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GNEP Material Transportation, Storage and Disposal Analysis FY-08 Summary Report

W. Halsey

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GNEP Material Transportation, Storage and Disposal Analysis FY-08 Summary Report

Global Nuclear Energy Partnership

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This report is a summary of efforts from five national laboratories and one university as participants in the GNEP, AFCI R&D, Systems Analysis, Material Storage, Transportation & Disposal task area.

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SUMMARY

This report provides a summary for FY-2008 of activities, analyses and products from the Material Transportation, Storage and Disposal (M-TSD) sub-task of Systems Analysis within the Advanced Fuel Cycle Research & Development area of the Global Nuclear Energy Partnership. The objective of this work is to evaluate near-term material management requirements for initial GNEP facilities and activities, long-term requirements for large-scale GNEP technology deployment, and alternatives and paths forward to meet these needs. For FY-08, the work expanded to include the Integrated Waste Management Strategy as well as integration with the newly formed Waste Forms Campaign. The M-TSD team was expanded with the addition of support from Savannah River National Lab (SRNL) to the existing team of Lawrence Livermore National Lab (LLNL), Argonne National Lab (ANL), Idaho National Lab (INL), Sandia National Lab (SNL) and University of Nevada – Reno (UN-R).

During the first half of the year, analysis was focused on providing supporting technical analysis and documentation to support anticipated high-level decisions on program direction. A number of analyses were conducted and reports prepared as program deliverables. This work is briefly summarized in this report. Analyses provided informally to other program efforts are included in this report to provide documentation.

This year-end summary was planned primarily as a compilation of activities following the anticipated programmatic decisions. These decisions were deferred beyond the end of the year, and funds were reallocated in a number of areas, thus reducing the M-TSD activities. This report summarizes the miscellaneous 'ad-hoc' work conducted during the later part of the year, such as support to the draft Programmatic Environmental Impact Statement (PEIS), and support to other program studies.

Major programmatic contributions from the M-TSD team during the year included:

- Completion of the IWMS in March 2008 as the baseline for waste management calculations for the GNEP Programmatic Environmental Impact Statement (PEIS). The IWMS represents a collaborative effort between the Systems Analysis, Waste Forms, and Separations Campaigns with contributing authors from multiple laboratories. The IWMS reference is: "Global Nuclear Energy Partnership Integrated Waste Management Strategy, D. Gombert, INL, et al, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008"
- As input to the IWMS and support for program decisions, an evaluation of the current regulatory framework in the U.S. pertaining to the disposal of radioactive wastes under an advanced nuclear fuel cycle was completed by ANL. This evaluation also investigated potential disposal pathways for these wastes. The entire evaluation is provided in Appendix A of this report.
- Support was provided to the development of the GNEP Programmatic Environmental Impact Statement from INL, SNL and ANL M-TSD staff.
- M-TSD staff prepared input for DSARR (Dynamic Systems Analysis Report for Nuclear Fuel Recycle) report. The DSARR is an INL led report to examine the time-dependent dynamics for a transition from the current open fuel cycle to either a 1-tier or 2-tier closed fuel cycle. Section 5.3 Waste Management Impacts was provided to INL for incorporation into the DSARR.
- SNL M-TSD staff prepared a M2 milestone report "*Material Transportation, Storage and Disposal Contribution for Secretarial Decision Package".* The report purpose was to comprehensively evaluate and discuss packaging, storage, and transportation for all potential

nuclear and radioactive materials in the process and waste streams being considered by the GNEP program. In particular, a systems view was used to capture all packaging, storage, and transport operations needed to link the various functional aspects of the fuel cycle.

- SRNL M-TSD staff developed a deliverable report *"Management of Decay Heat from Spent Nuclear Fuel"*. This report evaluated a range of options for managing the near-term decay heat associated with Cs and Sr in spent nuclear fuel (SNF) reprocessing wastes.
- M-TSD staff participated in a series of meetings of the US-Japan GNEP Working Group on Waste Management, developing the content for the first deliverable of the working group.

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SYSTEMS ANALYSIS MATERIAL TRANSPORTATION, STORAGE & DISPOSAL

1 INTRODUCTION

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The remainder of this report is activity summaries by participant. Analyses that have not been documented elsewhere are included as Appendices of this report.

1.1 Material Transportation, Storage & Disposal Highlights: FY-08

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- M-TSD staff participated in a series of meetings of the US-Japan GNEP Working Group on Waste Management, developing the content for the first deliverable of the working group.

1.2 Material Transportation, Storage & Disposal Deliverables: FY-08

The M-TSD team completed 15 programmatic milestones on schedule during the year. Several planned deliverables were canceled or deferred due to funding re-allocation.

Table 1.2-1 List of Material Transportation, Storage & Disposal Deliverables for FY-08.

Milestone No.	Lab	Level	Title	Due Date)
M506040202	ANL	M4	ANL Input - Waste Management and Disposal Disposition Alternatives Evaluation	2/29/2008
M506040201	ANL	M3	ANL Input to GNEP Material Transportation Storage and Disposal Analysis FY-08 Summary Report (Disposition Alternatives)	9/12/2008
M506040603	ANL	M4	Repository Performance Input to IWMS	2/29/2008
M506040602	ANL	M4	ANL Input to GNEP Material Transportation Storage and Disposal Analysis FY-08 Summary Report (Preliminary Risk Analysis)	9/12/2008
M506040302	INL	M2	Complete analysis on cost-effective means to manage short-term heat in repository	3/31/2008
M506040301	INL	M3	Regulatory analysis governing radioactive waste disposition	3/31/2008
M506040303	INL	M3	Provide input to the GNEP Material Transportation Storage and Disposal Analysis FY-08 Summary Report	9/15/2008
M506040102	LLNL	M4	Input to scoping level risk assessment for alternative disposal pathways.	2/29/2008
M506040101	LLNL	M4	LLNL Material Transportation, Storage & Disposal input to Decision Package.	3/14/2008
M506040103	LLNL	M2	GNEP Material Transportation Storage and Disposal Analysis FY-08 Summary Report	9/30/2008
M506040403	SNL	M4	Waste Streams, Alternatives and Proposed Disposition Report--SNL Input	2/15/2008
M506040401	SNL	M2	Material Transportation, Storage and Disposal contribution for Secretarial Decision Package	3/14/2008
M506040404	SNL	M4	Submit input to LLNL for final GNEP Material Transportation, Storage and Disposal Analysis FY08 Summary Report	9/12/2008
M506040503	SRNL	M4	Participate in an interlaboratory meeting to discuss options regarding the high-heat wastes.	12/14/2007
M506040501	SRNL	M2	Documentation of the strategic aspects of the waste stream disposition paths.	2/29/2008

2 M-TSD ACTIVITY SUMMARIES

FY-2008 M-TSD activity summaries follow below listed alphabetically by participant.

2.1 M-TSD Summary for Argonne National Laboratory

Milestones: M506040201 and M506040602

Argonne National Laboratory (ANL) completed four major tasks in two Materials – Transportation, Storage and Disposal work packages: AN0815060402 and AN0815060406. Funding for work package AN0815060406 was reduced in April, 2008 and this work package was subsequently closed. This report presents the results of ANL's activities under those work packages, satisfying milestone M506040201 and (milestone M506040602 from closed work package AN0815060406).

Initial efforts in Fiscal Year 2008 supported the development of the GNEP Integrated Waste Management Strategy (IWMS). ANL reviewed drafts of the IWMS and provided feedback to the author. A major activity involved evaluating the current regulatory framework in the U.S. pertaining to the disposal of radioactive wastes that could potentially be generated under an advanced nuclear fuel cycle. This evaluation also investigated potential disposal pathways for these wastes. The evaluation was provided to the IWMS author, satisfying milestone M506040202 and text from the evaluation was used in the IWMS itself. The entire evaluation is provided in Appendix A.

ANL staff attended a workshop in Denver CO on November 13-14, 2008 to discuss the heat management evaluation that was to be conducted by Savannah River National Laboratory (SRNL). One of the scenarios ultimately considered in that evaluation was the recovery of plutonium from spent nuclear fuel very soon after reactor discharge. ANL was tasked with evaluating the repository thermal response of such a scenario. That task was completed in December 2008 with the transmittal of the results to SRNL. That analysis is presented below.

ANL also completed two activities in preparation for future systems analyses related to waste management. Each of these activities is discussed in further detail below.

The first effort involved the upgrading of the simplified repository assessment model that has been used in past analyses of repository benefits associated with recycling to the latest version of the simulation software. This effort was a non-trivial task due to large architecture changes in the software. Ultimately, the upgraded model produced results that are very similar to those documented in previous analyses.

The second activity involved estimating the inventory of key radionuclides that would be present in waste streams and forms that may be generated under an advanced nuclear fuel cycle. Key radionuclides are identified through a screening process. The radionuclide inventory is calculated on a per MTHM of SNF processed basis for the various waste streams that would be generated under the UREX process. An example calculation of the radionuclide inventory in potential waste forms was also developed.

Repository Thermal Analysis – Early Plutonium Separation Waste Disposal

From a thermal perspective, recent studies [Ref. 1] have shown that the heat generated by five of the elements found in spent fuel are instrumental in limiting storage in geologic repositories. In rough order of importance, they are: Pu, Am, Cs, Sr, and Cm. However, readily available "separation and recycle technologies" are at the moment applicable only to the recovery of plutonium. Hence, it is reasonable to ask what repository loading benefits in the near term can be achieved by removing only plutonium from spent fuel.

Under these circumstances, additional benefits could potentially arise from removing Pu from the spent fuel as quickly as possible following discharge from the reactor. The motivation is that ^{241}Am, one of the most significant decay heat sources in spent fuel, arises principally as a beta-decay product of ^{241}Pu. Thus, quick enough separation of plutonium (relative to the 14 year half-life of ^{241}Pu), will also remove a significant fraction of the decay heat from americium. Given (1) relative amounts of ^{241}Pu and ^{241}Am in spent fuel at discharge and (2) the delay time-interval between discharge and processing, it is straightforward to estimate an "effective" americium removal fraction.

Using 50 GWd/MT spent fuel discharged from a PWR as an example [Ref. 2], Figure 1 shows the fraction of Am decay-heat remaining in the waste stream versus processing delay-time. These estimates assume ^{241}Am to be the element's predominant decay-heat source and the 433 year half-life of ^{241}Am (s) >> the 14 year half-life of ^{241}Pu. Figure 2 plots the actual spent fuel decay heat that results from the indicated removals of plutonium for several processing delay-times. As expected, the time-frame (~200-2000 years), where computed decay heat seems most sensitive to processing delay time, corresponds closely to where the contribution of ^{241}Am is most prominent [Figure 1 of Ref. 1).

To estimate possible repository loading benefits from recovering Pu-only, the decay heating curves from Figure 2 have been analyzed using a simplified thermal model of the proposed Yucca Mountain Repository that has been used in past analyses under the Advanced Nuclear Fuel Cycle Initiative [Ref. 1]. These analyses assumed wastes generated from processing PWR spent nuclear fuel is disposed 25 years after reactor discharge and cooling airflow in the repository (forced ventilation) is terminated 75 years later with repository closure. Temperature limits of 200 °C at the drift wall and 96 °C at locations midway between drifts, consistent with the thermal criteria being used in the design of the Yucca Mountain Repository [Ref. 1]. Repository benefit factors are found by comparing computed loading limits (GWd/m) that could be achieved while meeting the imposed temperature limits to that of direct spent fuel disposal.

Peak drift wall temperatures are principally dependent on decay heat power at times of repository disposal and closure. On the other hand, peak temperatures midway between drifts depend in large part on the integral of decay heat from repository closure onward- the time-frame where Pu-only removal is most influential (see Figure 2). To reduce the importance of the drift wall temperature limit somewhat, an option to delay disposal an additional 50 years beyond the 25 year reference has been added to the present study. This option is intended to lower peak drift

wall temperatures by reducing the decay heat contribution from short-lived isotopes of Cm, Cs, Sr, Ba, and Y.

A graph of computed benefit factors are shown in Figure 3 as a function of processing delay for 25 and 75 year disposal times. Figure 3 results clearly show that achieving the largest repository benefits require a combination of early processing and delayed disposal. As expected, strong dependence of benefit factor on processing delay is associated with requires a long enough disposal delay so that the between-drift temperature limit plays a significant role. Specifically, if waste is disposed at 25 years, a 1-10 year range of processing delays corresponds to a benefit factor range of only ~3-4. However, if waste is disposed at 75 years, benefit factors ~10 are possible provided processing takes place within 3 years. However, if processing is delayed much beyond ~5 years, any additional repository benefit from delaying disposal is likely to be small.

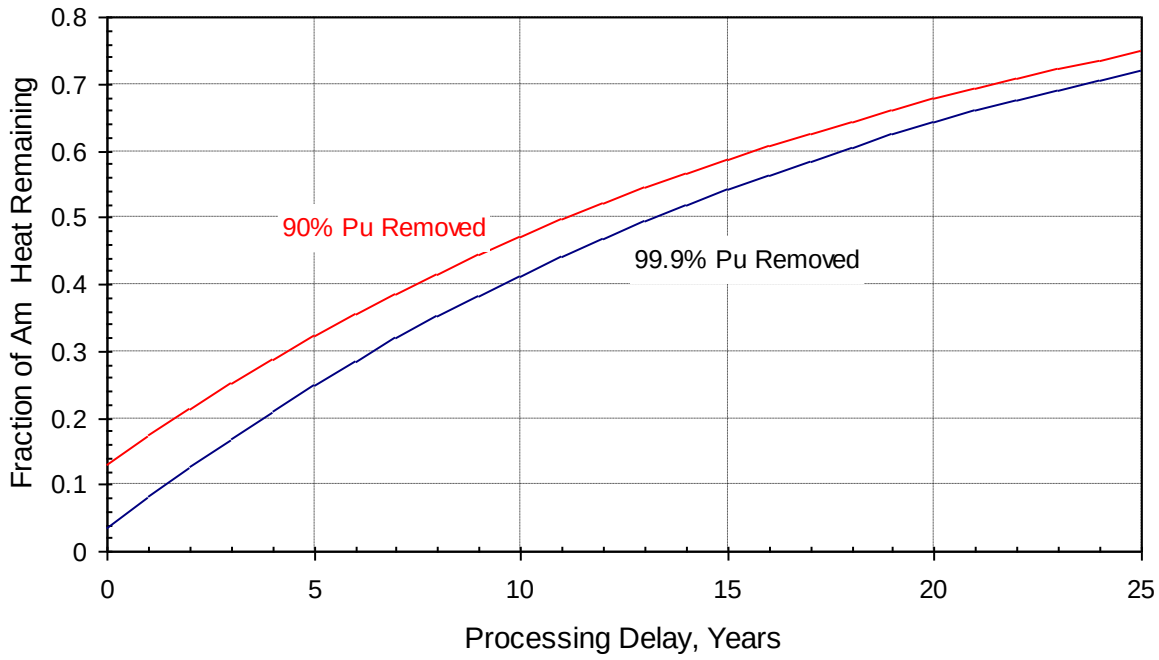


Figure 1 Effective Am "Removal" from Spent Fuel with Pu-Only Processing (50 GWd/MT PWR)

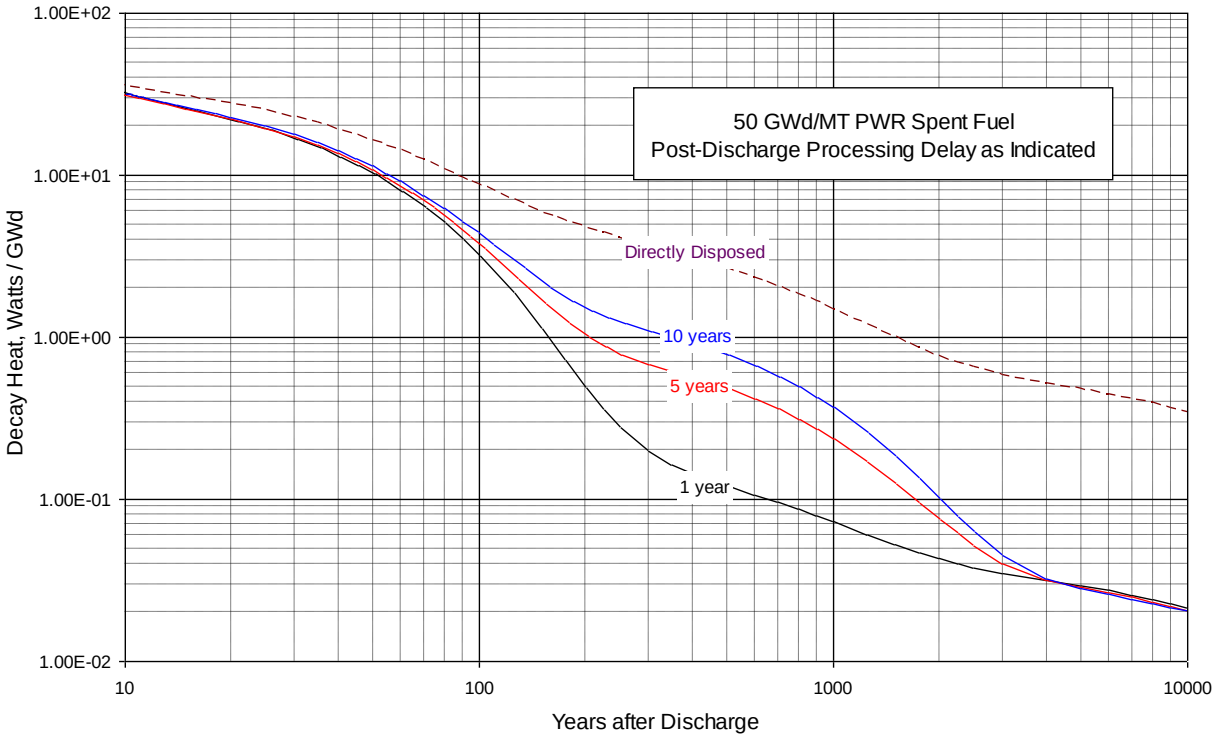


Figure 2 Decay Heat from Spent Fuel with 99.9% Pu Removed at Different Times

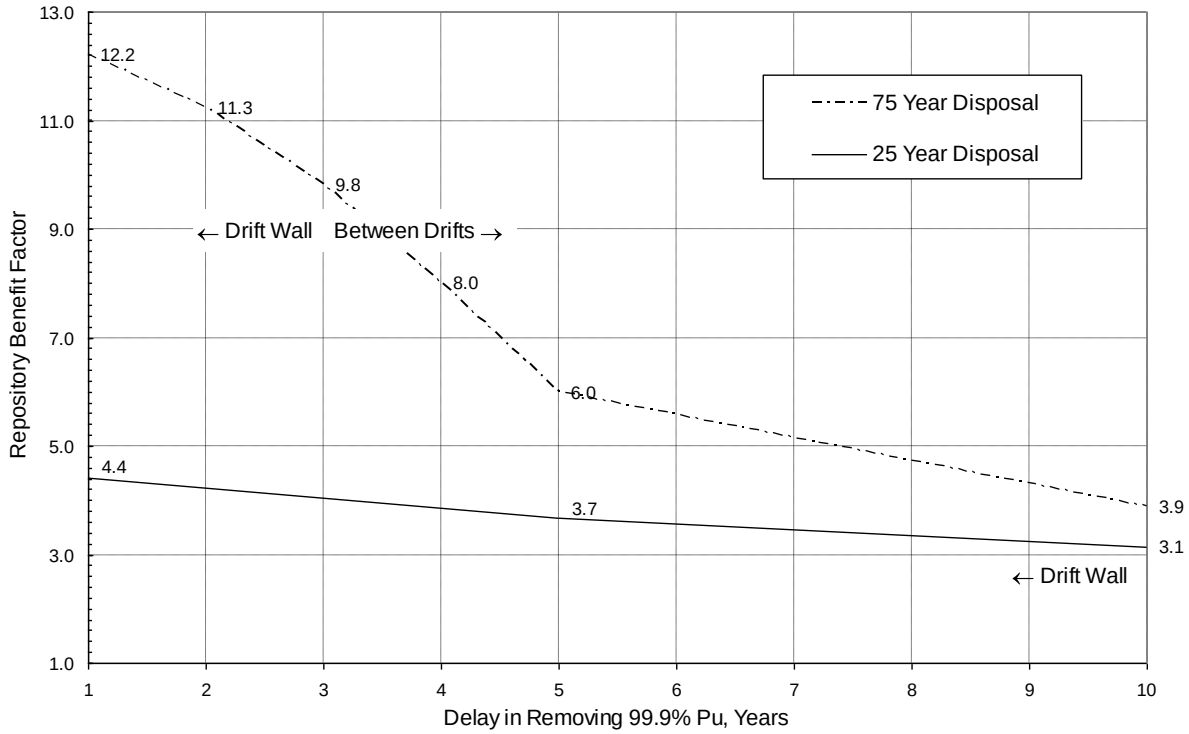


Figure 3 Effect of Processing and Disposal Delays on Potential Repository Benefits. Controlling Temperature Limit Locations are Indicated.

1. R. A. Wigeland, T. H. Bauer, T. H. Fanning, and E. E. Morris, “Separations and Transmutaion Criteria to Improve Utilization of a Geologic Repository”, Nucl. Tech. *154*, p.95 (2006).

2. E. A. Hoffman, W. S. Yang, R. N. Hill, “A Study on Variable Conversion Ratio for Fast Burner Reactor”, Trans. Am. Nucl. Soc., June (2007).

Updating of Simplified Repository Assessment Model

The simplified repository assessment model used in evaluating repository performance under the Advanced Nuclear Fuel Cycle Initiative [Ref. 1] has been updated to run with Version 9.60 SP3 of the GoldSim simulation software [Ref. 2]. Previously, the model was developed and executed under Version 8.01 SP1 of the GoldSim software. Note that this model is based on the Total System Performance Assessment - Site Recommendation model developed by the Yucca Mountain Project [Ref. 3] and no changes to the model itself, beyond those needed to execute the model under Version 9.60 SP3, have been made. In addition, the Version 9.60 SP3 model does not include the igneous intrusion or the seismic event models that were included in the Version 8.01 SP1 model.

The first step in the updating process was to load the model into Version 9.21 of GoldSim because the Version 8.01 GoldSim model would not load directly into Version 9.60 due to significant changes in software architecture between Versions 8.01 and 9.60 of the software. An intermediate update to Version 9.21 was required to make needed changes to the model prior to the final update to Version 9.60.

A large number of changes to the model were required to update the model into Version 9.21. These included moving associated cells into source element containers and introducing a sizeable number of previous time value elements to replace the use of "~" notation in functions to indicate use of a previous time value. Because of the number of changes involved, it was decided to simply recreate the model in Version 9.21. In some cases, this was as simple as cutting and pasting elements from the model as loaded into Version 8.01 to a new model in Version 9.21. In other cases, the elements from Version 8.01 could not be moved by cutting and pasting and they had to be recreated individually.

Once the complete model was recreated in Version 9.21, the recreated model was loaded into GoldSim Version 9.50 for further debugging prior to the final migration to Version 9.60. Initial attempts to run the model under Version 9.50 revealed that it could only be run in low precision mode (numerical solver for imposing dissolved concentration limits).

With assistance from staff of the GoldSim Technology Group, the reason for this limitation was traced to discrete changes being made in the waste form and invert cells associated with the radionuclide sources. These changes were made to reflect changes in the number of source containers in seepage and non-seepage environments. The changes involved adding mass from one cell while subtracting the same mass from another cell. This process sometimes resulted in too much mass being subtracted from a cell because the amount of mass involved on a given time step was determined based on the cell masses on a previous time step. The newer versions of the GoldSim software are much less forgiving of this error than the older version (8.01). To circumvent this problem, the amount of mass to be moved from one cell to another was used to calculate the new mass that should be present in the cell. This calculation was carried out in such a way that the amount of mass that should be present in the cell was never negative. Then discrete change elements were used to replace cell masses rather than add or subtract mass from the cells. With the introduction of this change, it was possible run the Version 9.60 model in medium or high precision as well as low precision.

