%

New Extraction Secondary Waste Stream Volumes

11.7 m³/MT

- Mixed GTCC 0.51%
- GTCC 5.13%
- Mixed LLW 0.34%
- LLW 94.02%

Figure 7-1 Summary of Waste Volumes from Recycling LWR Fuel

8.   GTCC Wastes and DOE Equivalent

GTCC Low-Level Radioactive Waste consists primarily of activated metals from the decommissioning of nuclear reactors, certain disused sealed radioactive sources, wastes from the production of medical isotopes, and small quantities of other radioactive wastes containing high concentrations of short-lived or long-lived radionuclides.

GTCC LLW is not generally acceptable for near-surface disposal, although there may be some circumstances where it might be acceptable for near surface disposal with special processing or design. The United States NRC regulations (10 Code of Federal Regulations Part 61.55) specify that in the absence of specific requirements, such waste must be disposed of in a geologic repository unless alternative methods for disposal of such waste are proposed to and approved by the NRC.

As the United States agency assigned to be responsible for developing a disposal capability for GTCC LLW, DOE is preparing an EIS [Ref 29]. that evaluates proposed disposal methods and locations for GTCC LLW and similar DOE waste. This EIS evaluates the impact of disposing the current and future inventory of GTCC waste resulting from the current fuel cycle. Wastes described in Sections 1 to 6 which may be classified as GTCC waste are in addition to those described in this section.

8.1   Types and Quantities of GTCC Waste

The GTCC waste inventory addressed in the EIS has a packaged volume of approximately 12,000 m^3 and contains a total activity of about 160 million curies (MCi). The inventory accounts for wastes in storage and wastes expected to be generated in the future. GTCC waste could be categorized into three types: activated metal, sealed sources, or other waste. Activated metals and sealed sources represent about 40 percent of the total inventory [Ref 28,29].

Activated metal is primarily generated by nuclear reactors during normal operations and becomes waste during facility decommissioning. The waste consists of components internal to the reactor that have become radioactive from exposure to a neutron flux. The bulk of the total activity in the activated metals initially is from the short-lived radionuclides cobalt-60 and iron-55, which do not have Class C limits as defined in 10 CFR 61.55. The high concentrations of the longer-lived radionuclides in the activated metal, such as Ni-63, Ni-59, and niobium-94, is responsible for the material resulting in GTCC LLW.

Sealed sources are commonly used to sterilize medical products, detect flaws and failures in pipelines and metal welds, and determine moisture content in soil and other materials (moisture gauges), and diagnose and treat illnesses such as cancer (teletherapy units). They can encompass several physical forms, including ceramic oxides, salts, or metals. Common radionuclides present in the sealed sources include cesium chloride, Pu-238, Pu-239, Pu-240, Am-241, Am-243, and Cm-244.

Other waste consists of a wide variety of materials, such as contaminated equipment, sludge, salts, charcoal, scrap metal, glove boxes, solidified solutions, particulate solids, filters, and organic and inorganic debris, including debris from future decontamination and decommissioning activities, the production of Pu-238 radioisotope power systems, and the production of medical isotopes (molybdenum-99). These wastes can come in a number of physical forms, and a wide range of radionuclides may be present.

9.   References

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