Most of the debugging work was carried out by comparing expected value runs with the 9.50 Version of the model and comparing with the corresponding results of an expected value run with Version 8.01. Because mean values for stochastic elements were used in the two calculations, in principle the two versions of the model should produce identical results. As will be seen, because of changes in GoldSim Versions 9.50 and 9.60 relative to GoldSim Version 8.01, the results are not identical. For comparison, all the new results included in this memorandum were computed with Version 9.60.

Figure 1 shows the release rate of ^{237}Np from the engineered barrier system for commercial spent nuclear fuel waste packages. The figure shows that while results from the two calculations are not identical, agreement is very close. Equally good agreement is obtained for ^{129}I, ^{230}Th, and ^{234}U.

Figure 2 shows that agreement is not as good for the corresponding release rate for ^{242}Pu. The difference shown in this figure is due mostly to the fact that Version 9.6 of GoldSim allows diffusion of suspended solids used to model the transport of radionuclides associated with colloids between the waste form cells and the invert cells while Version 8.01 did not allow such diffusion. Concentrations of the suspended ground-water and iron oxide colloids are much larger in the invert than in the waste form with the result that these colloids diffuse from the invert to the waste form, carrying sorbed radionuclides with them. The shape of the curves in Fig. 2 is determined mostly by the diffusion between the mixing cell representing the waste form and the mixing cell representing the invert in the non-seepage environment. Because plutonium is at the solubility limit during much of the first 700,000 years, the diffusion of the colloids results in a smaller diffusive flux from the waste form to the invert in the newer model. The abrupt drops between 700,000 and 750,000 years occur when the concentrations of plutonium drop below the solubility limit. This takes longer in the newer model because of the reduced diffusive flux. The large increase in the release rate that occurs at about 900,000 years is caused by a seismic event which causes all cladding to fail.

Engineered barrier release rates from commercial spent nuclear fuel with stainless steel cladding show agreement for ^{237}Np, ^{129}I, ^{242}Pu, ^{230}Th, and ^{234}U that is comparable to that shown in Figure 1 for ^{237}Np except for an anomaly in the release rate for ^{230}Th that occurs at the time of the seismic event mentioned at the end of the foregoing paragraph. The ^{230}Th result is shown in Figure 3. Aside from the observation that the spike occurs at the time of the seismic event (the plot includes only every other calculated time point and there is a downward spike at the subsequent time point that is not plotted), no reason has been found for the spike. All mass has been released from the stainless steel clad spent nuclear fuel waste form long before the seismic event occurs.

Figure 4 shows the release rate of ^{129}I from the engineered barrier system due to the release of radionuclides from the co-disposal waste packages. Agreement between the release rate as calculated with Version 8.01 and that from Version 9.60 is very good for this isotope and is equally good for ^{237}Np and ^{234}U. Agreement for ^{242}Pu is not as good and is particularly poor for ^{230}Th. The comparison for ^{242}Pu is shown in Figure 5 and for ^{230}Th in Figure 6. In both cases, the differences are due to the transport of colloids from the invert to the waste form cells in the case of Version 9.60 and the lack of this transport in Version 8.01. In addition to iron oxide and groundwater colloids, the model includes waste form colloids for the co-disposal waste packages due to the presence of borosilicate glass in these packages. The difference between the two calculations is particularly large in the case of ^{230}Th because for the waste form colloids the distribution coefficient for thorium is ten times larger in the invert than in the waste form. As was noted earlier, the colloid concentrations are much larger in the invert than in the waste form. However, because the iron oxide and waste form colloids form in the waste packages, this large difference in concentration may not be reasonable, particularly in the non-seepage environment where the only transport mechanism is diffusion.

Figures 7, 8, and 9 show comparisons of the release rates, respectively, for ^{129}I, ^{242}Pu, and ^{230}Th from the unsaturated zone to the saturated zone. The agreement for ^{129}I is excellent and equally good agreement is observed in plots (not shown) for ^{237}Np and ^{234}U. Agreement for ^{242}Pu and ^{230}Th is not as good for the reasons discussed above regarding the release rates from the engineered barrier system. Figure 10 shows the comparison of the total dose rates as computed by Versions 9.6 and 8.01 of GoldSim. The comparison shown in Figure 10 indicates good agreement between the two code versions notwithstanding the differences observed in the release rates for some of the individual isotopes. Comparisons of the dose rate contributions for selected individual isotopes are similar to those shown in Figures 7, 8, and 9 for the release rates from the unsaturated zone.

One thousand realization cases were run with each of the models considered above. Comparison of the mean values of the total dose rate for these cases is shown in Figure 11. The dashed curves shown on the figure indicate the range of plus or minus one standard deviation for the mean values obtained with the two models. Over much of the time range covered by the figure the standard deviations overlap and over

the remainder of the time range the plus-one-standard-deviation curve for the Version 9.60 result is about the same as the minus-one-standard-deviation curve for the Version 8.01 result. Based on the estimated standard deviations indicated in Figure 11, one can conclude that the differences between the two cases are not statistically significant.

1. R.A. Wigeland and E.E. Morris, Processing Requirements for PWR Spent Fuel to Reduce Estimated Peak Dose Rate Associated with Releases from a Geologic Repository, Argonne National Laboratory, ANL-AFCI-166, April 2006.
2. "Users Guide: GoldSim Probabilistic Simulation Environment," GoldSim Technology Group (October 2006).
3. "Total System Performance Assessment for Site Recommendation," TDR-WIS-PA-000001 REV 00 ICN 01 (December 2000).

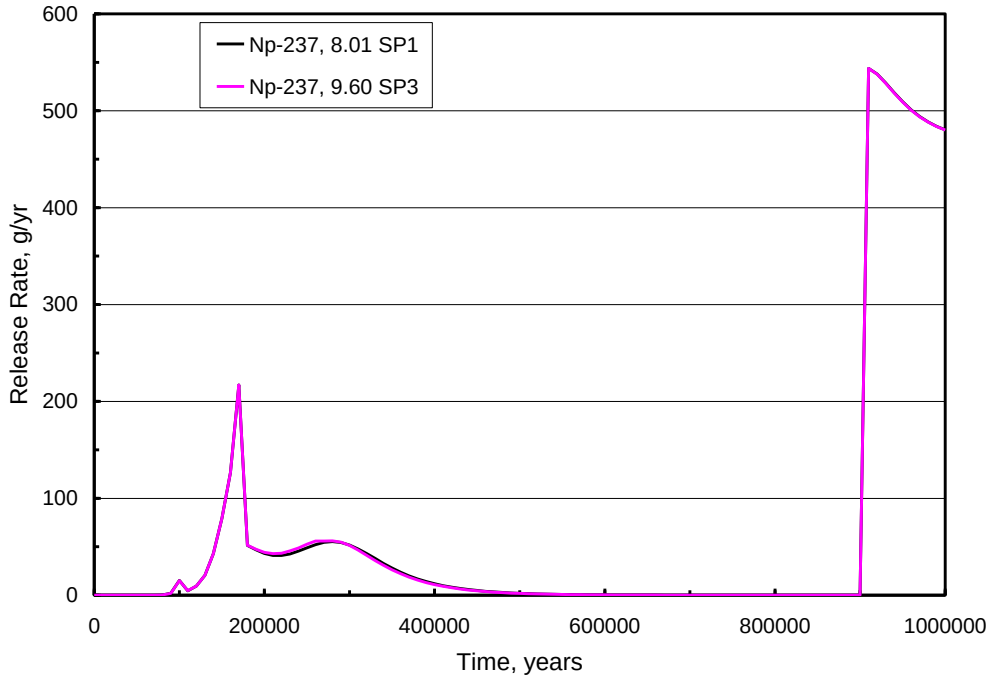


Figure 1 Comparison of release rates of ^{237}Np from the engineered barrier from commercial spent nuclear fuel waste packages.

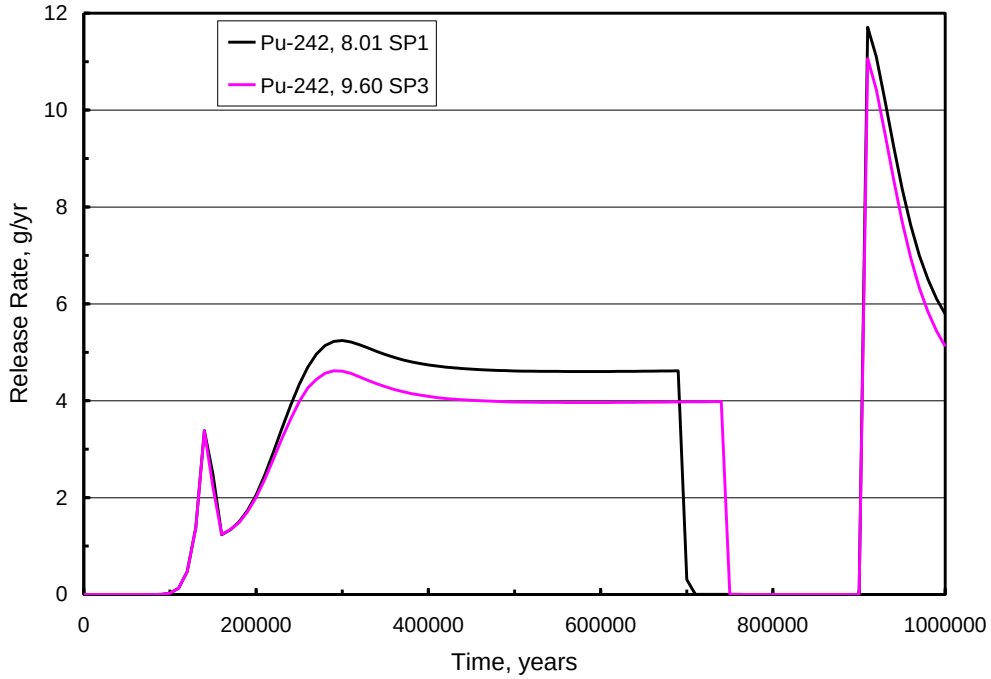


Figure 2 Comparison of release rates of ^{242}Pu from the engineered barrier from commercial spent nuclear fuel waste packages.

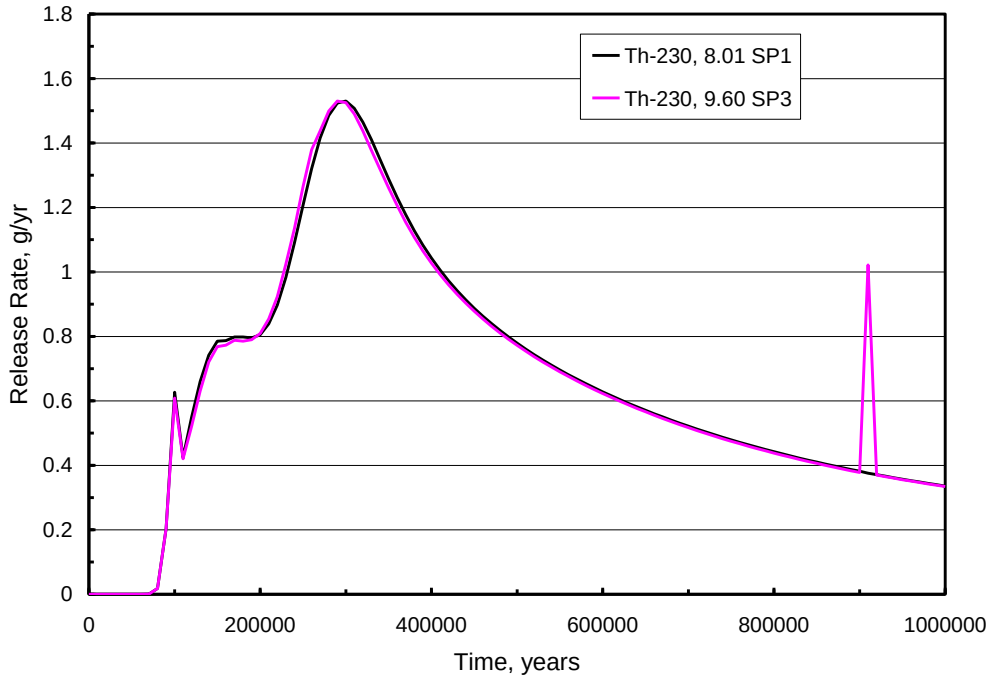


Figure 3 Comparison of release rates of ^{230}Th from the engineered barrier from waste packages containing commercial spent nuclear fuel with stainless steel cladding.

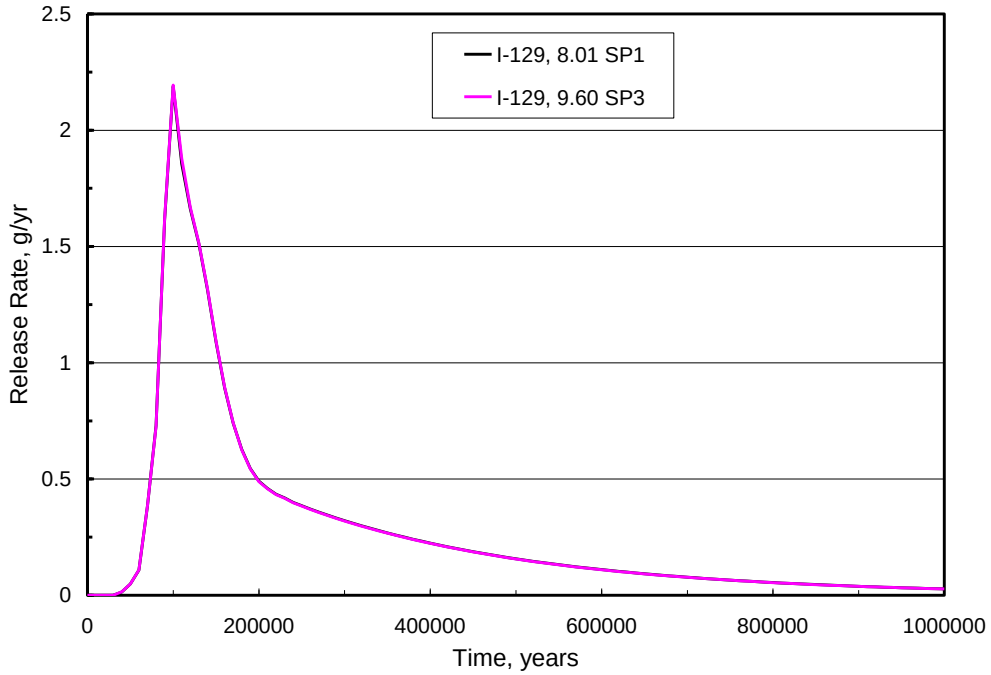


Figure 4 Comparison of release rates of ^{129}I from the engineered barrier for co-disposal waste packages.

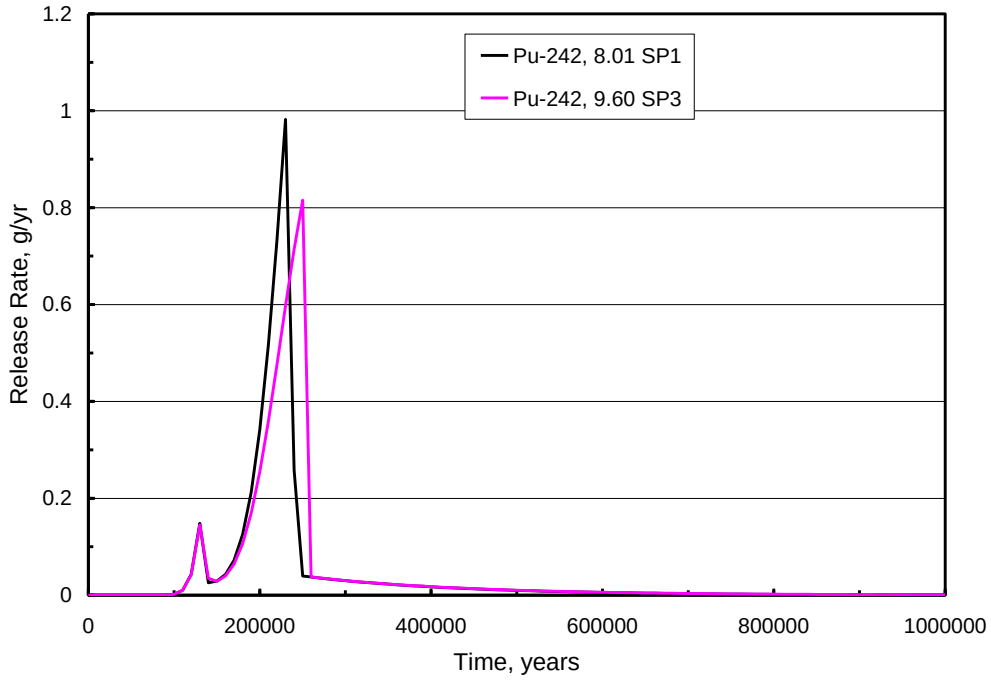


Figure 5 Comparison of release rates of ^{242}Pu from the engineered barrier for co-disposal waste packages.

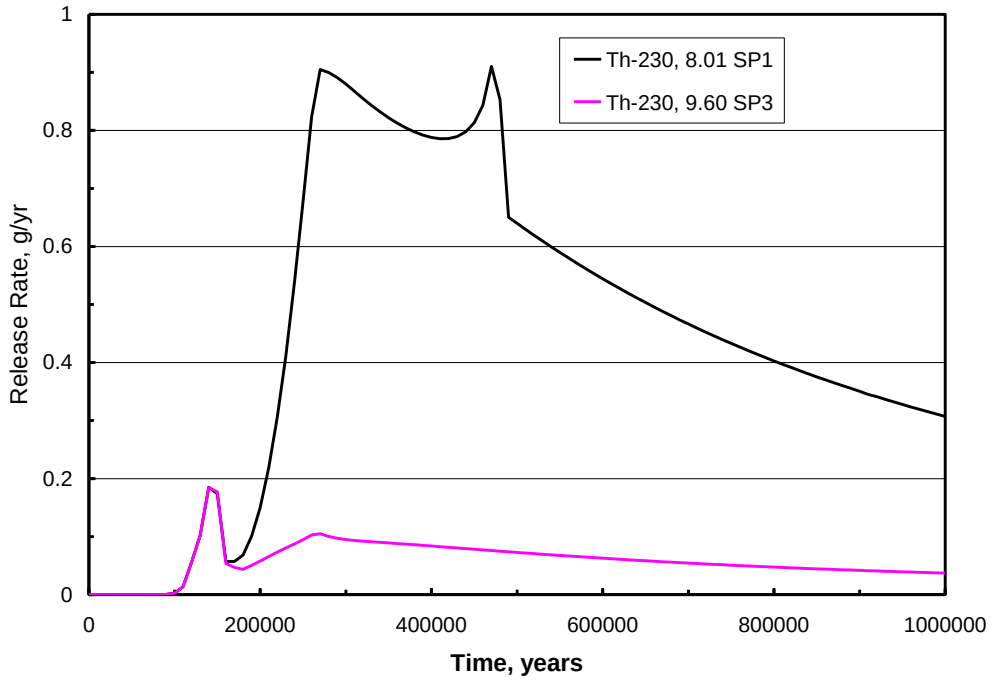


Figure 6 Comparison of release rates of ^{230}Th from the engineered barrier for co-disposal waste packages.

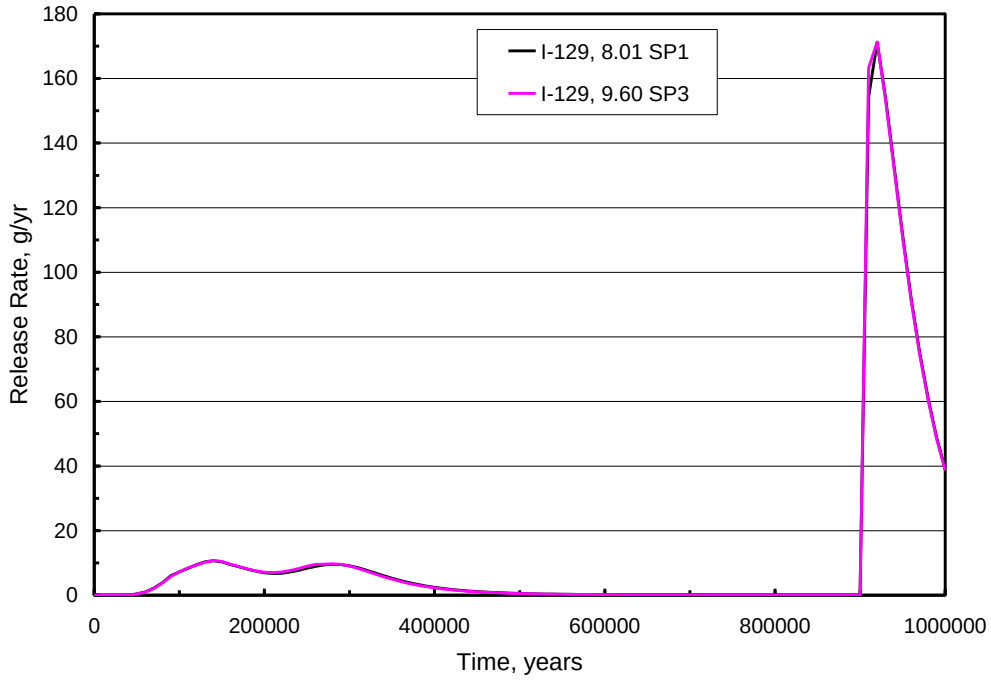


Figure 7 Comparison of release rates of ^{129}I from the unsaturated zone to the saturated zone.

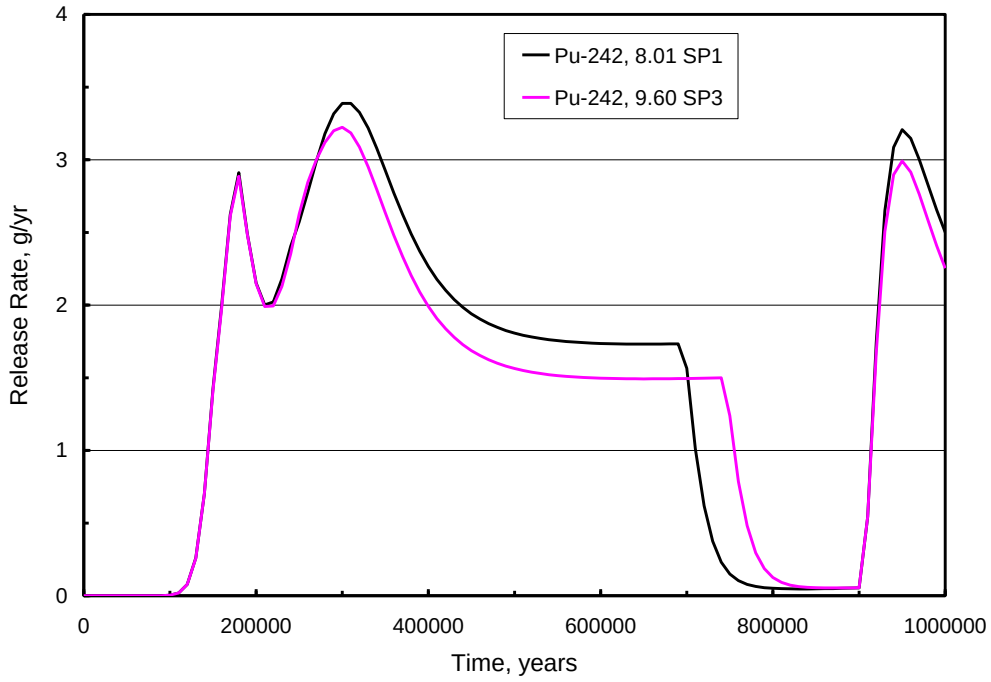


Figure 8 Comparison of release rates of ^{242}Pu from the unsaturated zone to the saturated zone.

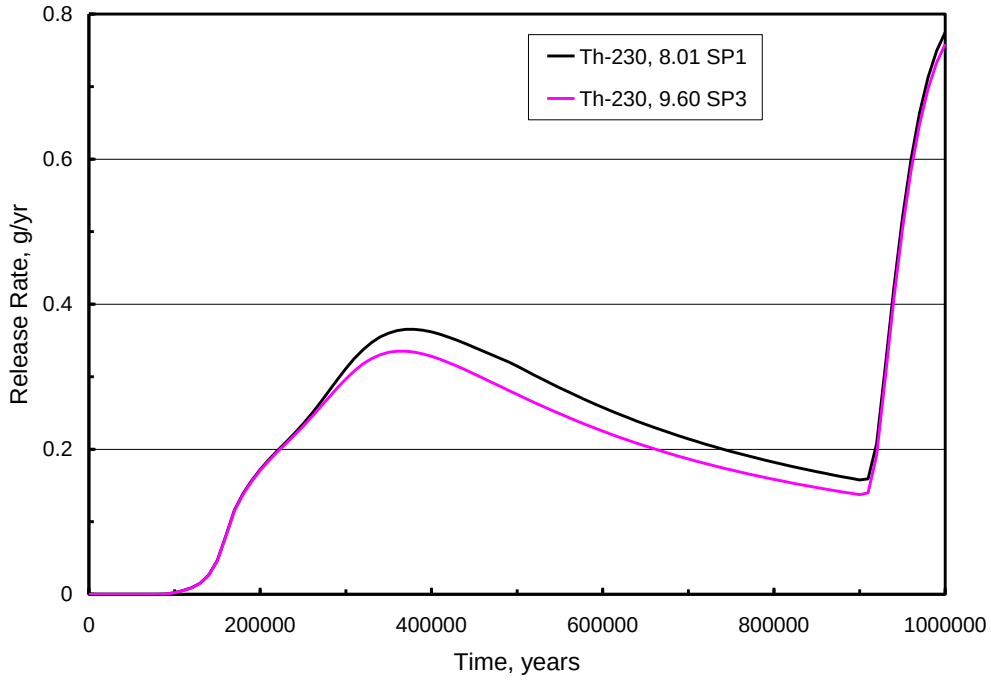


Figure 9 Comparison of release rates of ^{230}Th from the unsaturated zone to the saturated zone.

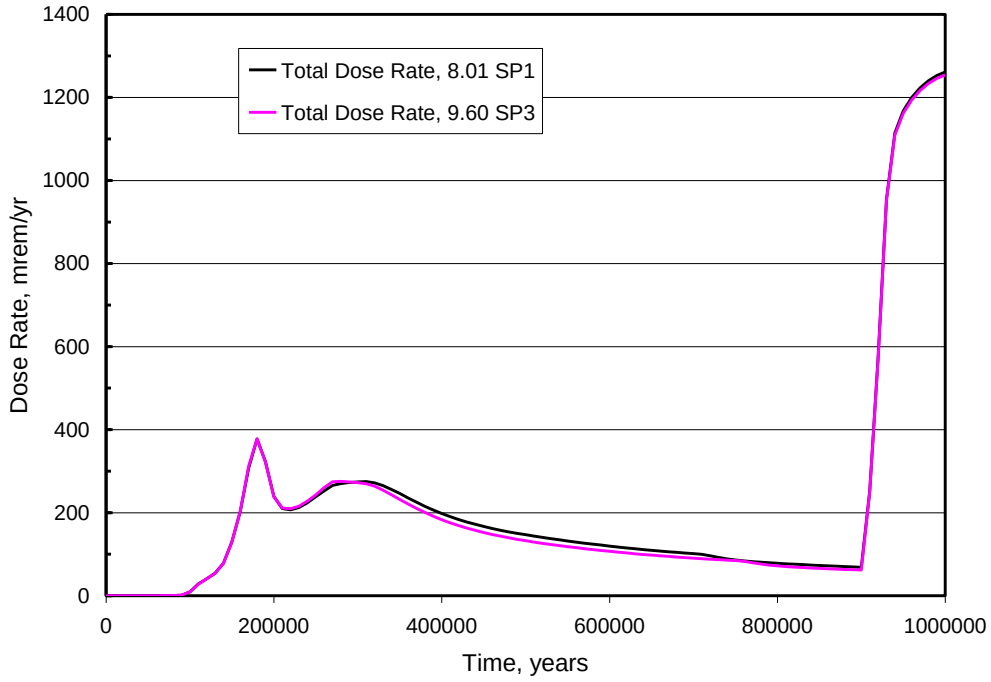


Figure 10 Comparison of total dose rates for expected value cases.

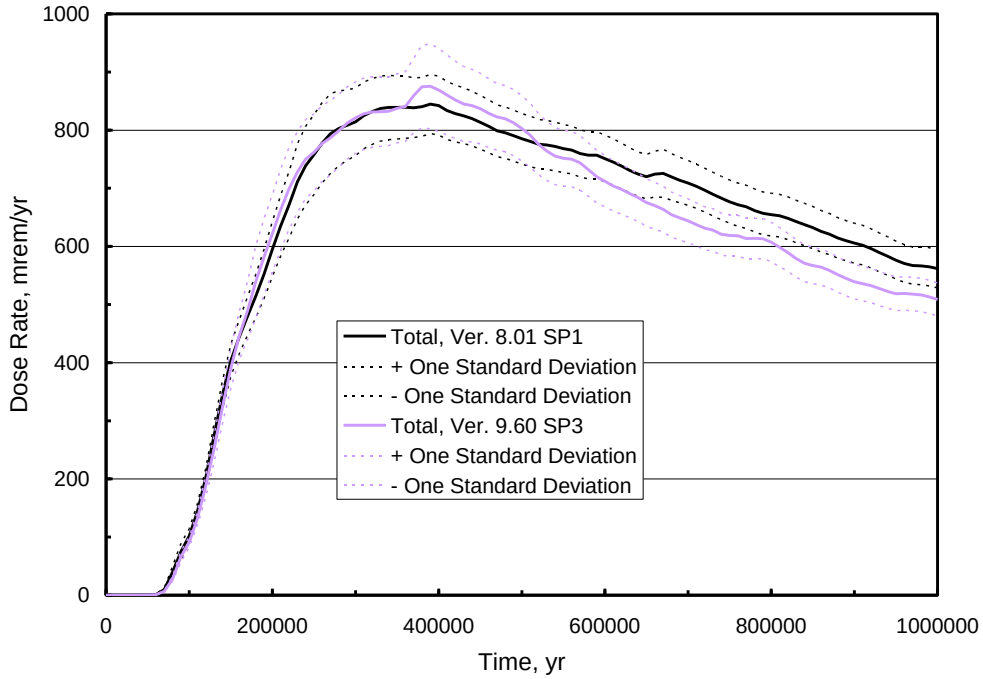


Figure 11 Comparison of total dose rates for 1000-realization cases.

Preliminary Estimate of the Inventory of Waste Generated under an Advanced Nuclear Fuel Cycle

The objective of this calculation is to estimate the radionuclide inventory in waste forms that could potentially be generated under an advanced nuclear fuel cycle where light water reactor (LWR) spent nuclear fuel (SNF) is reprocessed to recover uranium, plutonium, and the other minor actinides.

Inventory Calculation

The waste forms considered in this calculation are those that would result from the processing of LWR SNF using the UREX+ process that would likely require long term disposal (i.e., geologic isolation). These waste forms are described in the Global Nuclear Energy Partnership (GNEP) Integrated Waste Management Strategy (IWMS) [Ref. 1] and are listed in Table 1.

Table 1 Potential Waste Forms from UREX+ Processing of LWR SNF

UREX+ Waste Stream	Possible Waste Forms
Un-dissolved Solids (UDS)	Metal alloy potentially combined with Tc and TMFP.
Tc	Metal Alloy, possibly containing UDS and TMFP. Alloy may require Zr/Fe, which could come from cladding and hardware.
Cs/Sr	Glass or Ceramic, process design should consider ramifications of high heat, high radioactivity, powder handling should be avoided.
Lanthanides	Glass— borosilicate glass if segregated as separate Ln stream. Ln/FP borosilicate glass if Ln and TMFP streams are combined.
Transition Metal Fission Products (TMFP)	Metal alloy potentially combined with Tc and UDS. Borosilicate glass if combined with lanthanides.
Iodine	Grouted silver zeolite.
Source: Global Nuclear Energy Partnership Integrated Waste Management Strategy, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008. Table 1.	

The calculation began with the set of radionuclides that are present in spent LWR fuel enriched to 4.3 weight percent and irradiated to a burnup of 51 Gwd/MTHM. The radionuclides were separated into transuranic and fission product elements. The fission product radionuclide set was further segregated into the following waste streams:

- Lanthanides
- Transition Metal Fission Products (TMFP)
- Un-dissolved Solids (UDS)
- Cesium/Strontium (and decay products barium and yttrium)
- Iodine

It was assumed that 25% of Zr, Mo, Tc, Ru, Rh, and Pd remain un-dissolved and were partitioned to UDS [Ref. 1]. The remaining 75% of the elements were partitioned to the transition metal fission products.

The total mass of set, including stable isotopes, as a function of time after discharge from the reactor is shown in Figure 1. The total mass of lanthanide and TMFP are very similar. The mass of transuranics, assuming a separation efficiency of 99.9%, is three orders of magnitude lower than the mass of either lanthanide or transition metal fission products. If none of the transuranic isotopes were recovered, then their mass would be similar in magnitude to the mass of both the lanthanide and transition metal fission products. There is a slight build-up in the total mass of both the transition metal and UDS due to the generation of Zr-90 resulting from the decay of Sr-90.

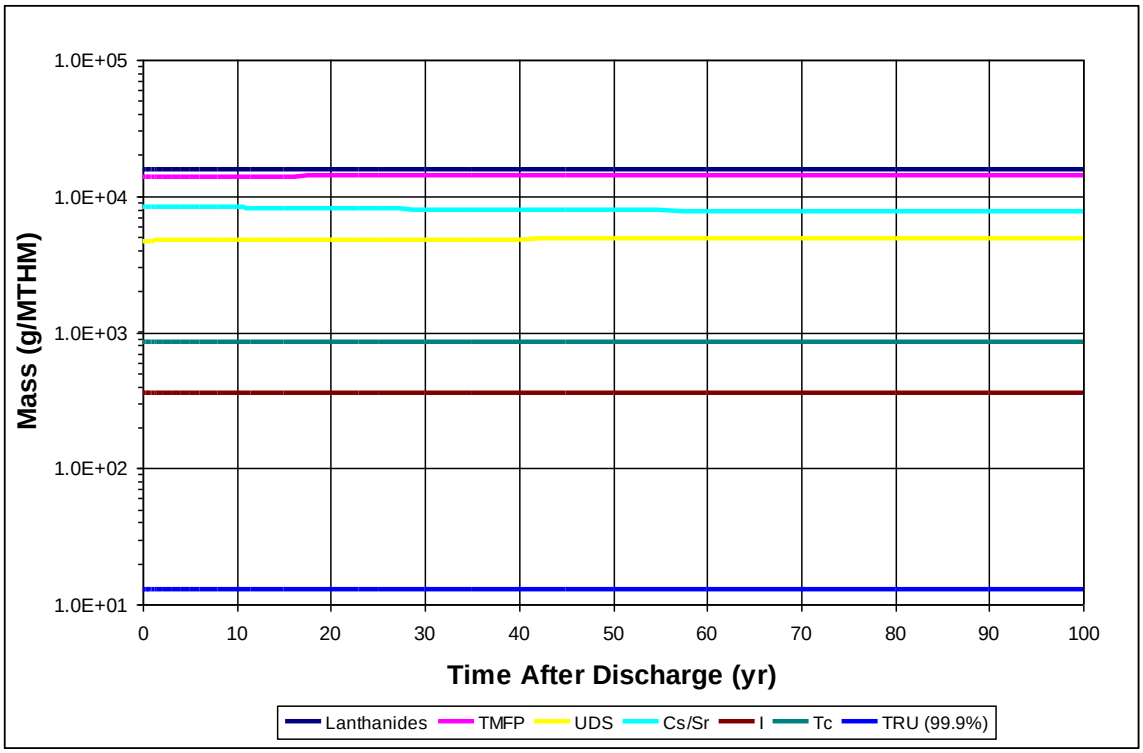


Figure 1 Total Mass of UREX+ Waste Streams

The stable isotopes were then removed from each waste stream set. Figure 2 shows the mass of radioactive isotopes in the lanthanide, TMFP, UDS, cesium/strontium, iodine, and krypton (noble gas) waste streams. A significant fraction of the processed wastes consist of stable isotopes. Figure 3 shows the fraction of recovered mass that is radioactive in each of the waste streams. Note that the Tc and transuranic element waste streams are entirely radioactive and are not shown in Figure 3.

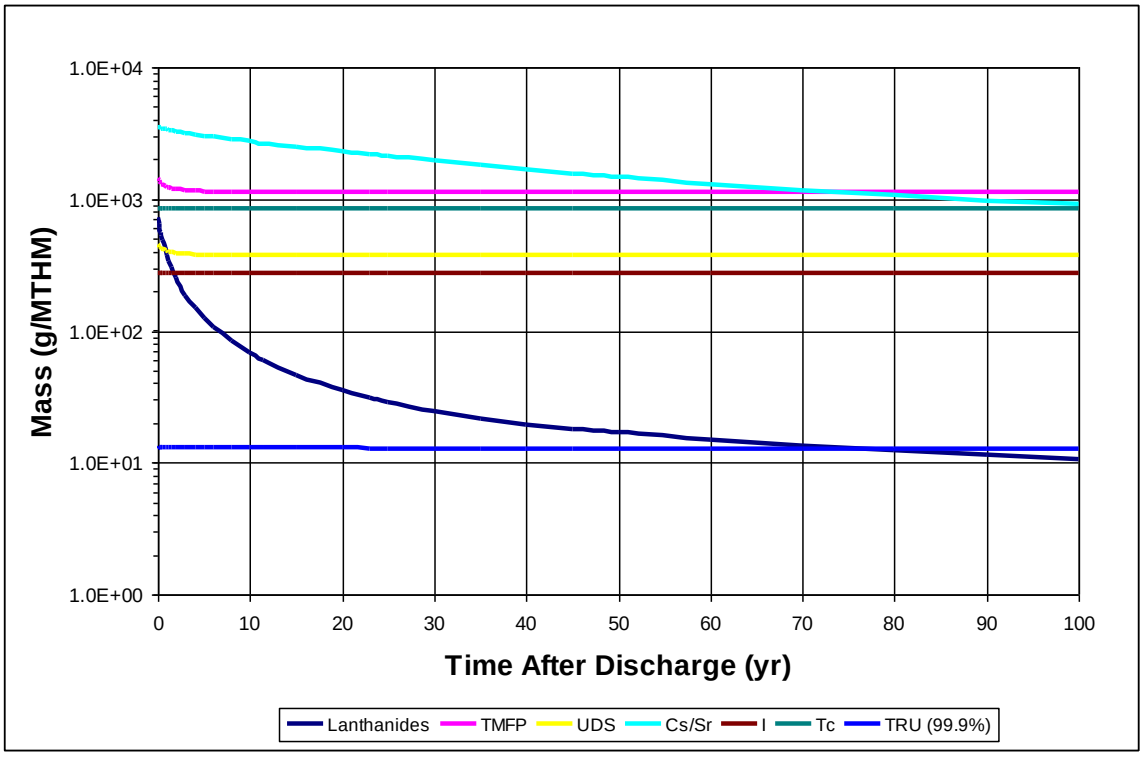


Figure 2 Total Mass Radioactive Isotopes in UREX+ Waste Streams

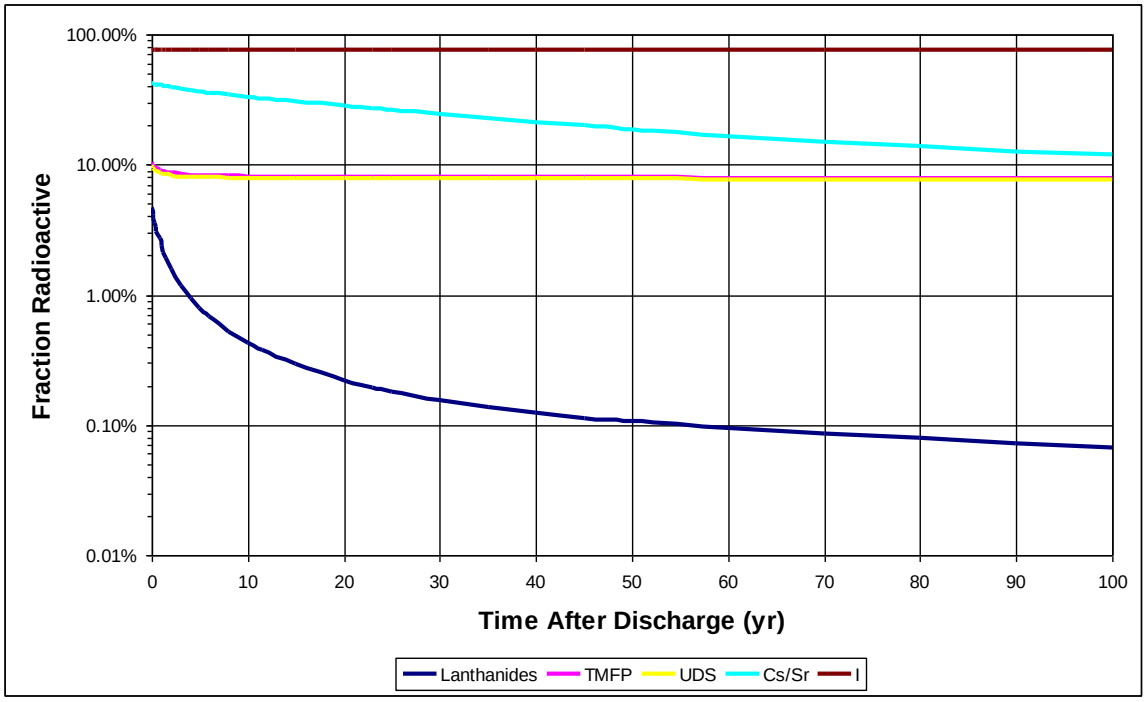


Figure 3 Fraction of Mass That is Radioactive in UREX+ Waste Streams

Table 2 shows the estimated loadings of potential waste forms that could be used to isolate wastes generated under an advanced nuclear fuel cycle, the fraction of radioactive isotopes in these streams, and the radioactive isotope waste loading for each waste potential waste form. For several of the potential waste forms (lanthanide, transition metal fission product, and un-dissolved solids) the radioactive fraction is less than 10 weight percent of the total mass of the waste that would be in the waste stream, with the remaining 90+ weight percent being stable isotopes. For those waste forms whose waste loadings are controlled by solubility (glass) or alloying (metal), the amount of radioactive waste that could be contained in the waste form would be controlled more by the loading of recovered stable isotopes rather than radioactive isotopes.

Table 2 Estimated Total Waste Loading and Radioactive Waste Loading in Potential UREX+ Waste Forms

Waste Stream[a]	**Estimated Waste Loading**[a] **(weight %)**	**Fraction of Radioactive Isotopes**[c] **(weight %)**	**Radioactive Isotope Waste Loading (weight %)**
Lanthanides/Glass	30 - 60	<0.5	0.15 – 0.3
Lanthanides + Transition Metals / Glass	20 - 30	< 5	1 - 1.5
Transition Metals + UDS + Tc / Metallic	40 - 55[b]	< 10	4.0 - 5.5
Cs/Sr / Glass or Ceramic	20 - 50	30	6 - 15
Tc / Metallic	40 - 55	100	40 - 55
I / Encapsulated Zeolite	2 - 7	75	1.5 - 5.25

[a]Global Nuclear Energy Partnership Integrated Waste Management Strategy, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008.
[b]The GNEP IWMS[a] estimates the upper end of range at 85% when including Fe added to process steps as waste. The Upper end in this calculation is estimated at 55 wt. % from the Global Nuclear Energy Partnership Integrated Waste Management Strategy Waste Treatment Baseline Study, Volume 1, GNEP-WAST-AI-RT-2007-000324, September 2007.
[c]Estimated at 10 years after reactor discharge

While each of the waste streams contains a large number of radioactive isotopes, it is only a limited number that pose a significant risk during storage and after disposal. The National Council on Radiation Protection and Measurements (NCRP) developed a series of simple screening techniques for the purpose of evaluating "small" releases of radioactive materials from point sources against environmental standards [Ref. 2]. The NCRP used conservative approaches in the development of screening factors. If compliance with environmental standards could be demonstrated for a small release using these conservative screening factors, then additional sophisticated modeling and analysis would not be required. While these screening factors are not appropriate for use in evaluating the risk associated with storing or disposing waste forms that would potentially be generated under an advanced nuclear fuel cycle, they can

be used to determine the importance of key radionuclides and to identify those that should be considered in future modeling and analysis efforts.

The activity of each radionuclide present in the waste stream was first calculated (on a per MTHM of SNF processed basis). The resultant activity was then multiplied by the NCRP screening factors to determine a "dose." The important radionuclides were determined by evaluating the "dose" contribution from a single radionuclide in the waste stream relative to the total "dose" from all radionuclides in the waste stream.

The NCRP ground release screening factors for a drinking water pathway, shown in Table 3, were used to determine the important radionuclides for consideration in disposal [Ref. 2, Table D.1]. The NCRP atmospheric release screening factors for inhalation and direct exposure from the plume and for ground deposition, shown in Table 4, were used to determine the important radionuclides for consideration in storage [Ref. 2, Table B.1].

The results of this screening analysis are shown in Table 5. The important radionuclides, pertaining to evaluating disposal system performance and waste storage, were determined by considering those radionuclides that contribute over 0.1% of the total screening "dose" for a waste form. The importance of radionuclides for storage considerations was determined over a 50-year period following reactor discharge. The key radionuclides are:

Waste Stream	Disposal	Storage
Lanthanides	Ho-166, Sm-146	Ho-166, Eu-150
TMPF	Sn-126, Se-79, Nb-93, Zr-93, Pd-107	Ru-106, Sb-125, Nb-95 (only for very short cooled)
UDS	Tc-99	Ru-106, Rh-102, Zr-95 (only for very short cooled
Cs/Sr	Cs-135	Cs-137, Sr-90
Tc	Tc-99	Tc-99, Tc-98
I	I-129	I-129
TRU	Pu-240, Pu-239, Np-237, Pu-242, Am-243, Am-241 (for ~1000 years)	Am-241, Pu-238, Pu-240, Pu-239, Cm-244, Am-243

Table 3 NCRP Ground Release, Water Pathway Screening Factors

Radionuclide	Water Pathway Screening Factor		Radionuclide	Water Pathway Screening Factor	
	mrem/bq	Sv/Bq		mrem/bq	Sv/Bq
AG107			PM147	1.7E-10	1.7E-15
AG108			PM148		
AG108M	4.2E-09	4.2E-14	PM148M	1.6E-15	1.6E-20
AG110			PR144		
AG110M	5.2E-10	5.2E-15	PR144M		
AM241	5.9E-08	5.9E-13	PU238	1.7E-07	1.7E-12
AM242M	1.6E-07	1.6E-12	PU239	2.0E-07	2.0E-12
AM243	6.0E-08	6.0E-13	PU240	2.0E-07	2.0E-12
BA137M			PU241	8.9E-10	8.9E-15
CD109	2.8E-09	2.8E-14	PU242	1.9E-07	1.9E-12
CD115M	4.0E-14	4.0E-19	PU244	2.2E-07	2.2E-12
CE141	9.6E-18	9.6E-23	RH102	8.4E-10	8.4E-15
CE144	3.6E-10	3.6E-15	RH103M		
CF249	2.7E-07	2.7E-12	RH106		
CF250	7.1E-08	7.1E-13	RU103	1.4E-15	1.4E-20
CF251	2.7E-07	2.7E-12	RU106	6.5E-09	6.5E-14
CM242	1.2E-10	1.2E-15	SB124	7.7E-13	7.7E-18
CM243	1.5E-08	1.5E-13	SB125	3.6E-10	3.6E-15
CM244	1.1E-08	1.1E-13	SB126	1.3E-27	1.3E-32
CM245	5.1E-08	5.1E-13	SB126M	1.9E-31	1.9E-36
CM246	2.9E-08	2.9E-13	SE 79	2.2E-09	2.2E-14
CM247	3.0E-08	3.0E-13	SM146	2.8E-08	2.8E-13
CM248	1.1E-07	1.1E-12	SM151	1.0E-10	1.0E-15
CM250	6.3E-07	6.3E-12	SN121M	1.1E-09	1.1E-14
CS134	4.2E-10	4.2E-15	SN123	4.3E-11	4.3E-16
CS135	1.4E-09	1.4E-14	SN126	1.1E-08	1.1E-13
CS137	7.7E-09	7.7E-14	SR 89	4.1E-13	4.1E-18
EU150	1.1E-09	1.1E-14	SR 90	3.5E-07	3.5E-12
EU152	9.1E-10	9.1E-15	TB160	4.7E-13	4.7E-18
EU154	1.1E-09	1.1E-14	TC 98	8.7E-07	8.7E-12
EU155	9.5E-11	9.5E-16	TC 99	3.2E-07	3.2E-12
GD153	2.5E-11	2.5E-16	TE123	8.1E-10	8.1E-15
HO166M	1.8E-09	1.8E-14	TE123M	1.0E-11	1.0E-16
I129	1.9E-05	1.9E-10	TE125M	4.7E-14	4.7E-19
IN114			TE127		
IN114M	2.1E-14	2.1E-19	TE127M	1.3E-11	1.3E-16
IN115M			TE129		
NB 93M	1.4E-10	1.4E-15	TE129M	1.0E-15	1.0E-20
NB 94	2.7E-09	2.7E-14	TM170	2.8E-09	2.8E-14
NB 95	7.0E-17	7.0E-22	TM171	1.0E-08	1.0E-13
NB 95M	8.0E-18	8.0E-23	Y 90		
NP236	9.4E-06	9.4E-11	Y 91	1.5E-13	1.5E-18
NP237	2.4E-05	2.4E-10	ZN 67		
PD107	2.2E-10	2.2E-15	ZR 93	1.7E-10	1.7E-15
PM146	2.5E-10	2.5E-15	ZR 95	8.8E-14	8.8E-19

Table 4 NCRP Atmospheric Release (Inhalation, and Direct Plume and Ground Exposure) Screening Factors

Radionuclide	Inhalation Screening Factor		Plume Screening Factor		Ground Screening Factor		Total Atmospheric Screening Factor	
	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3
AG107	0.0E+00		0.0E+00		0.0E+00		0.0E+00	0.0E+00
AG108	0.0E+00		1.9E-04	1.9E-09	1.0E-04	1.0E-09	2.9E-04	2.9E-09
AG108M	5.7E+01	5.7E-04	2.0E-01	2.0E-06	3.5E+04	3.5E-01	3.5E+04	3.5E-01
AG110	0.0E+00		3.0E-09	3.0E-14	3.1E-05	3.1E-10	3.1E-05	3.1E-10
AG110M	1.7E+01	1.7E-04	3.7E-01	3.7E-06	2.4E+03	2.4E-02	2.4E+03	2.4E-02
AM241	5.7E+04	5.7E-01	2.3E-03	2.3E-08	6.8E+02	6.8E-03	5.8E+04	5.8E-01
AM242M	5.4E+04	5.4E-01	7.1E-05	7.1E-10	5.2E+02	5.2E-03	5.5E+04	5.5E-01
AM243	5.6E+04	5.6E-01	6.4E-03	6.4E-08	6.0E+03	6.0E-02	6.2E+04	6.2E-01
BA137M	0.0E+00		8.0E-03	8.0E-08	3.8E-03	3.8E-08	1.2E-02	1.2E-07
CD109	1.3E+01	1.3E-04	2.7E-04	2.7E-09	1.5E+01	1.5E-04	2.8E+01	2.8E-04
CD115M	9.0E+00	9.0E-05	2.8E-03	2.8E-08	3.4E+00	3.4E-05	1.2E+01	1.2E-04
CE141	2.1E+00	2.1E-05	9.7E-03	9.7E-08	1.0E+01	1.0E-04	1.2E+01	1.2E-04
CE144	8.1E+01	8.1E-04	3.5E-03	3.5E-08	5.2E+01	5.2E-04	1.3E+02	1.3E-03
CF249	6.9E+04	6.9E-01	4.0E-02	4.0E-07	8.0E+03	8.0E-02	7.7E+04	7.7E-01
CF250	3.6E+04	3.6E-01	1.1E-05	1.1E-10	8.4E+00	8.4E-05	3.6E+04	3.6E-01
CF251	7.0E+04	7.0E-01	1.5E-02	1.5E-07	3.1E+03	3.1E-02	7.3E+04	7.3E-01
CM242	2.8E+03	2.8E-02	1.2E-05	1.2E-10	6.3E-01	6.3E-06	2.8E+03	2.8E-02
CM243	4.0E+04	4.0E-01	1.6E-02	1.6E-07	2.5E+03	2.5E-02	4.3E+04	4.3E-01
CM244	3.2E+04	3.2E-01	1.0E-05	1.0E-10	1.2E+01	1.2E-04	3.2E+04	3.2E-01
CM245	5.8E+04	5.8E-01	8.7E-03	8.7E-08	2.0E+03	2.0E-02	6.0E+04	6.0E-01
CM246	5.8E+04	5.8E-01	8.7E-06	8.7E-11	1.7E+01	1.7E-04	5.8E+04	5.8E-01
CM247	5.3E+04	5.3E-01	4.0E-02	4.0E-07	8.6E+03	8.6E-02	6.2E+04	6.2E-01
CM248	2.1E+05	2.1E+00	7.7E-06	7.7E-11	1.4E+01	1.4E-04	2.1E+05	2.1E+00
CM250	1.2E+06	1.2E+01	4.7E-04	4.7E-09	9.0E+03	9.0E-02	1.2E+06	1.2E+01
CS134	9.9E+00	9.9E-05	2.0E-01	2.0E-06	4.2E+03	4.2E-02	4.2E+03	4.2E-02
CS135	9.8E-01	9.8E-06	0.0E+00		0.0E+00		9.8E-01	9.8E-06
CS137	6.8E+00	6.8E-05	6.5E-02	6.5E-07	9.9E+03	9.9E-02	9.9E+03	9.9E-02
EU150	5.8E+01	5.8E-04	1.9E-01	1.9E-06	2.8E+03	2.8E-02	2.9E+03	2.9E-02
EU152	3.7E+01	3.7E-04	1.5E-01	1.5E-06	1.4E+04	1.4E-01	1.4E+04	1.4E-01
EU154	4.7E+01	4.7E-04	1.6E-01	1.6E-06	1.1E+04	1.1E-01	1.1E+04	1.1E-01
EU155	6.1E+00	6.1E-05	7.0E-03	7.0E-08	4.3E+02	4.3E-03	4.4E+02	4.4E-03
GD153	3.3E+00	3.3E-05	1.1E-02	1.1E-07	1.1E+02	1.1E-03	1.1E+02	1.1E-03
HO166M	1.1E+02	1.1E-03	2.0E-01	2.0E-06	3.8E+04	3.8E-01	3.8E+04	3.8E-01
I129	6.2E+01	6.2E-04	1.0E-03	1.0E-08	5.1E+02	5.1E-03	5.7E+02	5.7E-03
IN114	0.0E+00		3.2E-05	3.2E-10	9.6E-05	9.6E-10	1.3E-04	1.3E-09
IN114M	1.5E+01	1.5E-04	1.5E-02	1.5E-07	2.3E+01	2.3E-04	3.8E+01	3.8E-04
IN115M	2.5E-02	2.5E-07	1.9E-02	1.9E-07	1.2E-01	1.2E-06	1.6E-01	1.6E-06
NB 93M	6.3E+00	6.3E-05	1.7E-05	1.7E-10	1.3E+01	1.3E-04	1.9E+01	1.9E-04
NB 94	8.6E+01	8.6E-04	2.0E-01	2.0E-06	3.8E+04	3.8E-01	3.8E+04	3.8E-01
NB 95	1.3E+00	1.3E-05	9.7E-02	9.7E-07	9.6E+01	9.6E-04	9.7E+01	9.7E-04
NB 95M	6.0E-01	6.0E-06	7.3E-03	7.3E-08	8.8E-01	8.8E-06	1.5E+00	1.5E-05
NP236	2.2E+04	2.2E-01	1.6E-01	1.6E-06	3.8E+08	3.8E+03	3.8E+08	3.8E+03

Table 4 NCRP Atmospheric Release (Inhalation, and Direct Plume and Ground Exposure) Screening Factors (continued)

Radionuclide	Inhalation Screening Factor		Plume Screening Factor		Ground Screening Factor		Total Atmospheric Screening Factor	
	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3
NP237	6.2E+04	6.2E-01	2.8E-03	2.8E-08	6.1E+03	6.1E-02	6.8E+04	6.8E-01
PD107	2.8E+00	2.8E-05	0.0E+00		0.0E+00		2.8E+00	2.8E-05
PM146	3.0E+01	3.0E-04	9.3E-02	9.3E-07	5.0E+03	5.0E-02	5.0E+03	5.0E-02
PM147	8.2E+00	8.2E-05	4.7E-07	4.7E-12	1.3E-02	1.3E-07	8.2E+00	8.2E-05
PM148	2.7E+00	2.7E-05	7.3E-02	7.3E-07	1.0E+01	1.0E-04	1.3E+01	1.3E-04
PM148M	5.0E+00	5.0E-05	2.5E-01	2.5E-06	3.0E+02	3.0E-03	3.1E+02	3.1E-03
PR144	6.8E-03	6.8E-08	3.1E-03	3.1E-08	1.3E-03	1.3E-08	1.1E-02	1.1E-07
PR144M	0.0E+00		3.0E-04	3.0E-09	1.8E-04	1.8E-09	4.8E-04	4.8E-09
PU238	5.0E+04	5.0E-01	1.1E-05	1.1E-10	1.8E+01	1.8E-04	5.0E+04	5.0E-01
PU239	5.5E+04	5.5E-01	1.0E-05	1.0E-10	8.8E+00	8.8E-05	5.5E+04	5.5E-01
PU240	5.5E+04	5.5E-01	1.1E-05	1.1E-10	1.9E+01	1.9E-04	5.5E+04	5.5E-01
PU241	1.0E+03	1.0E-02	0.0E+00		0.0E+00		1.0E+03	1.0E-02
PU242	5.2E+04	5.2E-01	9.0E-06	9.0E-11	1.6E+01	1.6E-04	5.2E+04	5.2E-01
PU244	5.2E+04	5.2E-01	9.8E-05	9.8E-10	8.0E+03	8.0E-02	6.0E+04	6.0E-01
RH102	2.4E+01	2.4E-04	2.7E-01	2.7E-06	7.8E+03	7.8E-02	7.8E+03	7.8E-02
RH103M	1.0E-03	1.0E-08	2.2E-05	2.2E-10	1.6E-04	1.6E-09	1.2E-03	1.2E-08
RH106	0.0E+00		2.4E-07	2.4E-12	2.7E-04	2.7E-09	2.7E-04	2.7E-09
RU103	2.0E+00	2.0E-05	6.0E-02	6.0E-07	7.0E+01	7.0E-04	7.2E+01	7.2E-04
RU106	1.0E+02	1.0E-03	0.0E+00		0.0E+00		1.0E+02	1.0E-03
SB124	5.7E+01	5.7E-04	2.5E-01	2.5E-06	3.8E+02	3.8E-03	4.4E+02	4.4E-03
SB125	2.7E+00	2.7E-05	5.3E-02	5.3E-07	1.6E+03	1.6E-02	1.6E+03	1.6E-02
SB126	2.7E+00	2.7E-05	3.3E-01	3.3E-06	1.3E+02	1.3E-03	1.3E+02	1.3E-03
SB126M	5.9E-03	5.9E-08	1.5E-01	1.5E-06	7.6E-02	7.6E-07	2.3E-01	2.3E-06
SE 79	1.5E+00	1.5E-05	0.0E+00		0.0E+00		1.5E+00	1.5E-05
SM146	1.1E+04	1.1E-01	0.0E+00		0.0E+00		1.1E+04	1.1E-01
SM151	4.0E+00	4.0E-05	1.1E-07	1.1E-12	1.1E-01	1.1E-06	4.1E+00	4.1E-05
SN121M	2.5E+00	2.5E-05	1.5E-04	1.5E-09	8.9E+01	8.9E-04	9.2E+01	9.2E-04
SN123	7.4E+00	7.4E-05	9.0E-04	9.0E-09	3.0E+00	3.0E-05	1.0E+01	1.0E-04
SN126	2.2E+01	2.2E-04	6.0E-03	6.0E-08	1.5E+03	1.5E-02	1.5E+03	1.5E-02
SR 89	9.4E+00	9.4E-05	1.8E-05	1.8E-10	2.4E-02	2.4E-07	9.4E+00	9.4E-05
SR 90	2.8E+02	2.8E-03	0.0E+00		0.0E+00		2.8E+02	2.8E-03
TB160	5.1E+00	5.1E-05	1.4E-01	1.4E-06	2.7E+02	2.7E-03	2.8E+02	2.8E-03
TC 98	5.1E+00	5.1E-05	1.8E-01	1.8E-06	3.5E+04	3.5E-01	3.5E+04	3.5E-01
TC 99	1.9E+00	1.9E-05	6.7E-08	6.7E-13	1.5E-02	1.5E-07	1.9E+00	1.9E-05
TE123	1.1E+00	1.1E-05	3.7E-04	3.7E-09	1.7E+02	1.7E-03	1.7E+02	1.7E-03
TE123M	2.0E+00	2.0E-05	1.8E-02	1.8E-07	6.9E+01	6.9E-04	7.1E+01	7.1E-04
TE125M	1.5E+00	1.5E-05	1.2E-03	1.2E-08	4.5E+00	4.5E-05	6.0E+00	6.0E-05
TE127	6.8E-02	6.8E-07	5.9E-04	5.9E-09	7.1E-03	7.1E-08	7.6E-02	7.6E-07
TE127M	4.5E+00	4.5E-05	4.1E-04	4.1E-09	4.5E+00	4.5E-05	9.0E+00	9.0E-05
TE129	1.8E-02	1.8E-07	6.1E-03	6.1E-08	9.8E-03	9.8E-08	3.4E-02	3.4E-07
TE129M	5.4E+00	5.4E-05	4.7E-03	4.7E-08	8.8E+00	8.8E-05	1.4E+01	1.4E-04
TM170	5.7E+00	5.7E-05	5.7E-04	5.7E-09	2.8E+00	2.8E-05	8.5E+00	8.5E-05
TM171	1.2E+00	1.2E-05	6.7E-05	6.7E-10	1.9E+00	1.9E-05	3.1E+00	3.1E-05

Table 4 NCRP Atmospheric Release (Inhalation, and Direct Plume and Ground Exposure) Screening Factors (continued)

Radionuclide	Inhalation Screening Factor		Plume Screening Factor		Ground Screening Factor		Total Atmospheric Screening Factor	
	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3	mrem per Bq/m^3	Sv per Bq/m^3
Y 90	2.1E+00	2.1E-05	0.0E+00		0.0E+00		2.1E+00	2.1E-05
Y 91	1.1E+01	1.1E-04	4.7E-04	4.7E-09	6.7E-01	6.7E-06	1.2E+01	1.2E-04
ZR 93	3.4E+01	3.4E-04	0.0E+00		0.0E+00		3.4E+01	3.4E-04
ZR 95	5.1E+00	5.1E-05	9.3E-02	9.3E-07	3.4E+02	3.4E-03	3.5E+02	3.5E-03

Table 5 Inventory of Radionuclides in Potential Waste Streams Generated Under and Advanced Nuclear Fuel Cycle

	Radionuclide Mass ($g/MTHM_{SNF}$)										Radionuclide Activity ($Ci/MTHM_{SNF}$)									
Time =>	1	5	10	25	50	100	500	1000	5000	10000	1	5	10	25	50	100	500	1000	5000	10000
Lanthanides																				
EU154	6.1E+01	4.4E+01	2.9E+01	8.8E+00	1.2E+00	2.1E-02	2.1E-16	6.6E-34	0.0E+00	0.0E+00	1.6E+03	1.2E+03	7.8E+02	2.3E+02	3.1E+01	5.5E-01	5.5E-15	1.7E-32	0.0E+00	0.0E+00
SM151	2.3E+01	2.2E+01	2.1E+01	1.9E+01	1.6E+01	1.1E+01	4.9E-01	1.0E-02	4.3E-16	8.1E-33	6.2E+03	6.0E+03	5.8E+03	5.2E+03	4.3E+03	2.9E+03	1.3E+02	2.8E+00	1.2E-13	2.2E-30
CE144	1.6E+02	4.4E+00	5.1E-02	8.1E-08	1.7E-17	7.9E-37	0.0E+00	0.0E+00	0.0E+00	0.0E+00	2.7E+04	7.7E+02	9.0E+00	1.4E-05	3.0E-15	1.4E-34	0.0E+00	0.0E+00	0.0E+00	0.0E+00
SM146	1.1E-02	1.3E-02	1.3E-02	1.4E-02	1.4E-02	1.4E-02	1.4E-02	1.4E-02	1.4E-02	1.4E-02	7.5E-01	8.3E-01	8.9E-01	9.4E-01	9.5E-01	9.5E-01	9.5E-01	9.5E-01	9.5E-01	9.5E-01
HO166M	3.5E-03	3.5E-03	3.5E-03	3.5E-03	3.4E-03	3.3E-03	2.6E-03	2.0E-03	2.0E-04	1.1E-05	1.9E+05	1.9E+05	1.9E+05	1.8E+05	1.8E+05	1.8E+05	1.4E+05	1.1E+05	1.0E+04	5.8E+02
EU150	4.9E-07	4.5E-07	4.1E-07	3.1E-07	1.9E-07	7.3E-08	3.3E-11	2.2E-15	0.0E+00	0.0E+00	1.6E-03	1.4E-03	1.3E-03	9.8E-04	6.1E-04	2.3E-04	1.0E-07	6.9E-12	0.0E+00	0.0E+00
TM171	1.1E-06	2.7E-07	4.4E-08	2.0E-10	2.4E-14	3.4E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00	8.6E+01	2.0E+01	3.3E+00	1.5E-02	1.8E-06	2.5E-14	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Transition Metals																				
ZR 93	8.1E+02	8.1E+02	8.1E+02	8.1E+02	8.1E+02	8.1E+02	8.1E+02	8.1E+02	8.1E+02	8.1E+02	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00
PD107	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01
SN126	4.3E+01	4.3E+01	4.3E+01	4.3E+01	4.3E+01	4.3E+01	4.3E+01	4.2E+01	4.1E+01	4.0E+01	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.1E+00
SE 79	9.0E+00	9.0E+00	9.0E+00	9.0E+00	8.9E+00	8.9E+00	8.9E+00	8.9E+00	8.5E+00	8.0E+00	6.2E-01	6.2E-01	6.2E-01	6.2E-01	6.2E-01	6.2E-01	6.2E-01	6.2E-01	5.9E-01	5.6E-01
SB125	1.5E+01	5.5E+00	1.6E+00	3.7E-02	7.0E-05	2.6E-10	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.5E+04	5.7E+03	1.6E+03	3.8E+01	7.3E-02	2.7E-07	0.0E+00	0.0E+00	0.0E+00	0.0E+00
RU106	8.0E+01	5.1E+00	1.6E-01	5.4E-06	1.8E-13	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.3E+09	8.5E+07	2.7E+06	9.1E+01	3.1E-06	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
TE123	1.6E-02	1.6E-02	1.6E-02	1.6E-02	1.6E-02	1.6E-02	1.6E-02	1.6E-02	1.6E-02	1.6E-02	3.5E-12	3.6E-12	3.6E-12	3.6E-12	3.6E-12	3.6E-12	3.6E-12	3.6E-12	3.6E-12	3.6E-12
SN121M	5.4E-03	5.1E-03	4.8E-03	3.9E-03	2.7E-03	1.4E-03	5.3E-06	5.2E-09	4.2E-33	0.0E+00	2.9E-01	2.7E-01	2.6E-01	2.1E-01	1.5E-01	7.3E-02	2.9E-04	2.8E-07	2.2E-31	0.0E+00
NB 93M	1.4E-03	2.8E-03	4.2E-03	6.9E-03	8.5E-03	9.1E-03	9.2E-03	9.2E-03	9.1E-03	9.1E-03	3.9E-01	8.0E-01	1.2E+00	1.9E+00	2.4E+00	2.6E+00	2.6E+00	2.6E+00	2.6E+00	2.6E+00
NB 94	1.2E-03	1.2E-03	1.2E-03	1.2E-03	1.2E-03	1.2E-03	1.2E-03	1.1E-03	9.9E-04	8.3E-04	2.2E-04	2.2E-04	2.2E-04	2.2E-04	2.2E-04	2.2E-04	2.2E-04	2.2E-04	1.9E-04	1.6E-04
SB126	2.0E-06	2.0E-06	2.0E-06	2.0E-06	2.0E-06	2.0E-06	2.0E-06	2.0E-06	2.0E-06	1.9E-06	1.7E-01	1.7E-01	1.7E-01	1.7E-01	1.7E-01	1.7E-01	1.7E-01	1.7E-01	1.6E-01	1.6E-01
SB126M	1.5E-08	1.5E-08	1.5E-08	1.5E-08	1.5E-08	1.5E-08	1.5E-08	1.5E-08	1.5E-08	1.4E-08	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.2E+02	1.2E+02	1.2E+02
NB 95	1.5E+00	2.2E-07	5.3E-16	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.1E+09	1.6E+02	3.8E-07	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Undissolved Solids																				
TC 99	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.8E+02	2.8E+02	4.8E+00	4.8E+00	4.8E+00	4.8E+00	4.8E+00	4.8E+00	4.8E+00	4.8E+00	4.8E+00	4.7E+00
PD107	9.0E+01	9.0E+01	9.0E+01	9.0E+01	9.0E+01	9.0E+01	9.0E+01	9.0E+01	9.0E+01	9.0E+01	4.6E-02	4.6E-02	4.6E-02	4.6E-02	4.6E-02	4.6E-02	4.6E-02	4.6E-02	4.6E-02	4.6E-02
RU106	2.7E+01	1.7E+00	5.4E-02	1.8E-06	6.2E-14	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	8.8E+04	5.6E+03	1.8E+02	6.0E-03	2.0E-10	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
TC 98	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.7E-03	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06
RH102	3.0E-04	1.1E-04	3.5E-05	9.7E-07	2.5E-09	1.6E-14	0.0E+00	0.0E+00	0.0E+00	0.0E+00	3.6E-01	1.4E-01	4.2E-02	1.2E-03	3.0E-06	1.9E-11	0.0E+00	0.0E+00	0.0E+00	0.0E+00
ZR 95	3.2E-01	4.3E-08	1.1E-16	1.8E-42	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	6.9E+03	9.2E-04	2.3E-12	3.9E-38	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
RU103	1.9E-02	1.2E-13	1.2E-27	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	6.2E+02	3.9E-09	4.0E-23	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cs/Sr																				
CS137	1.8E+03	1.6E+03	1.4E+03	1.0E+03	5.7E+02	1.8E+02	1.7E-02	1.7E-07	0.0E+00	0.0E+00	1.5E+05	1.4E+05	1.2E+05	8.8E+04	4.9E+04	1.6E+04	1.5E+00	1.5E-05	0.0E+00	0.0E+00
CS135	6.6E+02	6.6E+02	6.6E+02	6.6E+02	6.6E+02	6.6E+02	6.6E+02	6.6E+02	6.6E+02	6.6E+02	5.8E-01	5.8E-01	5.8E-01	5.8E-01	5.8E-01	5.8E-01	5.8E-01	5.8E-01	5.8E-01	5.8E-01
SR 90	7.7E+02	7.0E+02	6.2E+02	4.3E+02	2.4E+02	7.3E+01	5.3E-03	3.6E-08	0.0E+00	0.0E+00	1.1E+05	9.6E+04	8.5E+04	5.9E+04	3.3E+04	1.0E+04	7.3E-01	5.0E-06	0.0E+00	0.0E+00
CS134	1.6E+02	4.3E+01	8.0E+00	5.1E-02	1.2E-05	5.8E-13	0.0E+00	0.0E+00	0.0E+00	0.0E+00	2.1E+05	5.5E+04	1.0E+04	6.7E+01	1.5E-02	7.5E-10	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Tc																				
TC 99	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.4E+02	8.3E+02	1.5E+01	1.5E+01	1.5E+01	1.5E+01	1.5E+01	1.5E+01	1.5E+01	1.5E+01	1.4E+01	1.4E+01
TC 98	8.1E-03	8.1E-03	8.1E-03	8.1E-03	8.1E-03	8.1E-03	8.1E-03	8.1E-03	8.1E-03	8.1E-03	7.1E-06	7.1E-06	7.1E-06	7.1E-06	7.1E-06	7.1E-06	7.1E-06	7.1E-06	7.1E-06	7.1E-06
I																				
I129	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	2.7E+02	4.8E-02	4.8E-02	4.8E-02	4.8E-02	4.8E-02	4.8E-02	4.8E-02	4.8E-02	4.8E-02	4.8E-02
TRU (Unprocessed)																				
NP237	6.2E+02	6.2E+02	6.3E+02	6.5E+02	7.1E+02	8.3E+02	1.6E+03	2.0E+03	2.4E+03	2.4E+03	4.4E-01	4.4E-01	4.4E-01	4.6E-01	5.0E-01	5.9E-01	1.1E+00	1.4E+00	1.7E+00	1.7E+00
PU238	3.1E+02	3.1E+02	2.9E+02	2.6E+02	2.2E+02	1.5E+02	6.2E+00	1.3E-01	5.2E-11	6.5E-21	5.3E+03	5.3E+03	5.1E+03	4.5E+03	3.7E+03	2.5E+03	1.1E+02	2.2E+00	8.9E-10	1.1E-19
PU239	6.2E+03	6.2E+03	6.2E+03	6.2E+03	6.1E+03	6.1E+03	6.1E+03	6.0E+03	5.4E+03	4.7E+03	3.8E+02	3.8E+02	3.8E+02	3.8E+02	3.8E+02	3.8E+02	3.8E+02	3.7E+02	3.4E+02	2.9E+02
PU240	2.9E+03	2.9E+03	2.9E+03	3.0E+03	3.0E+03	3.0E+03	2.8E+03	2.7E+03	1.8E+03	1.0E+03	6.6E+02	6.6E+02	6.7E+02	6.7E+02	6.7E+02	6.7E+02	6.4E+02	6.1E+02	4.0E+02	2.4E+02
PU241	1.7E+03	1.4E+03	1.1E+03	5.3E+02	1.6E+02	1.4E+01	9.2E-03	8.8E-03	6.4E-03	4.2E-03	1.7E+05	1.4E+05	1.1E+05	5.5E+04	1.6E+04	1.5E+03	9.5E-01	9.1E-01	6.6E-01	4.4E-01
PU242	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.6E+02	8.5E+02	3.4E+00	3.4E+00	3.4E+00	3.4E+00	3.4E+00	3.4E+00	3.4E+00	3.4E+00	3.4E+00	3.3E+00
AM241	1.5E+02	4.4E+02	7.3E+02	1.3E+03	1.6E+03	1.6E+03	8.4E+02	3.8E+02	8.2E-01	1.3E-01	5.0E+02	1.5E+03	2.5E+03	4.3E+03	5.4E+03	5.5E+03	2.9E+03	1.3E+03	2.8E+00	4.5E-01
AM242M	8.5E-01	8.3E-01	8.2E-01	7.6E-01	6.8E-01	5.4E-01	8.7E-02	8.9E-03	1.1E-10	1.3E-20	8.9E+00	8.7E+00	8.5E+00	8.0E+00	7.1E+00	5.7E+00	9.2E-01	9.4E-02	1.1E-09	1.4E-19
AM243	2.0E+02	2.0E+02	2.0E+02	2.0E+02	2.0E+02	2.0E+02	1.9E+02	1.8E+02	1.2E+02	7.8E+01	4.0E+01	4.0E+01	4.0E+01	4.0E+01	3.9E+01	3.9E+01	3.8E+01	3.6E+01	2.5E+01	1.6E+01
CM243	7.5E-01	6.8E-01	6.0E-01	4.2E-01	2.3E-01	6.8E-02	4.0E-06	2.1E-11	0.0E+00	0.0E+00	3.9E+01	3.5E+01	3.1E+01	2.2E+01	1.2E+01	3.5E+00	2.1E-04	1.1E-09	0.0E+00	0.0E+00
CM244	8.3E+01	7.1E+01	5.9E+01	3.3E+01	1.3E+01	1.9E+00	4.2E-07	2.0E-15	0.0E+00	0.0E+00	6.7E+03	5.7E+03	4.7E+03	2.7E+03	1.0E+03	1.5E+02	3.4E-05	1.7E-13	0.0E+00	0.0E+00
TRU (99.9% recovered)																				
NP237	6.2E-01	6.2E-01	6.3E-01	6.5E-01	7.1E-01	8.3E-01	1.6E+00	2.0E+00	2.4E+00	2.4E+00	4.4E-04	4.4E-04	4.4E-04	4.6E-04	5.0E-04	5.9E-04	1.1E-03	1.4E-03	1.7E-03	1.7E-03
PU238	3.1E-01	3.1E-01	2.9E-01	2.6E-01	2.2E-01	1.5E-01	6.2E-03	1.3E-04	5.2E-14	6.5E-24	5.3E+00	5.3E+00	5.1E+00	4.5E+00	3.7E+00	2.5E+00	1.1E-01	2.2E-03	8.9E-13	1.1E-22
PU239	6.2E+00	6.2E+00	6.2E+00	6.2E+00	6.1E+00	6.1E+00	6.1E+00	6.0E+00	5.4E+00	4.7E+00	3.8E-01	3.8E-01	3.8E-01	3.8E-01	3.8E-01	3.8E-01	3.8E-01	3.7E-01	3.4E-01	2.9E-01
PU240	2.9E+00	2.9E+00	2.9E+00	3.0E+00	3.0E+00	3.0E+00	2.8E+00	2.7E+00	1.8E+00	1.0E+00	6.6E-01	6.6E-01	6.7E-01	6.7E-01	6.7E-01	6.7E-01	6.4E-01	6.1E-01	4.0E-01	2.4E-01
PU241	1.7E+00	1.4E+00	1.1E+00	5.3E-01	1.6E-01	1.4E-02	9.2E-06	8.8E-06	6.4E-06	4.2E-06	1.7E+02	1.4E+02	1.1E+02	5.5E+01	1.6E+01	1.5E+00	9.5E-04	9.1E-04	6.6E-04	4.4E-04
PU242	8.6E-01	8.6E-01	8.6E-01	8.6E-01	8.6E-01	8.6E-01	8.6E-01	8.6E-01	8.6E-01	8.5E-01	3.4E-03	3.4E-03	3.4E-03	3.4E-03	3.4E-03	3.4E-03	3.4E-03	3.4E-03	3.4E-03	3.3E-03
AM241	1.5E-01	4.4E-01	7.3E-01	1.3E+00	1.6E+00	1.6E+00	8.4E-01	3.8E-01	8.2E-04	1.3E-04	5.0E-01	1.5E+00	2.5E+00	4.3E+00	5.4E+00	5.5E+00	2.9E+00	1.3E+00	2.8E-03	4.5E-04
AM242M	8.5E-04	8.3E-04	8.2E-04	7.6E-04	6.8E-04	5.4E-04	8.7E-05	8.9E-06	1.1E-13	1.3E-23	8.9E-03	8.7E-03	8.5E-03	8.0E-03	7.1E-03	5.7E-03	9.2E-04	9.4E-05	1.1E-12	1.4E-22
AM243	2.0E-01	2.0E-01	2.0E-01	2.0E-01	2.0E-01	2.0E-01	1.9E-01	1.8E-01	1.2E-01	7.8E-02	4.0E-02	4.0E-02	4.0E-02	4.0E-02	3.9E-02	3.9E-02	3.8E-02	3.6E-02	2.5E-02	1.6E-02
CM243	7.5E-04	6.8E-04	6.0E-04	4.2E-04	2.3E-04	6.8E-05	4.0E-09	2.1E-14	0.0E+00	0.0E+00	3.9E-02	3.5E-02	3.1E-02	2.2E-02	1.2E-02	3.5E-03	2.1E-07	1.1E-12	0.0E+00	0.0E+00
CM244	8.3E-02	7.1E-02	5.9E-02	3.3E-02	1.3E-02	1.9E-03	4.2E-10	2.0E-18	0.0E+00	0.0E+00	6.7E+00	5.7E+00	4.7E+00	2.7E+00	1.0E+00	1.5E-01	3.4E-08	1.7E-16	0.0E+00	0.0E+00

	Ground Screening Dose, Water Only										Atmosphere Screening Dose, Inhalation, Plume, Ground Exposure Only					
Time =>	1	5	10	25	50	100	500	1000	5000	10000	1	5	10	25	50	100
Lanthanides																
EU154	0.33%	0.28%	0.21%	0.07%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%	0.16%	0.14%	0.10%	0.04%	0.00%	0.00%
SM151	0.11%	0.13%	0.15%	0.15%	0.13%	0.09%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
CE144	1.82%	0.06%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.03%	0.00%	0.00%	0.00%	0.00%	0.00%
SM146	0.00%	0.01%	0.01%	0.01%	0.01%	0.01%	0.01%	0.01%	0.14%	2.47%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
HO166M	62.45%	74.20%	84.29%	97.08%	99.73%	99.90%	99.98%	99.99%	99.86%	97.53%	65.04%	75.45%	85.18%	97.40%	99.88%	100.00%
EU150	35.12%	25.28%	15.33%	2.69%	0.12%	0.00%	0.00%	0.00%	0.00%	0.00%	34.78%	24.41%	14.72%	2.56%	0.11%	0.00%
TM171	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
Transition Metals																
ZR 93	0.00%	0.00%	0.00%	0.06%	2.27%	2.28%	2.30%	2.31%	2.37%	2.44%	0.00%	0.00%	0.00%	0.10%	3.29%	3.48%
PD107	0.00%	0.00%	0.00%	0.00%	0.20%	0.20%	0.20%	0.20%	0.20%	0.21%	0.00%	0.00%	0.00%	0.00%	0.02%	0.02%
SN126	0.00%	0.00%	0.00%	2.16%	85.39%	85.85%	86.29%	86.29%	86.29%	86.30%	0.00%	0.00%	0.00%	2.56%	85.60%	90.68%
SE 79	0.00%	0.00%	0.00%	0.22%	8.78%	8.82%	8.86%	8.84%	8.71%	8.55%	0.00%	0.00%	0.00%	0.00%	0.04%	0.05%
SB125	0.00%	0.00%	0.00%	2.21%	0.17%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%	0.11%	0.94%	84.52%	5.42%	0.00%
RU106	100.00%	100.00%	100.00%	95.26%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	55.43%	99.89%	99.06%	12.56%	0.00%	0.00%
TE123	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
SN121M	0.00%	0.00%	0.00%	0.04%	1.03%	0.52%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.03%	0.62%	0.33%
NB 93M	0.00%	0.00%	0.00%	0.04%	2.16%	2.32%	2.35%	2.36%	2.42%	2.50%	0.00%	0.00%	0.00%	0.05%	2.16%	2.44%
NB 94	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%	0.39%	0.42%
SB126	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.03%	1.05%	1.11%
SB126M	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.04%	1.35%	1.43%
NB 95	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	44.55%	0.00%	0.00%	0.00%	0.00%	0.00%
Undissolved Solids																
TC 99	0.27%	4.09%	57.02%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	0.00%	0.00%	0.05%	48.29%	97.53%	97.77%
PD107	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.67%	1.36%	1.36%
RU106	99.73%	95.91%	42.97%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	78.36%	99.80%	98.15%	3.10%	0.00%	0.00%
TC 98	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.43%	0.87%	0.87%
RH102	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.03%	0.19%	1.80%	47.50%	0.24%	0.00%
ZR 95	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	21.19%	0.00%	0.00%	0.00%	0.00%	0.00%
RU103	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.40%	0.00%	0.00%	0.00%	0.00%	0.00%
Cs/Sr																
CS137	3.10%	3.11%	3.13%	3.16%	3.21%	3.32%	4.33%	0.01%	0.00%	0.00%	62.21%	84.21%	94.83%	98.08%	98.15%	98.21%
CS135	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.30%	99.77%	100.00%	100.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
SR 90	96.67%	96.82%	96.86%	96.84%	96.79%	96.68%	95.37%	0.21%	0.00%	0.00%	1.21%	1.63%	1.83%	1.87%	1.84%	1.78%
CS134	0.23%	0.07%	0.01%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	36.57%	14.15%	3.33%	0.03%	0.00%	0.00%
Tc																
TC 99	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	99.12%	99.12%	99.12%	99.12%	99.12%	99.12%
TC 98	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.88%	0.88%	0.88%	0.88%	0.88%	0.88%
I																
I129	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%
TRU (Unprocessed)																
NP237	0.76%	0.75%	0.77%	0.83%	1.00%	1.45%	6.32%	11.09%	21.45%	27.53%	0.00%	0.00%	0.00%	0.00%	0.01%	0.01%
PU238	65.29%	63.94%	62.05%	57.66%	52.25%	43.26%	4.28%	0.12%	0.00%	0.00%	35.76%	35.59%	34.92%	33.16%	30.35%	24.52%
PU239	5.51%	5.47%	5.52%	5.77%	6.37%	7.81%	17.81%	23.92%	35.25%	39.56%	2.82%	2.85%	2.91%	3.10%	3.46%	4.14%
PU240	9.53%	9.50%	9.62%	10.13%	11.24%	13.77%	30.43%	39.27%	42.07%	31.81%	4.88%	4.94%	5.06%	5.45%	6.10%	7.30%
PU241	11.11%	9.11%	7.23%	3.67%	1.22%	0.13%	0.00%	0.00%	0.00%	0.00%	23.25%	19.36%	15.53%	8.07%	2.70%	0.29%
PU242	0.05%	0.05%	0.05%	0.05%	0.05%	0.07%	0.15%	0.21%	0.34%	0.43%	0.02%	0.02%	0.02%	0.03%	0.03%	0.03%
AM241	2.15%	6.36%	10.69%	19.36%	26.63%	33.00%	40.43%	24.68%	0.09%	0.02%	3.91%	11.77%	19.98%	36.99%	51.39%	62.16%
AM242M	0.10%	0.10%	0.10%	0.10%	0.10%	0.09%	0.03%	0.00%	0.00%	0.00%	0.07%	0.06%	0.06%	0.06%	0.06%	0.06%
AM243	0.17%	0.17%	0.17%	0.18%	0.20%	0.24%	0.54%	0.70%	0.78%	0.63%	0.33%	0.33%	0.34%	0.36%	0.40%	0.48%
CM243	0.04%	0.04%	0.03%	0.02%	0.01%	0.01%	0.00%	0.00%	0.00%	0.00%	0.22%	0.20%	0.18%	0.14%	0.08%	0.03%
CM244	5.30%	4.52%	3.77%	2.22%	0.94%	0.17%	0.00%	0.00%	0.00%	0.00%	28.71%	24.86%	20.96%	12.62%	5.40%	0.96%
TRU (99.9% recovered)																
NP237	0.76%	0.75%	0.77%	0.83%	1.00%	1.45%	6.32%	11.09%	21.45%	27.53%	0.00%	0.00%	0.00%	0.00%	0.01%	0.01%
PU238	65.29%	63.94%	62.05%	57.66%	52.25%	43.26%	4.28%	0.12%	0.00%	0.00%	35.76%	35.59%	34.92%	33.16%	29.72%	24.52%
PU239	5.51%	5.47%	5.52%	5.77%	6.37%	7.81%	17.81%	23.92%	35.25%	39.56%	2.82%	2.85%	2.91%	3.10%	3.39%	4.14%
PU240	9.53%	9.50%	9.62%	10.13%	11.24%	13.77%	30.43%	39.27%	42.07%	31.81%	4.88%	4.94%	5.06%	5.45%	5.98%	7.30%
PU241	11.11%	9.11%	7.23%	3.67%	1.22%	0.13%	0.00%	0.00%	0.00%	0.00%	23.25%	19.36%	15.53%	8.07%	2.64%	0.29%
PU242	0.05%	0.05%	0.05%	0.05%	0.05%	0.07%	0.15%	0.21%	0.34%	0.43%	0.02%	0.02%	0.02%	0.03%	0.03%	0.03%
AM241	2.15%	6.36%	10.69%	19.36%	26.63%	33.00%	40.43%	24.68%	0.09%	0.02%	3.91%	11.77%	19.98%	36.99%	50.33%	62.16%
AM242M	0.10%	0.10%	0.10%	0.10%	0.10%	0.09%	0.03%	0.00%	0.00%	0.00%	0.07%	0.06%	0.06%	0.06%	0.06%	0.06%
AM243	0.17%	0.17%	0.17%	0.18%	0.20%	0.24%	0.54%	0.70%	0.78%	0.63%	0.33%	0.33%	0.34%	0.36%	0.40%	0.48%
CM243	0.04%	0.04%	0.03%	0.02%	0.01%	0.01%	0.00%	0.00%	0.00%	0.00%	0.22%	0.20%	0.18%	0.14%	0.08%	0.03%
CM244	5.30%	4.52%	3.77%	2.22%	0.94%	0.17%	0.00%	0.00%	0.00%	0.00%	28.71%	24.86%	20.96%	12.62%	5.29%	0.96%

Notes: Radionuclides sorted in descending order based on ground screening importance at 10 years
Red highlight indicates those radionuclides that contribute over 0.1% of the total screening “dose” for the waste stream

Example Waste Form Inventory Development

The radionuclide inventories above were used to demonstrate the development of waste form specific radionuclide inventories. In this example it was assumed that lanthanides and lanthanides + TMFP would be isolated in a glass waste form, and Tc + UDS + TMFP would be isolated in a metallic alloy waste form. The isolation of Cs and Sr into the combined lanthanides+UDS+TMFP glass waste form was also considered.

The transuranic radionuclides (0.1 weight percent lost to the waste streams) were assumed to report to the lanthanide or lanthanide + transition metal fission product glass waste forms. It was assumed that a very small fraction (10^{-6}) would be lost to the Tc + UDS + transition metal fission product metallic waste form.

The volume of the canisters containing the glass and metallic waste forms were estimated from information provided in the Yucca Mountain Final Environmental Impact Statement at 0.95 m^3 and 0.2 m^3, respectively [Ref. 3, Tables A-26 and A-27]. The equivalent amount of processes waste that would be disposed in each waste canister, expressed in terms of $MTHM_{SNF}$, is given by:

$$MTHM_{SNF} = \frac{V_{WF}}{V_{SP}}$$

Where: V_{WF} = volume of the waste form (0.95m^3 for glass, 0.2 m^3 for metal alloy)
V_{SP} = specific volume of the waste form required to isolate the waste from one MTHM of processed SNF ($m^3/MTHM_{SNF}$)

Estimates for the specific volume of the waste forms were obtained from the GNEP IWMS [Ref. 1] and are shown in Table 6. Both low and high specific volumes were used to estimate the range in the amount of waste that would be disposed ($MTHM_{SNF}$) in each waste form, also shown in Table 6.

Table 6 Example Waste Form Loadings

	$V_{SP, Low}$[a] ($m^3/MTHM_{SNF}$)	$V_{SP, High}$[a] ($m^3/MTHM_{SNF}$)	$MTHM_{SNF, Low}$	$MTHM_{SNF, High}$
Ln	6.6×10^{-3}	1.8×10^{-2}	52.7	143.7
Ln/FP	6.4×10^{-2}	1.2×10^{-1}	7.9	14.8
Metallic	6.2×10^{-3}	1.4×10^{-2}	14.3	32.3

Notes:
[a]Global Nuclear Energy Partnership Integrated Waste Management Strategy, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008.
[b] $MTHM_{SNF, Low} = V_{WF} / V_{SP, High}$; $MTHM_{SNF, High} = V_{WF} / V_{SP, Low}$

The total mass of the waste form was estimated using the waste form volume (0.95m^3 for glass, 0.2 m^3 for metal alloy) and the density of the various waste forms. The densities of the potential waste forms were obtained from the GNEP IWMS [Ref. 1] and are shown in Table 7. The resultant waste form mass, again a range due to uncertainties in both the specific volume and density of the waste forms, are also shown in Table 7.

Table 7 Example Waste Form Mass

	$\rho_{WF,Low}$[a] ($MTHM_{WF}$ /m^3)	$\rho_{WF,High}$[a] ($MTHM_{WF}$ /m^3)	$Mass_{WF,\ Low}$ (MT)	$Mass_{WF,\ High}$ (MT)
Ln	3	4	2.8	3.8
Ln/FP	2.5	3.2	2.4	3.0
Metallic	7.6	8.2	1.5	1.6

Notes:
[a]Global Nuclear Energy Partnership Integrated Waste Management Strategy, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008.
[b] $Mass_{WF,\ Low} = V_{WF} \bullet \rho_{WF,Low}$; $Mass_{WF,\ High} = V_{WF} \bullet \rho_{WF,High}$

The resulting waste form specific radionuclide inventory (g), and radionuclide activity for each waste form are shown in Table 8. The transuranic concentration of each waste form is determined by summing the total transuranic activity and dividing it by the total mass of the waste form.

It can be seen that the glass waste forms have transuranic concentrations, expressed as nCi/g, that are over three orders of magnitude larger than the 100 nCi/g limit for class C low-level radioactive waste. It can also be seen that a very low amount of transuranic material reporting to the metallic waste form would likely result in the transuranic content exceeding the 100 nCi/g limit (although the Tc-99 content itself would render the waste beyond the class C limits).

Table 8 Results of Example Waste Form Inventory Calculation

	Radionuclide Mass (g/WF)										Radionuclide Activity (Ci/WF)									
Time =>	**1**	**5**	**10**	**25**	**50**	**100**	**500**	**1000**	**5000**	**10000**	**1**	**5**	**10**	**25**	**50**	**100**	**500**	**1000**	**5000**	**10000**
	Lanthanide, Low				MTHM$_{SNF-Equivalent}$/Canister			52.7			**Lanthanide, Low**				MTHM$_{SNF-Equivalent}$/Canister			52.7		
					Waste Form Weight (g)			2.8E+06							Waste Form Weight (g)			2.8E+06		
HO166M	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.7E-01	1.4E-01	1.0E-01	1.0E-02	5.7E-04	9.9E+06	9.9E+06	9.8E+06	9.7E+06	9.6E+06	9.3E+06	7.4E+06	5.5E+06	5.5E+05	3.1E+04
SM146	6.0E-01	6.6E-01	7.0E-01	7.5E-01	7.5E-01	7.5E-01	7.5E-01	7.5E-01	7.5E-01	7.5E-01	4.0E+01	4.4E+01	4.7E+01	4.9E+01	5.0E+01	5.0E+01	5.0E+01	5.0E+01	5.0E+01	5.0E+01
EU150	2.6E-05	2.4E-05	2.2E-05	1.6E-05	1.0E-05	3.8E-06	1.7E-09	1.1E-13	0.0E+00	0.0E+00	8.2E-02	7.6E-02	6.9E-02	5.2E-02	3.2E-02	1.2E-02	5.5E-06	3.6E-10	0.0E+00	0.0E+00
NP237	3.3E+01	3.3E+01	3.3E+01	3.4E+01	3.7E+01	4.4E+01	8.3E+01	1.1E+02	1.3E+02	1.3E+02	2.3E-02	2.3E-02	2.3E-02	2.4E-02	2.6E-02	3.1E-02	5.9E-02	7.6E-02	9.0E-02	8.9E-02
PU238	1.6E+01	1.6E+01	1.6E+01	1.4E+01	1.1E+01	7.6E+00	3.3E-01	6.7E-03	2.8E-12	3.4E-22	2.8E+02	2.8E+02	2.7E+02	2.4E+02	1.9E+02	1.3E+02	5.6E+00	1.2E-01	4.7E-11	5.9E-21
PU239	3.2E+02	3.2E+02	3.2E+02	3.2E+02	3.2E+02	3.2E+02	3.2E+02	3.2E+02	2.8E+02	2.5E+02	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	1.8E+01	1.5E+01
PU240	1.5E+02	1.5E+02	1.5E+02	1.6E+02	1.6E+02	1.6E+02	1.5E+02	1.4E+02	9.3E+01	5.5E+01	3.5E+01	3.5E+01	3.5E+01	3.5E+01	3.5E+01	3.5E+01	3.4E+01	3.2E+01	2.1E+01	1.2E+01
PU242	4.6E+01	4.6E+01	4.6E+01	4.6E+01	4.6E+01	4.6E+01	4.5E+01	4.5E+01	4.5E+01	4.5E+01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01
AM241	7.7E+00	2.3E+01	3.8E+01	6.7E+01	8.3E+01	8.4E+01	4.4E+01	2.0E+01	4.3E-02	7.0E-03	2.7E+01	7.9E+01	1.3E+02	2.3E+02	2.8E+02	2.9E+02	1.5E+02	6.9E+01	1.5E-01	2.4E-02
AM243	1.0E+01	1.0E+01	1.0E+01	1.0E+01	1.0E+01	1.0E+01	1.0E+01	9.5E+00	6.5E+00	4.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.0E+00	1.9E+00	1.3E+00	8.2E-01
CM244	4.4E+00	3.7E+00	3.1E+00	1.7E+00	6.7E-01	9.8E-02	2.2E-08	1.1E-16	0.0E+00	0.0E+00	3.5E+02	3.0E+02	2.5E+02	1.4E+02	5.4E+01	8.0E+00	1.8E-06	8.7E-15	0.0E+00	0.0E+00
															TRU Activity Concentration (nCi/g)					
											2.5E+05	2.5E+05	2.5E+05	2.3E+05	2.1E+05	1.7E+05	7.5E+04	4.3E+04	1.4E+04	1.0E+04
	Lanthanide, High				MTHM$_{SNF-Equivalent}$/Canister			143.7			**Lanthanide, High**				MTHM$_{SNF-Equivalent}$/Canister			143.7		
					Waste Form Weight (g)			3.8E+06							Waste Form Weight (g)			3.8E+06		
HO166M	5.0E-01	5.0E-01	5.0E-01	5.0E-01	4.9E-01	4.8E-01	3.8E-01	2.8E-01	2.8E-02	1.6E-03	2.7E+07	2.7E+07	2.7E+07	2.7E+07	2.6E+07	2.5E+07	2.0E+07	1.5E+07	1.5E+06	8.4E+04
SM146	1.6E+00	1.8E+00	1.9E+00	2.0E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	1.1E+02	1.2E+02	1.3E+02	1.3E+02	1.4E+02	1.4E+02	1.4E+02	1.4E+02	1.4E+02	1.4E+02
EU150	7.0E-05	6.5E-05	5.9E-05	4.4E-05	2.7E-05	1.0E-05	4.7E-09	3.1E-13	0.0E+00	0.0E+00	2.2E-01	2.1E-01	1.9E-01	1.4E-01	8.7E-02	3.3E-02	1.5E-05	9.9E-10	0.0E+00	0.0E+00
NP237	8.9E+01	8.9E+01	9.0E+01	9.3E+01	1.0E+02	1.2E+02	2.3E+02	2.9E+02	3.5E+02	3.5E+02	6.3E-02	6.3E-02	6.3E-02	6.6E-02	7.2E-02	8.5E-02	1.6E-01	2.1E-01	2.4E-01	2.4E-01
PU238	4.5E+01	4.4E+01	4.2E+01	3.8E+01	3.1E+01	2.1E+01	8.9E-01	1.8E-02	7.5E-12	9.4E-22	7.7E+02	7.5E+02	7.3E+02	6.4E+02	5.3E+02	3.6E+02	1.5E+01	3.1E-01	1.3E-10	1.6E-20
PU239	8.8E+02	8.8E+02	8.8E+02	8.8E+02	8.8E+02	8.8E+02	8.7E+02	8.6E+02	7.8E+02	6.8E+02	5.5E+01	5.5E+01	5.5E+01	5.5E+01	5.5E+01	5.5E+01	5.4E+01	5.3E+01	4.8E+01	4.2E+01
PU240	4.2E+02	4.2E+02	4.2E+02	4.2E+02	4.3E+02	4.2E+02	4.1E+02	3.9E+02	2.5E+02	1.5E+02	9.5E+01	9.5E+01	9.6E+01	9.6E+01	9.7E+01	9.7E+01	9.3E+01	8.8E+01	5.7E+01	3.4E+01
PU242	1.2E+02	1.2E+02	1.2E+02	1.2E+02	1.2E+02	1.2E+02	1.2E+02	1.2E+02	1.2E+02	1.2E+02	4.9E-01	4.9E-01	4.9E-01	4.9E-01	4.9E-01	4.9E-01	4.9E-01	4.9E-01	4.8E-01	4.8E-01
AM241	2.1E+01	6.3E+01	1.0E+02	1.8E+02	2.3E+02	2.3E+02	1.2E+02	5.4E+01	1.2E-01	1.9E-02	7.3E+01	2.2E+02	3.6E+02	6.2E+02	7.8E+02	7.8E+02	4.2E+02	1.9E+02	4.0E-01	6.5E-02
AM243	2.9E+01	2.9E+01	2.9E+01	2.8E+01	2.8E+01	2.8E+01	2.7E+01	2.6E+01	1.8E+01	1.1E+01	5.7E+00	5.7E+00	5.7E+00	5.7E+00	5.7E+00	5.6E+00	5.4E+00	5.2E+00	3.6E+00	2.2E+00
CM244	1.2E+01	1.0E+01	8.4E+00	4.7E+00	1.8E+00	2.7E-01	6.0E-08	2.9E-16	0.0E+00	0.0E+00	9.6E+02	8.2E+02	6.8E+02	3.8E+02	1.5E+02	2.2E+01	4.9E-06	2.4E-14	0.0E+00	0.0E+00
															TRU Activity Concentration (nCi/g)					
											5.2E+05	5.2E+05	5.1E+05	4.8E+05	4.3E+05	3.5E+05	1.6E+05	8.9E+04	2.9E+04	2.1E+04
	Lanthanide+FP (& Cs/SR), Low				MTHM$_{SNF-Equivalent}$/Canister			7.9			**Lanthanide+FP (& Cs/SR), Low**				MTHM$_{SNF-Equivalent}$/Canister			7.9		
					Waste Form Weight (g)			2.4E+06							Waste Form Weight (g)			2.4E+06		
HO166M	2.8E-02	2.8E-02	2.8E-02	2.7E-02	2.7E-02	2.6E-02	2.1E-02	1.6E-02	1.5E-03	8.6E-05	1.5E+06	1.5E+06	1.5E+06	1.5E+06	1.4E+06	1.4E+06	1.1E+06	8.3E+05	8.3E+04	4.6E+03
SM146	9.0E-02	9.9E-02	1.1E-01	1.1E-01	1.1E-01	1.1E-01	1.1E-01	1.1E-01	1.1E-01	1.1E-01	6.0E+00	6.6E+00	7.0E+00	7.4E+00	7.5E+00	7.5E+00	7.5E+00	7.5E+00	7.5E+00	7.5E+00
EU150	3.9E-06	3.6E-06	3.2E-06	2.4E-06	1.5E-06	5.7E-07	2.6E-10	1.7E-14	0.0E+00	0.0E+00	1.2E-02	1.1E-02	1.0E-02	7.7E-03	4.8E-03	1.8E-03	8.3E-07	5.4E-11	0.0E+00	0.0E+00
NP237	4.9E+00	4.9E+00	4.9E+00	5.1E+00	5.6E+00	6.6E+00	1.2E+01	1.6E+01	1.9E+01	1.9E+01	3.5E-03	3.5E-03	3.5E-03	3.6E-03	3.9E-03	4.7E-03	8.8E-03	1.1E-02	1.3E-02	1.3E-02
PU238	2.5E+00	2.4E+00	2.3E+00	2.1E+00	1.7E+00	1.1E+00	4.9E-02	1.0E-03	4.1E-13	5.2E-23	4.2E+01	4.2E+01	4.0E+01	3.5E+01	2.9E+01	2.0E+01	8.4E-01	1.7E-02	7.1E-12	8.8E-22
PU239	4.9E+01	4.9E+01	4.9E+01	4.9E+01	4.9E+01	4.9E+01	4.8E+01	4.7E+01	4.3E+01	3.7E+01	3.0E+00	3.0E+00	3.0E+00	3.0E+00	3.0E+00	3.0E+00	3.0E+00	2.9E+00	2.6E+00	2.3E+00
PU240	2.3E+01	2.3E+01	2.3E+01	2.3E+01	2.3E+01	2.3E+01	2.2E+01	2.1E+01	1.4E+01	8.2E+00	5.2E+00	5.2E+00	5.3E+00	5.3E+00	5.3E+00	5.3E+00	5.1E+00	4.8E+00	3.2E+00	1.9E+00
PU242	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.7E+00	2.7E-02	2.7E-02	2.7E-02	2.7E-02	2.7E-02	2.7E-02	2.7E-02	2.7E-02	2.7E-02	2.6E-02
AM241	1.2E+00	3.5E+00	5.8E+00	1.0E+01	1.2E+01	1.3E+01	6.7E+00	3.0E+00	6.5E-03	1.0E-03	4.0E+00	1.2E+01	2.0E+01	3.4E+01	4.3E+01	4.3E+01	2.3E+01	1.0E+01	2.2E-02	3.6E-03
AM243	1.6E+00	1.6E+00	1.6E+00	1.6E+00	1.6E+00	1.6E+00	1.5E+00	1.4E+00	9.8E-01	6.1E-01	3.1E-01	3.1E-01	3.1E-01	3.1E-01	3.1E-01	3.1E-01	3.0E-01	2.9E-01	2.0E-01	1.2E-01
CM244	6.5E-01	5.6E-01	4.6E-01	2.6E-01	1.0E-01	1.5E-02	3.3E-09	1.6E-17	0.0E+00	0.0E+00	5.3E+01	4.5E+01	3.7E+01	2.1E+01	8.1E+00	1.2E+00	2.7E-07	1.3E-15	0.0E+00	0.0E+00
ZR 93	6.4E+03	6.4E+03	6.4E+03	6.4E+03	6.4E+03	6.4E+03	6.4E+03	6.4E+03	6.4E+03	6.4E+03	1.6E+01	1.6E+01	1.6E+01	1.6E+01	1.6E+01	1.6E+01	1.6E+01	1.6E+01	1.6E+01	1.6E+01
PD107	2.1E+03	2.1E+03	2.1E+03	2.1E+03	2.1E+03	2.1E+03	2.1E+03	2.1E+03	2.1E+03	2.1E+03	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
SN126	3.4E+02	3.4E+02	3.4E+02	3.4E+02	3.4E+02	3.4E+02	3.4E+02	3.4E+02	3.3E+02	3.2E+02	9.6E+00	9.6E+00	9.6E+00	9.6E+00	9.6E+00	9.6E+00	9.6E+00	9.5E+00	9.3E+00	8.9E+00
SE 79	7.1E+01	7.1E+01	7.1E+01	7.1E+01	7.1E+01	7.1E+01	7.0E+01	7.0E+01	6.7E+01	6.4E+01	4.9E+00	4.9E+00	4.9E+00	4.9E+00	4.9E+00	4.9E+00	4.9E+00	4.9E+00	4.7E+00	4.4E+00
NB 93M	1.1E-02	2.2E-02	3.4E-02	5.4E-02	6.7E-02	7.2E-02	7.2E-02	7.2E-02	7.2E-02	7.2E-02	3.1E+00	6.3E+00	9.5E+00	1.5E+01	1.9E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01
CS135	5.2E+03	5.2E+03	5.2E+03	5.2E+03	5.2E+03	5.2E+03	5.2E+03	5.2E+03	5.2E+03	5.2E+03	4.6E+00	4.6E+00	4.6E+00	4.6E+00	4.6E+00	4.6E+00	4.6E+00	4.6E+00	4.6E+00	4.6E+00
															TRU Activity Concentration (nCi/g)					
											4.5E+04	4.5E+04	4.5E+04	4.2E+04	3.7E+04	3.1E+04	1.4E+04	7.8E+03	2.6E+03	1.8E+03

Table 8 Results of Example Waste Form Inventory Calculation (continued)

	Radionuclide Mass (g/WF)										Radionuclide Activity (Ci/WF)									
Time =>	**1**	**5**	**10**	**25**	**50**	**100**	**500**	**1000**	**5000**	**10000**	**1**	**5**	**10**	**25**	**50**	**100**	**500**	**1000**	**5000**	**10000**
	Lanthanide+FP (& Cs/SR), High				MTHM$_{SNF-Equivalent}$/Canister			14.8			**Lanthanide+FP (& Cs/SR), High**				MTHM$_{SNF-Equivalent}$/Canister			14.8		
					Waste Form Weight (g)			3.0E+06							Waste Form Weight (g)			3.0E+06		
HO166M	5.2E-02	5.2E-02	5.2E-02	5.1E-02	5.1E-02	4.9E-02	3.9E-02	2.9E-02	2.9E-03	1.6E-04	2.8E+06	2.8E+06	2.8E+06	2.7E+06	2.7E+06	2.6E+06	2.1E+06	1.6E+06	1.5E+05	8.6E+03
SM146	1.7E-01	1.9E-01	2.0E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	1.1E+01	1.2E+01	1.3E+01	1.4E+01	1.4E+01	1.4E+01	1.4E+01	1.4E+01	1.4E+01	1.4E+01
EU150	7.2E-06	6.7E-06	6.1E-06	4.6E-06	2.8E-06	1.1E-06	4.9E-10	3.2E-14	0.0E+00	0.0E+00	2.3E-02	2.1E-02	1.9E-02	1.5E-02	9.0E-03	3.4E-03	1.5E-06	1.0E-10	0.0E+00	0.0E+00
NP237	9.2E+00	9.2E+00	9.3E+00	9.6E+00	1.0E+01	1.2E+01	2.3E+01	3.0E+01	3.6E+01	3.6E+01	6.5E-03	6.5E-03	6.5E-03	6.8E-03	7.4E-03	8.7E-03	1.7E-02	2.1E-02	2.5E-02	2.5E-02
PU238	4.6E+00	4.5E+00	4.4E+00	3.9E+00	3.2E+00	2.1E+00	9.2E-02	1.9E-03	7.7E-13	9.7E-23	7.9E+01	7.8E+01	7.5E+01	6.6E+01	5.5E+01	3.7E+01	1.6E+00	3.2E-02	1.3E-11	1.7E-21
PU239	9.1E+01	9.1E+01	9.1E+01	9.1E+01	9.1E+01	9.1E+01	9.0E+01	8.9E+01	8.0E+01	7.0E+01	5.7E+00	5.7E+00	5.7E+00	5.7E+00	5.7E+00	5.6E+00	5.6E+00	5.5E+00	5.0E+00	4.3E+00
PU240	4.3E+01	4.3E+01	4.3E+01	4.4E+01	4.4E+01	4.4E+01	4.2E+01	4.0E+01	2.6E+01	1.5E+01	9.8E+00	9.8E+00	9.9E+00	9.9E+00	1.0E+01	1.0E+01	9.5E+00	9.1E+00	5.9E+00	3.5E+00
PU242	1.3E+01	1.3E+01	1.3E+01	1.3E+01	1.3E+01	1.3E+01	1.3E+01	1.3E+01	1.3E+01	1.3E+01	5.0E-02	5.0E-02	5.0E-02	5.0E-02	5.0E-02	5.0E-02	5.0E-02	5.0E-02	5.0E-02	4.9E-02
AM241	2.2E+00	6.5E+00	1.1E+01	1.9E+01	2.3E+01	2.4E+01	1.3E+01	5.6E+00	1.2E-02	2.0E-03	7.5E+00	2.2E+01	3.7E+01	6.4E+01	8.0E+01	8.1E+01	4.3E+01	1.9E+01	4.2E-02	6.7E-03
AM243	2.9E+00	2.9E+00	2.9E+00	2.9E+00	2.9E+00	2.9E+00	2.8E+00	2.7E+00	1.8E+00	1.2E+00	5.9E-01	5.9E-01	5.9E-01	5.9E-01	5.9E-01	5.8E-01	5.6E-01	5.4E-01	3.7E-01	2.3E-01
CM244	1.2E+00	1.0E+00	8.7E-01	4.9E-01	1.9E-01	2.8E-02	6.2E-09	3.0E-17	0.0E+00	0.0E+00	9.9E+01	8.5E+01	7.0E+01	4.0E+01	1.5E+01	2.2E+00	5.0E-07	2.5E-15	0.0E+00	0.0E+00
ZR 93	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	3.1E+01	3.1E+01	3.1E+01	3.1E+01	3.1E+01	3.1E+01	3.1E+01	3.1E+01	3.1E+01	3.1E+01
PD107	4.0E+03	4.0E+03	4.0E+03	4.0E+03	4.0E+03	4.0E+03	4.0E+03	4.0E+03	4.0E+03	4.0E+03	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00	2.1E+00
SN126	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.1E+02	5.9E+02	1.8E+01	1.8E+01	1.8E+01	1.8E+01	1.8E+01	1.8E+01	1.8E+01	1.8E+01	1.7E+01	1.7E+01
SE 79	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.2E+02	9.2E+00	9.2E+00	9.2E+00	9.2E+00	9.2E+00	9.2E+00	9.2E+00	9.1E+00	8.8E+00	8.3E+00
NB 93M	2.1E-02	4.2E-02	6.3E-02	1.0E-01	1.3E-01	1.3E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	5.8E+00	1.2E+01	1.8E+01	2.9E+01	3.6E+01	3.8E+01	3.8E+01	3.8E+01	3.8E+01	3.8E+01
CS135	9.8E+03	9.8E+03	9.8E+03	9.8E+03	9.8E+03	9.8E+03	9.8E+03	9.8E+03	9.8E+03	9.7E+03	8.6E+00	8.6E+00	8.6E+00	8.6E+00	8.6E+00	8.6E+00	8.6E+00	8.6E+00	8.6E+00	8.6E+00
											TRU Activity Concentration (nCi/g)									
											6.6E+04	6.6E+04	6.5E+04	6.1E+04	5.5E+04	4.5E+04	2.0E+04	1.1E+04	3.7E+03	2.7E+03
	TC/UDS/FP, Low				MTHM$_{SNF-Equivalent}$/Canister			14.3			**TC/UDS/FP, Low**				MTHM$_{SNF-Equivalent}$/Canister			14.3		
					Waste Form Weight (g)			1.5E+06							Waste Form Weight (g)			1.5E+06		
TC 99	1.6E+04	1.6E+04	1.6E+04	1.6E+04	1.6E+04	1.6E+04	1.6E+04	1.6E+04	1.6E+04	1.6E+04	2.8E+02	2.8E+02	2.8E+02	2.8E+02	2.8E+02	2.8E+02	2.8E+02	2.8E+02	2.7E+02	2.7E+02
ZR 93	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	1.2E+04	3.0E+01	3.0E+01	3.0E+01	3.0E+01	3.0E+01	3.0E+01	3.0E+01	3.0E+01	3.0E+01	3.0E+01
PD107	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.8E+03	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00	2.0E+00
SN126	6.1E+02	6.1E+02	6.1E+02	6.1E+02	6.1E+02	6.1E+02	6.1E+02	6.1E+02	5.9E+02	5.7E+02	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.7E+01	1.6E+01
SE 79	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.3E+02	1.2E+02	1.1E+02	8.9E+00	8.9E+00	8.9E+00	8.9E+00	8.9E+00	8.9E+00	8.9E+00	8.8E+00	8.4E+00	8.0E+00
NB 93M	2.0E-02	4.0E-02	6.1E-02	9.8E-02	1.2E-01	1.3E-01	1.3E-01	1.3E-01	1.3E-01	1.3E-01	5.6E+00	1.1E+01	1.7E+01	2.8E+01	3.4E+01	3.7E+01	3.7E+01	3.7E+01	3.7E+01	3.7E+01
	Actinides, assume 1 ppm (1e-6) lost to waste stream										Actinides, assume 1 ppm (1e-6) lost to waste stream									
NP237	8.8E-03	8.9E-03	8.9E-03	9.3E-03	1.0E-02	1.2E-02	2.3E-02	2.9E-02	3.4E-02	3.4E-02	6.2E-06	6.3E-06	6.3E-06	6.6E-06	7.1E-06	8.4E-06	1.6E-05	2.1E-05	2.4E-05	2.4E-05
PU238	4.4E-03	4.4E-03	4.2E-03	3.7E-03	3.1E-03	2.1E-03	8.9E-05	1.8E-06	7.5E-16	9.3E-26	7.6E-02	7.5E-02	7.2E-02	6.4E-02	5.3E-02	3.5E-02	1.5E-03	3.1E-05	1.3E-14	1.6E-24
PU239	8.8E-02	8.8E-02	8.8E-02	8.8E-02	8.8E-02	8.8E-02	8.7E-02	8.6E-02	7.7E-02	6.7E-02	5.5E-03	5.5E-03	5.5E-03	5.5E-03	5.5E-03	5.4E-03	5.4E-03	5.3E-03	4.8E-03	4.2E-03
PU240	4.2E-02	4.2E-02	4.2E-02	4.2E-02	4.2E-02	4.2E-02	4.1E-02	3.8E-02	2.5E-02	1.5E-02	9.4E-03	9.5E-03	9.5E-03	9.6E-03	9.6E-03	9.6E-03	9.2E-03	8.7E-03	5.7E-03	3.4E-03
PU242	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	4.9E-05	4.9E-05	4.9E-05	4.9E-05	4.9E-05	4.9E-05	4.8E-05	4.8E-05	4.8E-05	4.8E-05
AM241	2.1E-03	6.3E-03	1.0E-02	1.8E-02	2.2E-02	2.3E-02	1.2E-02	5.4E-03	1.2E-05	1.9E-06	7.2E-03	2.2E-02	3.6E-02	6.2E-02	7.7E-02	7.8E-02	4.1E-02	1.9E-02	4.0E-05	6.5E-06
AM243	2.8E-03	2.8E-03	2.8E-03	2.8E-03	2.8E-03	2.8E-03	2.7E-03	2.6E-03	1.8E-03	1.1E-03	5.7E-04	5.7E-04	5.7E-04	5.7E-04	5.6E-04	5.6E-04	5.4E-04	5.2E-04	3.5E-04	2.2E-04
CM244	1.2E-03	1.0E-03	8.4E-04	4.7E-04	1.8E-04	2.7E-05	6.0E-12	2.9E-20	0.0E+00	0.0E+00	9.6E-02	8.2E-02	6.8E-02	3.8E-02	1.5E-02	2.2E-03	4.8E-10	2.4E-18	0.0E+00	0.0E+00
											TRU Activity Concentration (nCi/g)									
											1.3E+02	1.3E+04	1.5E+04	1.8E+04	2.1E+04	2.1E+04	2.1E+04	2.1E+04	2.1E+04	2.1E+04

Table 8 Results of Example Waste Form Inventory Calculation (continued)

	Radionuclide Mass (g/WF)										Radionuclide Activity (Ci/WF)									
Time =>	**1**	**5**	**10**	**25**	**50**	**100**	**500**	**1000**	**5000**	**10000**	**1**	**5**	**10**	**25**	**50**	**100**	**500**	**1000**	**5000**	**10000**
	TC/UDS/FP, High				$MTHM_{SNF\text{-}Equivalent}$/Canister			32.3			**TC/UDS/FP, High**				$MTHM_{SNF\text{-}Equivalent}$/Canister			32.3		
					Waste Form Weight (g)			1.6E+06							Waste Form Weight (g)			1.6E+06		
TC 99	3.7E+04	3.7E+04	3.7E+04	3.7E+04	3.7E+04	3.7E+04	3.7E+04	3.7E+04	3.6E+04	3.6E+04	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.3E+02	6.2E+02	6.2E+02	6.2E+02	6.1E+02
ZR 93	2.6E+04	2.6E+04	2.6E+04	2.6E+04	2.6E+04	2.6E+04	2.6E+04	2.6E+04	2.6E+04	2.6E+04	6.7E+01	6.7E+01	6.7E+01	6.7E+01	6.7E+01	6.7E+01	6.7E+01	6.7E+01	6.7E+01	6.7E+01
PD107	8.7E+03	8.7E+03	8.7E+03	8.7E+03	8.7E+03	8.7E+03	8.7E+03	8.7E+03	8.7E+03	8.7E+03	4.5E+00	4.5E+00	4.5E+00	4.5E+00	4.5E+00	4.5E+00	4.5E+00	4.5E+00	4.5E+00	4.5E+00
SN126	1.4E+03	1.4E+03	1.4E+03	1.4E+03	1.4E+03	1.4E+03	1.4E+03	1.4E+03	1.3E+03	1.3E+03	3.9E+01	3.9E+01	3.9E+01	3.9E+01	3.9E+01	3.9E+01	3.9E+01	3.9E+01	3.8E+01	3.7E+01
SE 79	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.9E+02	2.7E+02	2.6E+02	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	2.0E+01	1.9E+01	1.8E+01
NB 93M	4.5E-02	9.1E-02	1.4E-01	2.2E-01	2.7E-01	2.9E-01	3.0E-01	3.0E-01	2.9E-01	2.9E-01	1.3E+01	2.6E+01	3.9E+01	6.3E+01	7.8E+01	8.3E+01	8.4E+01	8.4E+01	8.3E+01	8.3E+01
	Actinides, assume 1 ppm (1e-6) lost to waste stream										Actinides, assume 1 ppm (1e-6) lost to waste stream									
NP237	2.0E-02	2.0E-02	2.0E-02	2.1E-02	2.3E-02	2.7E-02	5.1E-02	6.6E-02	7.8E-02	7.8E-02	1.4E-05	1.4E-05	1.4E-05	1.5E-05	1.6E-05	1.9E-05	3.6E-05	4.6E-05	5.5E-05	5.5E-05
PU238	1.0E-02	9.9E-03	9.5E-03	8.4E-03	6.9E-03	4.7E-03	2.0E-04	4.1E-06	1.7E-15	2.1E-25	1.7E-01	1.7E-01	1.6E-01	1.4E-01	1.2E-01	8.0E-02	3.4E-03	7.1E-05	2.9E-14	3.6E-24
PU239	2.0E-01	2.0E-01	2.0E-01	2.0E-01	2.0E-01	2.0E-01	2.0E-01	1.9E-01	1.7E-01	1.5E-01	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.2E-02	1.1E-02	9.4E-03
PU240	9.4E-02	9.4E-02	9.5E-02	9.5E-02	9.6E-02	9.5E-02	9.1E-02	8.7E-02	5.7E-02	3.3E-02	2.1E-02	2.1E-02	2.1E-02	2.2E-02	2.2E-02	2.2E-02	2.1E-02	2.0E-02	1.3E-02	7.6E-03
PU242	2.8E-02	2.8E-02	2.8E-02	2.8E-02	2.8E-02	2.8E-02	2.8E-02	2.8E-02	2.8E-02	2.7E-02	1.1E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04
AM241	4.7E-03	1.4E-02	2.4E-02	4.1E-02	5.1E-02	5.1E-02	2.7E-02	1.2E-02	2.6E-05	4.3E-06	1.6E-02	4.9E-02	8.1E-02	1.4E-01	1.7E-01	1.8E-01	9.4E-02	4.2E-02	9.1E-05	1.5E-05
AM243	6.4E-03	6.4E-03	6.4E-03	6.4E-03	6.4E-03	6.3E-03	6.1E-03	5.8E-03	4.0E-03	2.5E-03	1.3E-03	1.3E-03	1.3E-03	1.3E-03	1.3E-03	1.3E-03	1.2E-03	1.2E-03	8.0E-04	5.0E-04
CM244	2.7E-03	2.3E-03	1.9E-03	1.1E-03	4.1E-04	6.0E-05	1.4E-11	6.6E-20	0.0E+00	0.0E+00	2.2E-01	1.9E-01	1.5E-01	8.6E-02	3.3E-02	4.9E-03	1.1E-09	5.3E-18	0.0E+00	0.0E+00
											TRU Activity Concentration (nCi/g)									
											2.7E+02	2.7E+02	2.6E+02	2.5E+02	2.2E+02	1.8E+02	8.0E+01	4.6E+01	1.5E+01	1.1E+01

1. Global Nuclear Energy Partnership Integrated Waste Management Strategy, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008.

2. Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground, National Council on Radiation Protection and Measurements, NCRP Report No. 123 I, January 1996.

3. Final 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, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, DOE/EIS-0250, February 2002.

2.2 M-TSD Summary for Idaho National Laboratory

General – INL Activities
INL funding was used primarily to develop the Integrated Waste Management Strategy, provide technical support to the GNEP Programmatic Environmental Impact Statement and evaluate industry submittals for implementing the GNEP strategy, and to ensure active collaboration with the Waste Forms and Separations Campaigns. All milestones were met on time and within budget.

Specific Activities:

Integrated Waste Management Strategy (IWMS)
In March the IWMS was published and was quickly adopted as the baseline for waste management calculations for the GNEP Programmatic Environmental Impact Statement (PEIS). The IWMS was truly a collaborative effort amongst the Systems Analysis, Waste Forms, and Separations Campaigns with contributing authors from laboratories across the DOE complex. The IWMS publication reference is:

Global Nuclear Energy Partnership Integrated Waste Management Strategy, D. Gombert, INL, et al, GNEP-WAST-WAST-AI-RT-2008-000214, March 2008

The IWMS incorporated text from supporting reports written by the Savannah River and Argonne National Laboratories which met supporting milestones:

1. Waste Management and Disposal Disposition Alternatives Evaluation Input, Mark Nutt, Argonne National Laboratory
2. Follow-on Engineering Alternative Studies Management of Decay Heat from Spent Nuclear Fuel, R. H. Jones, EAS-G-ESR-G-00069, March 2008

The IWMS also built on earlier collaborative work published in FY-07:

Global Nuclear Energy Partnership Integrated Waste Management Strategy Waste Treatment Baseline Study, D. Gombert, INL, et al, GNEP-WAST-AI-RT-2007-000324, September 2007
Global Nuclear Energy Partnership Integrated Waste Management Strategy Technical and Policy Issues for Implementation, D. Gombert and J. Roach, INL, et al, INL/EXT-07-12620, May 2007.

Abstract
The IWMS was developed considering the need for a sustainable nuclear fuel cycle effectively integrating waste management in a manner that can be commercialized and eventually adapted to be used internationally. It provides a logical basis for radioactive waste disposal on which potential changes to current programs and policies can be formulated and evaluated. Significant waste management efficiencies can be realized if the partitioned wastes can be dispositioned based on their characteristics rather than their origin. This IWMS is based on the premise that the NWPA can be clarified or amended to use a graded scale of waste management considering the actual risks of specific waste streams similar to the structure of chemical waste regulations currently accepted by the public. Such an alternative approach would allow for more efficient, and cost-effective, management and disposal of nuclear waste while safely protecting the public from unacceptable risk.
Advanced separations: 1) make possible recycling of long-lived hazardous elements as nuclear fuel so they can be transmuted into shorter-lived wastes, 2) allow greater flexibility in managing the individual waste streams based on the duration, type, and magnitude of risk, and 3) enable development of specialized waste forms to more effectively immobilize groups of radionuclides per this waste

management strategy. The IWMS proposes an approach that is consistent with the current regulatory framework established in 10 CFR 61 and considers characteristics intrinsic to each radioactive waste to address risk in the most efficient way. Wastes would no longer be classified by point of origin, but would be dispositioned based on potential health and environmental risk. Emphasis is also placed on energy recovery, recycle, and beneficial reuse, concepts that are clearly important to a sustainable energy future. In addition, this regulatory development would build on the best aspects of U.S. and international guidance on waste regulation (for example the International Atomic Energy Agency), and bring both into closer congruence in support of a global nuclear energy program with cooperative waste management.

Support to the PEIS
Throughout the year reviews have been conducted to verify calculations on waste forms and projected volumes under various scenarios described in the GNEP Programmatic Environmental Impact Statement. This work was supported by the Systems Analysis Campaign and added a significant opportunity to ensure consistency between the IWMS and the PEIS.

Industry Submittal Evaluation
Submittals by industrial consortia on how to implement the GNEP strategy for fuel reprocessing and deployment of fast reactors were reviewed in January 2008, with a follow-up review in July. INL personnel participated in both reviews and led in developing the overall management reports, defining critical issues and suggested areas of follow-on work.

Collaboration with Waste Forms Campaign
In addition to frequent involvement in Waste Forms Campaign meetings and collaboration on the IWMS, specific support was given to the 3rd Meeting of the GNEP Waste Management Working Group, December 10-14, 2007 with the Japan Atomic Energy Agency. The primary purpose of the trip was the 3rd Meeting of the GNEP Waste Management Group to further develop the outline for the Advanced Waste Management for Various Fuel Cycle Options Phase 1 Report. Presentations were made at the JAEA offices in Tokyo and text from the IWMS was adapted for inclusion into the Working Group Phase I report. Attendees also toured the Tokai R&D Center and the Mitzunami Underground Research Laboratory to see what research is being conducted and to discuss potential areas of mutually beneficial collaboration.

Collaboration with Waste Forms and Separations Campaigns
Specific support was given to the Waste and Separations Campaigns at the GNEP WP2 collaboration meeting between DOE and CEA to discuss areas of potentially mutually interesting work on waste treatment/characterization. The trip also provided opportunities to present a paper on the GNEP Waste Treatment Baseline and participate in the Atalante 2008 Conference and to tour and discuss waste issues at the Areva MELOX and La Hague facilities. The paper presented at Atalante was the first international presentation of DOE R&D plans for waste forms based on advanced separations. The WP2 meetings with CEA covered a fairly broad range of potential collaboration on capture of volatile radionuclides and advanced waste forms for metals, electrochemical salts, and the volatiles. Volatile radionuclides are currently released to the environment, but they will likely be captured in future reprocessing in the USA. Tours and discussions at MELOX and La Hague facilitated understanding of the current state of the art.

Developed preliminary trade-study on combined waste forms.
A preliminary trade-study was undertaken to evaluate some potential combinations of waste streams in common waste forms and identify issues that should be considered when comparing the benefits and costs of options. The premise is that partitioning used fuel into many streams and stabilizing the wastes into many forms maximizes the options for waste management, but at a cost of more complex operations, more supporting systems and facilities, and likely more secondary wastes. Equally credible is a scenario combining the waste streams based on target element chemistry, with all of the easily oxidized elements

stabilized as oxides in glass or ceramics, and all readily reducible elements combined in a metallic alloy. A spreadsheet-based trade study was initiated to determine if the cost-savings produced from reducing HLW volume would justify the additional costs incurred to reduce the volume. Cost impacts considered included: capital and operating costs of separation and waste form processes, decay storage, and waste disposal costs.

The analysis considered stabilizing Cs/Sr as a dedicated waste form and compared the costs of immobilizing the combined waste streams from FPEX and TALSPEAK into a silicate glass waste form and the combined wastes from UDS and Tc recovered from the UREX solution into a single alloy waste form. The TRUEX transition metal fission products (TMFP) could either be combined with the wastes immobilized in a glass or with the wastes immobilized in an alloy, and both options were considered.

The three options evaluated were:

- Three waste forms - Cs/Sr and Ln/TMFP* each in glass, Tc/UDS in metal alloy
- Two waste forms - Cs/Sr/Ln/TMFP in glass and Tc/UDS in metal alloy
- Two waste forms - Cs/Sr/Ln in glass and Tc/UDS/TMFP in metal alloy

*TMFP is transition metal fission product

Initial Results:

- The case with three waste forms allows the potential for disposal of the Cs/Sr waste form as LLW after sufficient decay to meet LLW limits, but the potential cost savings appear to be more than offset by the combined costs (capital and operating) of the initial Cs/Sr separation (FPEX) and the dedicated Cs/Sr waste form process.
- The third case with all of the readily reducible elements in a metal alloy significantly reduces total HLW volume, but due to constraints on the centerline temperature of the glass, the total number of HLW canisters is roughly the same as the second case with more glass generated, but poured into larger canisters.
- The expected costs to process the TRUEX waste (TMFP) into a metallic form offset the potential cost savings due to lower volume and fewer containers.
- Thus, based on the preliminary evaluation and the assumption that HLW disposal costs are primarily driven by the number of containers to be disposed (regardless of the size of the container up to a practical handling limit), the simplest case with the least processing (the middle option above) appears to be the least costly.

Proposed Follow-On Analyses

Cost of glass production should be further analyzed to incorporate natural inflection points in cost vs throughput; transition from an integrated hot-cell process to a dedicated facility may impact preliminary results. HLW disposal costs should be analyzed further to determine if costs are linear and purely a function of canister count. This analysis should be incorporated with a thermal management study to optimize repository impacts versus cost.

2.3 M-TSD Summary for Lawrence Livermore National Laboratory

Material Transportation, Storage & Disposal Leadership and General Program Support

LLNL provides leadership of the M-TSD sub-task. This includes coordinating work planning and reviewing products for five national laboratories and one university, providing monthly reporting and participation in periodic working group meetings.

In addition, LLNL staff provided support to other program components as needed:

- The M-TSD leader was asked to participate in a meetings held to evaluate as assignment to the Advanced Fuel Cycle Facility (AFCF) Project Team to evaluate the potential for achieving some of the AFCF mission objective using existing facilities. An effort to meet any significant fraction of the AFCF mission with distributed existing facilities would result in additional material transportation needs. Coordination with AFC-R&D Systems Material TSD was desired for consistency.
- M-TSD staff participated in Waste Campaign working group meetings to provide coordination of technical products between M-TSD, IWMS and the Waste Campaign. The Waste Campaign is using results from the Systems Campaign to define waste management criteria, and results from the IWMS that establish 'baseline' waste forms.
- M-TSD staff participated in the review of applications for the AFCI/GNEP University Fellowship Program for FY-08.

Technical Analysis Support – Packaging Survey Summary

During FY-2007, LLNL staff began a survey of existing packages for radioactive material storage and transport. There is an extensive range of existing packages to be reviewed for applicability and availability for GNEP applications. The Global Nuclear Energy Partnership (GNEP)[i] will require the use of Type A and Type B packages for the transportation and storage and disposition of nuclear materials and nuclear wastes.[ii] Many packages are currently certified by the Nuclear Regulatory Commission (NRC) under 10 CFR 71, Packaging and Transportation of Nuclear Material, for the transportation of un-irradiated and irradiated nuclear fuel in addition to specific separated fission products such as ^{90}Sr and ^{137}Cs. A number of these packages have been identified using the RAMPAC (Radioactive Material Packaging) Website [http://www.rampac.com/], maintained for EM-60's (Office of Safety Management and Operations) Packaging Certification Program by the Eagle Research Group for the Department of Energy (DOE).

As funding to continue this work was unavailable in FY-2008, the initial results were revised and package information updated with carryover funds. The resulting information is attached as Appendix-B to this report ("*A Survey of Potentially Useful Packages for Storage, Transportation, and Disposition of Un-irradiated, Irradiated, and Source Materials for the Global Nuclear Energy Partnership (GNEP)*", M. West, Lawrence Livermore National Laboratory, October, 2007).

Technical Analysis Support - DSARR

LLNL M-TSD staff prepared input for DSARR (Dynamic Systems Analysis Report for Nuclear Fuel Recycle) report. The DSARR is an INL led report to examine the time-dependent dynamics for a transition from the current open fuel cycle to either a 1-tier or 2-tier closed fuel cycle. Section 5.3 Waste Management Impacts was provided to INL for incorporation into the DSARR *"Dynamic Systems Analysis Report for Nuclear Fuel Recycle"*, GNEP-SYSA-AI-SS-RT-2008-000264, May, 2008.

International Cooperation

M-TSD staff participated in a series of meetings of the US-Japan GNEP Working Group on Waste Management. The general outline and annotated content for the first deliverable of the group was

developed in meetings held at JAEA in Tokyo December 10-12. On December 13 the group toured experimental facilities for waste management research at Tokai, and on December 14 toured the underground research laboratory for geologic disposal science at Mizunami.

Public Outreach Activities:

Program staff participated in a number of external events to provide insights on the potential benefits from development of advanced nuclear energy technologies:

- An invited talk was given on 'Sustainable Nuclear Energy" at San Jose State University on October 9, 2007 as part of a "Seminar Series on Energy Alternatives".
- A panel session was co-chaired on Nuclear Energy at the California Clean Innovation – 2008 Conference held May 9 at UCLA, with nuclear energy included in the agenda for the first time. The panel included representatives from industry, academia and professional organizations. CACI brings together business leaders, technology experts, entrepreneurs and venture capitalists to seek innovative opportunities in energy and environmental development.
- A panel session was co-chaired on Fuel Cycle and Waste Management Technology at the University of California Office of the President Asia Pacific Forum on Integration of Sustainability, Safety and Security of Nuclear Technology held June 12-13 at UC-Berkeley and LBNL. This forum brought together representatives from industry, academia, research labs, government and NGOs from the US and Asia to identify opportunities for the development, deployment and life cycle use of nuclear energy / nuclear technology that better meets societal goals for safety, security, sustainability, environmental protection and economics.
- Nuclear energy was represented in a 45 minute 'current events' program on public radio station KPBS (San Diego) September 9, 2008. This program addressed the forces behind the emerging renewal of nuclear energy.

2.4 M-TSD Summary for Sandia National Laboratory

Sandia National Laboratories has supported the GNEP Systems Analysis Campaign for the past three years in the areas of economic analyses, VISION code V&V, and systems analyses associated with packaging, storage, and transportation of nuclear and radioactive materials that may be part of the GNEP reprocessing nuclear fuel cycle. This later effort was performed as a member of a sub-group within the Systems Analysis Campaign, the Materials Transportation, Storage, and Disposal Working Group (M-TSD). This year, in addition to the systems analyses performed for M-TSD, support was provided to the writing of the Integrated Waste Management report and to review of the draft PEIS.

This summary provides an overview of all the Sandia support work that was done as part of the M-TSD Working Group. This report summarizes the work done for each milestone/deliverable that has been submitted, as well as work performed to support the review of the draft PEIS. This summary is presented in the following order:

1. Waste Streams, Alternatives and Proposed Disposition Report: M4 Deliverable—2/15/2008
2. Material Transportation, Storage and Disposal contribution for Secretarial Decision Package: M2 Deliverable – 3/14/2008
3. Submit input to LLNL for final GNEP Material Transportation, Storage and Disposal Analysis FY08 Summary Report: M4 Deliverable – 9/12/2008
4. PEIS Support

5. SAWG Annual Meeting- June 2008
6. Anticipated work for FY09

1. Waste Streams, Alternatives and Proposed Disposition Report

This effort focused on supporting the development and writing of the Integrated Waste Management Strategy (IWMS) Report. Potential impacts of material packaging, storage, and transportation on waste forms were identified and discussed. The culmination of this effort was input into the IWMS strategy report titled; "Global Nuclear Energy Partnership Integrated Waste Management Strategy", Dirk Gombert, et al., GNEP-WAST-AI-RT-2008-000214. For the transportation section, concluding statements with four main points were identified for the GNEP fuel cycle;

"GNEP will use existing technologies and packages to the fullest extent possible. The system for transporting radioactive materials is mature, with a robust regulatory infrastructure and over 50 years experience in package design, manufacture, testing, certification, use and maintenance. During the early developmental phases, it is likely that existing packages can be used to meet GNEP needs. However, GNEP technology is likely to result in waste streams that have not been accommodated under the current transportation infrastructure. Therefore, four types of actions will likely be required:

- acquire additional casks conforming to existing, certified designs;
- amend some certifications of existing casks to address specific new payloads;
- design new cask interior structures to stabilize and customize the fit of the payload; and
- develop new casks for specific types of wastes, such as Cs/Sr.

As the GNEP program develops and moves to commercial operations, there will be a need to optimize cask designs. This will create a need for the development and qualification of casks specific to the GNEP waste forms."

The M4 milestone due date was 2/15/2008. The final input was submitted on 2/12/2008.

2. Material Transportation, Storage and Disposal Contribution for Secretarial Decision Package

The focus of this M2 milestone was to comprehensively evaluate and discuss packaging, storage, and transportation for all potential nuclear and radioactive materials in the process and waste streams being considered by the GNEP program. In particular, a systems view was used to capture all packaging, storage, and transport operations needed to link the various functional aspects of the fuel cycle. Results of this evaluation were to be used to support the writing of the Dynamic Systems Analysis Report (DSAAR), which in turn, was to be used to support the June 2008 Secretarial Decision.

The final product of this effort was a draft GNEP report entitled: "Nuclear Materials Transportation and Storage Assessment", March 14, 2008. While this report went through a thorough 3rd party review at LANL, LLNL, and INL, it did not receive final review and approval from DOE. This was a result of postponement of the June 2008 Secretarial Decision.

Similar to the conclusions for the IWMS report, this report concludes that the transportation infrastructure is largely in-place for all process and waste streams that GNEP may develop. One major area that is evolving is the regulatory framework for materials that may fall into a security category (e.g., Cat I/II) that will require enhanced physical protection. Rule-making is currently underway for the transport of Radioactive Materials in Quantities of Concern (RAMQC). This area could have significant implication and impact on the packaging, storage, and transport of candidate materials.

As one basis for concluding that the transportation infrastructure is largely in-place, the report also identified material process and waste streams and provided suggestions for potential packaging options for each material.

The table from the report that lists these options follows:

Table 2.4-1 Potential Material Streams and Packaging Requirements.

Transportation Segment	Package Type	Archetypal Package*	MT HM per GW yr		
			Once-Through	Two-Tier	One-Tier
U Concentrates from Mine/Mill to Conversion	IP to A	Type 7a	195.8	121.1	121.8
UF6 from Conversion to Enrichment	A	Paducah Tiger	195.8	121.1	121.8
UF6 from Enrichment to LWR UOX Fuel Fabrication	A	UX-30	21.7	13.4	13.5
DU from Enrichment Plant to DU Storage	A	Paducah Tiger	174.1	107.7	108.3
LWR UOX Fuel from LWR Fuel Fabrication to LWR	A	SP-2, -3 (BWR) MCC-3, -4, -5 (PWR)	21.7	13.4	13.5
LWR UOX SNF to Repository	B	HISTAR	21.7	N/A	N/A
LWR UOX SNF to UOX Separations	B	HISTAR	N/A	13.4	13.5
LWR UOX U+TRU to LWR MOX Fuel Fabrication		N/A	N/A	1.37	N/A
LWR MOX Fuel from MOX Fabrication to LWR	B	HISTAR	N/A	1.37	N/A
LWR MOX SNF to MOX Separations	B	HISTAR	N/A	1.37	N/A
LWR UOX & MOX HLW to Repository (MT HLW)	B	HISTAR	N/A	3.37	2.89
LWR UOX & MOX Irradiated U to Waste	A to B	CHT-OP-TU	N/A	12.36	12.44
LWR U+TRU to FR Fuel Fabrication	B	9975	N/A	0.23	0.34
FR fuel from FR Fuel Fabrication to FR	- -	N/A	N/A	2.92	3.39
FR SNF to FR Fuel Separations	- -	N/A	N/A	2.92	3.39
FR U+TRU to FR Fuel Fabrication	- -	N/A	N/A	2.92	3.06
FR HLW to Repository (MT vitrified HLW)	B	HISTAR	N/A	0.91	1.35
Low Level Waste to Disposal (m^3)	IP to A	Standard Waste Box	115	100	97
GTCC Waste to Disposal (m^3)	B	RH-TRU 72-B	N/A	16	14

The "Archetypal" reference to the suggested packaging option refers to a package that could be a viable option, recognizing that the final material form, physical form, loading, etc. could all affect the final packaging decision. However, the table does provide a comprehensive perspective of all the steps in the advanced fuel cycle that will require packaging, storage, and transportation. The color coding in this table refers to the functional aspects of the three fuel cycles discussed in the report (i.e., once-through, two-tier, one-tier). The M4 milestone report was submitted on 3/14/2008.

3. Input to LLNL for final GNEP Material Transportation, Storage and Disposal Analysis FY08 Summary Report

This letter report satisfies the M4 milestone. The due date was 9/12/2008. The report was submitted on 2/10/2008.

4. PEIS Support

Sandia, as well as other national laboratories, contributed significant effort to support the internal review of the draft PEIS. In addition to random requests for review assistance, Sandia conducted a more formal review of the reactor accident, intentional destructive acts, and transportation sections of the draft PEIS this past February. The focus of the review was to evaluate accuracy and consistency of stated consequence estimates resulting from defined accident scenarios. This review covered the draft PEIS Summary Chapter, Chapters 4 and 11, and Appendices C, D, E, as well as the OUO Intentional Destructive Acts appendix. This review was conducted with Tetra Tech and with process oversight from Jeff Perry, DOE/ID.

SNL conducted additional reviews at the request of Headquarters in March and in May to address specific issues associated with results listed for external event consequences as well as verification of core inventory data used for the reactor accident consequence analyses.

The draft PEIS reviews by Sandia, as well as by other national laboratories in the DOE complex, have substantially improved the document and facilitated DOE's objective to distribute a technically rigorous PEIS for the public review and comment process. There were no scoped deliverables or milestone dates for this work.

5. Systems Analysis Working Group Annual Meeting- June 2008

Sandia supported the SAWG Annual Meeting on June 19, 2008 with a presentation of the work associated with the M-TSD work scope that had been done over the past year. The presentation covered the work performed and deliverables completed in FY08, a detailed overview of the Level II milestone report on transportation and storage, and an overview of proposed work for FY09 that extends the work that has been done to date.

6. Anticipated work for FY09

In order to meaningfully leverage the work completed over the past two years, it is recommended that a comprehensive systems analysis be performed for storage and transportation that extends the work initiated in the Level 2 milestone report. With the completion of the DSAAR and the Integrated Waste Management Report, waste form decisions are beginning to come into focus. This will allow the ability to extend the Level 2 milestone report to quantify material flows. Based on material flows, reasonable estimates can be made regarding types and quantities of casks that will be needed to support specific operations. This, in turn, will allow for conducting reasonable economic analyses to support development of a complete systems picture for packaging, storage, and transportation.

[a] GNEP—Basis Document, *Integrated Strategy for Nuclear Material Transportation, Storage, and Disposal Strategy Under the Global Nuclear Energy Partnership*, Review Draft 2—October 4, 2006. See Appendix A of this report.

[a] *Transportation and Storage Regulations Applicable to the GNEP*, Richard H. Yoshimura, Paul McConnell, Ken B. Sorenson, Sandia National Laboratory, January 31, 2007.

2.5 M-TSD Summary for Savannah River National Laboratory

SRNL Summary

In the Advanced Fuel Cycle Initiative program sponsored by the U.S. Department of Energy (DOE), numerous strategies involving reactors, fuel processing, and recycling are being examined to determine the impact on issues important to the viability of nuclear electricity generation, including the disposal of spent nuclear fuel and nuclear waste. As part of this program, studies are being performed to determine how processing spent nuclear fuel to separate certain elements, followed by transmutation of these elements and managing the wastes would benefit a geologic repository by altering the decay profile of the emplaced waste.

At the Global Nuclear Energy Partnership (GNEP) Heat Management Strategy meeting held in Denver, Colorado, November 12-14, 2007, assumptions were defined and options determined for managing the near-term decay heat associated with Cs and Sr in spent nuclear fuel (SNF) reprocessing wastes. This meeting was jointly sponsored by the Idaho National Laboratory (INL) and the Savannah River National Laboratory (SRNL) under work package SR0815060405 "Material Storage, Transportation and Disposal – SRNL". As a result of the meeting, several general options for managing Cs/Sr decay heat in SNF were identified based on prior work performed in support of the Yucca Mountain repository. The options identified provide varying degrees of benefit to the repository in terms of thermal performance.

The constituents in spent nuclear fuel that primarily contribute to the heat loading of the repository are the actinides plutonium (Pu), americium (Am) and curium (Cm) and fission products cesium (Cs) and strontium (Sr). Heat management models developed for the Yucca Mountain repository have shown that removal of Pu and Am alone from spent PWR fuel has the potential for either increasing the drift loading or reducing the size of a repository by a factor of 4.3 to 5.4. Combining this with removal of Cs and Sr allows for much greater reductions in size, upwards of a factor of 40. Further separation of Cm would provide for even greater reductions, up to a factor of 225 compared with direct disposal.

This study identifies considerations for future cost analyses for managing the disposal of spent nuclear fuel or the high level wastes resulting from reprocessing of spent nuclear fuel. Several cases are considered that provide varying degrees of benefit to the repository in terms of thermal performance. This study does not determine the costs for the various cases but identifies the major cost drivers for each case. The cases considered in this study are as follows:

Case 1: Direct Disposal 25 Years after Discharge (Reference Case)
Case 2: Direct Disposal after Extended Interim Storage for Decay
Case 3: Reprocess SNF Early To Remove Plutonium and Limit In-Growth of Am
Case 4: Limited Reprocessing with Delayed Emplacement
Case 5: Limited Reprocessing – 99.9% Pu/Am Recovery
Case 6: Reprocessing – 99.9% Pu/Am Recovery and Interim Storage for Decay
Case 7: Reprocessing – 99.9% Pu, Am, Cs and Sr Recovery and Interim Storage of Cs/Sr Waste (At Reprocessing Site)
Case 8: Reprocessing – 99.9% Pu, Am, Cs and Sr Recovery and Interim Storage of Cs/Sr Waste (At Centralized Site)
Case 9: Reprocessing – 99.9% TRU, Cs and Sr Recovery and Interim Storage of Cs/Sr Waste

Since this study only identifies the major cost drivers associated with each case and does not provide a quantitative evaluation and comparison of the cases, only qualitative conclusions can be identified. Removal of the minor actinides (Pu, Am and Cm) and the short lived fission products (Cs and Sr) can have significant beneficial impacts to a geologic repository; however, achieving these benefits will

require increasingly more capital investment in process development, equipment development, infrastructure and facilities. The additional processing required to achieve the repository benefits will increase operating and maintenance costs over the life of the facilities. The additional interim storage of the wastes prior to emplacement in a geologic repository is a new concept and will result in additional operating and maintenance costs for the duration of the interim storage period. Decommissioning and demolition (D&D) costs at the end of facility life will be greater for the cases involving more involved separation of minor actinides and short lived fission products. Although there are some major differences between the cases, some aspects of the various cases considered should be similar in scope and cost such as transportation of the spent nuclear fuel either directly to the repository (Cases 1 and 2) or to a reprocessing plant (Cases 3 through 9) and transportation of recovered uranium and high level waste for those cases involving reprocessing (Cases 3 through 9).

The work at SRNL/SRS stopped after the L2 milestone was met in February as work was defunded in April to support other program priorities. The SRNL L2 deliverable was the report by Robert Jones, ""Management of Decay Heat from Spent Nuclear Fuel", GNEP-SVSA-PMO-MI-DV-2008-000180. The non-OUO-marked version was Rev. 2, transmitted 3/19/08.

2.6 M-TSD Summary for University of Nevada - Reno

Background

The objective of this research is to develop computational tools to analyze the thermal performance of transfer, storage and transport casks used for advanced cycle fuels and materials. This work is an extension of research funded previously by the DOE Office of Civilian Radioactive Waste Management (OCRWM) and focuses on Light Water Reactor (LWR) spent fuel. Expertise developed from this project will be used for Advanced Fuel Cycle Research and Development (AFCRD) fuels and materials once those designs are available.

Work on this research program started on July 1, 2006 (but it was not funded until August 27, 2006) and is scheduled for completion on December 31, 2008. Two projects were performed during this reporting period. Their goals were to develop and experimentally benchmark computational tools to accurately predict:

(1) Heat transfer to rail casks from large pool fires; and
(2) The temperature of LWR fuel cladding inside casks during normal and fire accident conditions.

Summery of Current Research Products

The primary products of this research are:

(1) Graduate and undergraduate students trained in thermal analysis of casks under normal and fire accident conditions, and
(2) Papers in peer-reviewed conference proceedings and archival journals

2.1 Graduate Students

Two Ph.D. candidates were funded by this program during the current reporting period. One plans to graduate in 2008, the other in 2009. Two Masters Degree students were also funded, one plans to graduate in 2008, and the other in 2009. This program also funded one undergraduate research assistant who will complete his BS degree in 2008. The students are:

1. Mr. P. Araya, Ph.D. 2008 (expected)
2. Mr. N. Chalasani, Ph.D. 2008 (expected)
3. Mr. M. del Valle, M.S. 2008 (expected)
4. Mr. K. Kamichetty, M.S. 2009 (expected)
5. Mr. T. Bullard, BS 2008 (expected)

In addition, Professor Esad S. Hadziselimovic from the University of Sarajevo, who was working with our research group on a Fulbright Foundation Fellowship (starting in September 2007) completed his visit in May 2008. He worked with the graduate students to develop expertise simulating massive objects in large fires. Mr. M.A. Kramer worked as a part time research associate and to analyze fire test data. Two masters degree students who were funded by this project graduated in earlier reporting periods.

Peer-Reviewed Papers

Two conference papers on this work were peer-reviewed and accepted for presentation at the 2008 ASME Pressure Vessel and Piping Conference. They are:

1. Kramer, M.A., del Valle, M.A., and Greiner, M., 2008, "Measurement and Uncertainty of Heat Flux to a Rail-Cask Size Pipe Calorimeter in a Pool Fire," PVP2008-61600, Proceedings of the 2008 ASME Pressure Vessels and Piping Conference, PVP2008, July 27-31, Chicago.
2. Araya, P., and Greiner, M., "CFD Simulations of an 8x8 Heated Rod Array inside of an Isothermal Enclosure filled with a Rarefied Gas," PVP2008-61582, Proceedings of the 2008 ASME Pressure Vessels and Piping Conference, PVP2008, July 27-31, Chicago.

Appendix A

Waste Management and Disposal Disposition Alternatives Evaluation Input

ANL - M506040201

1 Introduction

The recycling of spent nuclear fuel (SNF) as envisioned in the Global Nuclear Energy Partnership (GNEP) will result in waste streams and waste forms that differ from those generated under a once-through nuclear fuel cycle. The separation and recovery of uranium, transuranic radionuclides, and fission product radionuclides, as ultimately envisioned under the GNEP, also presents an opportunity to approach the management and disposal of nuclear wastes in a manner that also differs from the current once-through nuclear fuel cycle.

National policies and regulations for the management and disposal of nuclear waste have evolved based on either the once-through nuclear fuel cycle or reprocessing only with the intent of recovering uranium and plutonium (e.g., PUREX). This has resulted in the current waste classification system that includes SNF, high-level nuclear waste (HLW), and various classifications of low-level radioactive waste (LLRW). It has also resulted in the current waste disposal policies such as the geologic disposal of SNF and HLW and the near-surface disposal of LLRW along with the associated framework for regulating such disposal.

While it would be possible to manage and dispose wastes that would be generated under the GNEP under a policy and regulatory framework that is essentially identical to what is currently in place, a more optimal approach may exist. Such an alternative approach may allow for more efficient, and cost-effective, management and disposal of nuclear waste while safely protecting the public from risk.

The objective of this report is to evaluate potential disposal paths for wastes that would be generated under the GNEP. This report first summarizes the waste streams and waste forms that may be generated under the GNEP and presents possible disposal paths. The disposal systems that could potentially be utilized are then summarized. The current disposal capacity and the status in developing disposal capacity for the various waste types with the United States are then discussed. The current policy and regulatory framework regarding nuclear waste disposal within the United States is then discussed. Potential changes to this policy and regulatory framework that could help optimize the management and disposal of GNEP wastes are then presented.

2 Summary of Potential GNEP Waste Streams & Possible Disposal Paths

The current policy and regulatory framework within the United States, in particular the waste classification system, dictates the disposal pathway for nuclear waste. Under the current framework, SNF and HLW must be disposed in a geologic repository, Class A, B, and C LLRW can be disposed in near-surface facilities and the disposal of Greater than Class C (GTCC) LLRW is not defined on a routine basis (perhaps geologic or near-surface). Changing the definition, or the classification, of nuclear waste would allow for disposal pathways that would depend more on the overall risk of the waste being disposed rather than how or where it was generated. This would allow for the utilization of different disposal pathways for GNEP wastes that could result in a more efficient waste management and disposal system within the United States.

This section presents a matrix of potential disposal pathways under both the current policy and regulatory framework in the United States and a revised framework where the waste classification system is revised.

This matrix considers wastes that would be generated in the co-extraction (COEX), Uranium Extraction (UREX) and Electro-Chemical (ECHEM) processes.

2.1 Waste Classification

The primary factor in determining a disposal pathway is the classification of the waste. Current United States policy and regulations define HLW and four classes of LLRW.

2.1.1 Definition of High Level Waste

The definition of HLW was developed in 1982 based on SNF reprocessing technologies present at that time, in particular the PUREX process. Only plutonium and uranium are recovered in the PUREX process with all other transuranic and the vast majority of the fission product elements remaining as waste. The resultant waste form remained extremely hazardous for a very long period of time, presenting a large risk to the public. As such, the Nuclear Waste Policy Act defines the term high-level radioactive waste as [Ref. 1]:

(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and
(B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

Regulations enacted by the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission (NRC) for the geologic disposal of SNF and HLW include either exact or similar definitions, as shown in Table 1.

Table 1. Definition of HLW in Current U.S. Regulations

Regulatory Agency	Regulation	Section	High Level Waste Definition
EPA	40 CFR 191: ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR MANAGEMENT AND DISPOSAL OF SPENT NUCLEAR FUEL, HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTES	191.02h	As defined in the Nuclear Waste Policy Act of 1982 (Pub. L. 97-425).

	40 CFR 197: PUBLIC HEALTH AND ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR YUCCA MOUNTAIN, NEVADA	197.2	1) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and 2) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.
NRC	10 CFR 60: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN GEOLOGIC REPOSITORIES	60.2	1) Irradiated reactor fuel, 2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and 3) solids into which such liquid wastes have been converted.
	10 CFR 63: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN A GEOLOGIC REPOSITORY AT YUCCA MOUNTAIN, NEVADA	197.2	1) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; 2) Irradiated reactor fuel; and 3) Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

2.1.2 Definition and Classification of Low Level Radioactive Waste

The Low-Level Radioactive Waste Policy Act (LLRWPA) defines low-level radioactive wastes as radioactive waste that [Ref. 2]:

(A) is not high-level radioactive waste, spent nuclear fuel, or byproduct material ; and
(B) the Nuclear Regulatory Commission, consistent with existing law and in accordance with paragraph (A) [immediately above], classifies as low-level radioactive waste.

The NRC classifies LLRW into one of four categories at 10 CFR 61.55: Classes A, B, C, and beyond Class C. The bases for the classifications are described at 10 CFR 61.7 and pertain to protection of the general population from releases of radioactivity, stability of the waste, and protection of individuals from inadvertent human intrusion.

Class C wastes are differentiated on the protection of individuals from inadvertent human intrusion. 10 CFR 61.7(b) states:

> (at item 4) "Institutional control of access to the site is required for up to 100 years. This permits the disposal of Class A and Class B waste without special provisions for intrusion protection, since these classes of waste contain types and quantities of radioisotopes that will decay during the 100-year period and will present an acceptable hazard to an intruder."
>
> (at item 5) "Waste that will not decay to levels which present an acceptable hazard to an intruder within 100 years is designated as Class C waste." Further, "Waste with concentrations above these limits is generally unacceptable for near-surface disposal. There may be some instances where waste with concentrations greater than permitted for Class C would be acceptable for near-surface disposal with special processing or design. These will be evaluated on a case-by-case basis."

Class A and B wastes are differentiated primarily on stability. As discussed at 10 CFR 61.7(b)(2), stability is a cornerstone of the disposal system, stating "stability of the waste and the disposal site so that once emplaced and covered, the access of water to the waste can be minimized. Migration of radionuclides is thus minimized, long-term active maintenance can be avoided, and potential exposures to intruders reduced." Class A wastes do not have sufficient amounts of radionuclides to be of great concern from a stability aspect and are not required to meet stability criteria. Class B and C waste forms should be designed to be stable (i.e., maintain gross physical properties and identity) over 300 years. Class A waste forms must be segregated from Class B and C wastes unless they meet the stability criteria at 10 CFR 61.56(b).

10 CFR 61.55(a)(2)(iv) states that waste that is not generally acceptable for near-surface disposal, greater than Class C (GTCC) must utilize waste forms and disposal methods that are different, and in more general more stringent, than for Class A, B, and C low-level wastes. The section further states that such wastes must be disposed of in a geologic repository as defined in 10 CFR 60 or 10 CFR 63 unless proposals for disposal of such waste in a disposal site licensed pursuant to the requirements in 10 CFR 61 are approved by the NRC.

2.1.3 Classification System Proposed by the International Atomic Energy Agency

In 1981 the International Atomic Energy Agency proposed classifying wastes into one of three classes as discussed below [Ref. 3].

> *High level waste*: (i) The highly radioactive liquid, containing mainly fission products, as well as some actinides, which is separated during chemical reprocessing of irradiated fuel (aqueous waste from the first solvent extraction cycle and those waste streams combined with it), (ii) Any other waste with radioactivity levels intense enough to generate significant quantities of heat by the radioactive decay process, (iii) Spent reactor fuel, if it is declared a waste.

Intermediate level waste (medium level waste): Waste which, because of its radionuclide content requires shielding but needs little or no provision for heat dissipation during its handling and transportation.

Low level waste: Waste which, because of its low radionuclide content, does not require shielding during normal handling and transportation.

The IAEA also differentiated between short and long lived waste, as well as alpha bearing waste within the intermediate and low level classifications. In those classifications, short lived waste and long-lived radioactive waste are differentiated based on the activity level of the waste decaying to acceptable levels during the time which administrative controls can be expected to last. Alpha bearing wastes were defined as containing one or more alpha emitting radionuclides, usually actinides, in quantities above acceptable limits established by a national regulatory body.

The current U.S. waste classifications/definitions match very closely with the classification proposed by the IAEA in 1981. In particular, the definition of HLW is explicitly tied to reprocessing wastes and GTCC LLW matches well with the IAEA classification of intermediate level waste. The U.S. LLRW class A, B, and C wastes also match classifications according to acceptable levels of radioactivity during the time which administrative controls can be expected to last.

In 1994 the IAEA revised their proposed waste classifications to address limitations indentified in their original classification approach [Ref. 4]. In particular, the IAEA identified that their original classification did not have a clear linkage to safety aspects of radioactive waste management, especially disposal, it lacked quantitative boundaries between classification, and it did not recognize a class of waste that contains so little radioactive material that it cannot be considered as 'radioactive.' The definitions under this revised classification and the waste characteristics within each of the proposed IAEA classification are shown in Table 2.

In general, the waste definitions and classifications under current U.S. policy and regulations are similar to the classification proposed by the IAEA. The intent of the definition of HLW is very similar, calling for a high degree of isolation for extremely radioactive wastes. However there is a key difference in that the proposed IAEA classification no longer explicitly links HLW to reprocessing (or recycling).

Table 2. Waste Classification System Proposed by the IAEA

Waste Classification	Waste Characteristics
High Level Waste	– Large concentrations of both short and long-lived radionuclides requiring a high degree of isolation from the biosphere – Generates significant quantities of heat, in excess of 2 kW/m^3 – Continues to generate heat for several centuries
Intermediate Level Waste	– Shielding Required: contact dose > 2 mSv/hr (200 mrem/hr) – Little or no heat removal provisions required

Short Lived	– Allowable activity limits depend on waste management option (disposal system) and properties of individual radionuclides disposed – Hazard can be significantly reduced during the period of institutional control – General limits ▪ < 4000 Bq/g (100 nCi/g) long-lived alpha emitters in individual waste packages ▪ average of < 400 Bq/g (10 nCi/g) long-lived alpha emitters in all waste packages – Need to also consider the concentration of long-lived beta and gamma emitters (i.e., ^{99}Tc and ^{129}I) within context of the disposal system
Long-Lived	– Exceeds limits for short lived
Low Level Waste	– No Shielding Required: contact dose ≤ 2 mSv/hr (200 mrem/hr) – Little or no heat removal provisions required
Short Lived	– Allowable activity limits depend on waste management option (disposal system) and properties of individual radionuclides disposed – Hazard can be significantly reduced during the period of institutional control – General limits ▪ < 4000 Bq/g (100 nCi/g) long-lived alpha emitters in individual waste packages ▪ average of < 400 Bq/g (10 nCi/g) long-lived alpha emitters in all waste packages – Need to also consider the concentration of long-lived beta and gamma emitters (i.e., ^{99}Tc and ^{129}I) within context of the disposal system
Long-Lived	– Exceeds limits for short lived
Exempt	– Annual dose to members of the public ≤ 0.01 mSv (1 mrem), consistent with guidelines in IAEA Safety Series No. 89 (Principles for the Exemption of Radiation Sources and Practices from Regulatory Control)

The IAEA classifications also match very well with the LLRW classification system used in the United States. As discussed above, this classification system is based on concentrations of short- and long-lived radionuclides, requiring that higher activity waste forms be more robust through the application of stability requirements (Class B and C) or disposed in a more robust facility (GTCC).

The IAEA also proposes an exempt classification where wastes might be exempted from regulatory control. A similar classification does not exist in the United States.

2.1.4 Assumed Revised U.S. Classification System Used In This Evaluation

A revised U.S. radioactive waste classification system is assumed in this evaluation for the purposes of identifying potential disposal pathways for GNEP wastes that could result in a more efficient waste management and disposal system within the United States. The revised waste classification system used in this evaluation is:

- The HLW classification proposed by the IAEA is assumed. Although this is primarily based on heat, the individual protection requirements assume above will also play a role. It is further assumed that HLW is not explicitly linked to any processing technique, but rather to the content and potential risk associated with the waste.
- The LLRW classifications in 10 CFR 61 are assumed.
- The exempt waste classification proposed by the IAEA is assumed.

In determining potential disposal pathways for the various GNEP waste forms it is also assumed that all disposal facilities (LLRW, GTCC LLRW, SNF/HLW) will be required to meet individual protection standards similar to those in current regulations. This requirement alone could potentially limit the choice of disposal pathways given radionuclide concentrations within waste forms and overall disposal facility inventories for a given disposal system. For example, a waste form may be classified as GTCC yet it may contain sufficient quantities of key radionuclides that warrant a greater degree of isolation offered by deep geologic disposal.

2.2 Potential Disposal Pathway Matrix

Three overall classes of disposal systems are defined for the purposes of this evaluation. Specific disposal systems within each of these classes is discussed further in this report. The disposal system classes and the waste classifications that could be disposed in each are shown schematically in Figure 1.

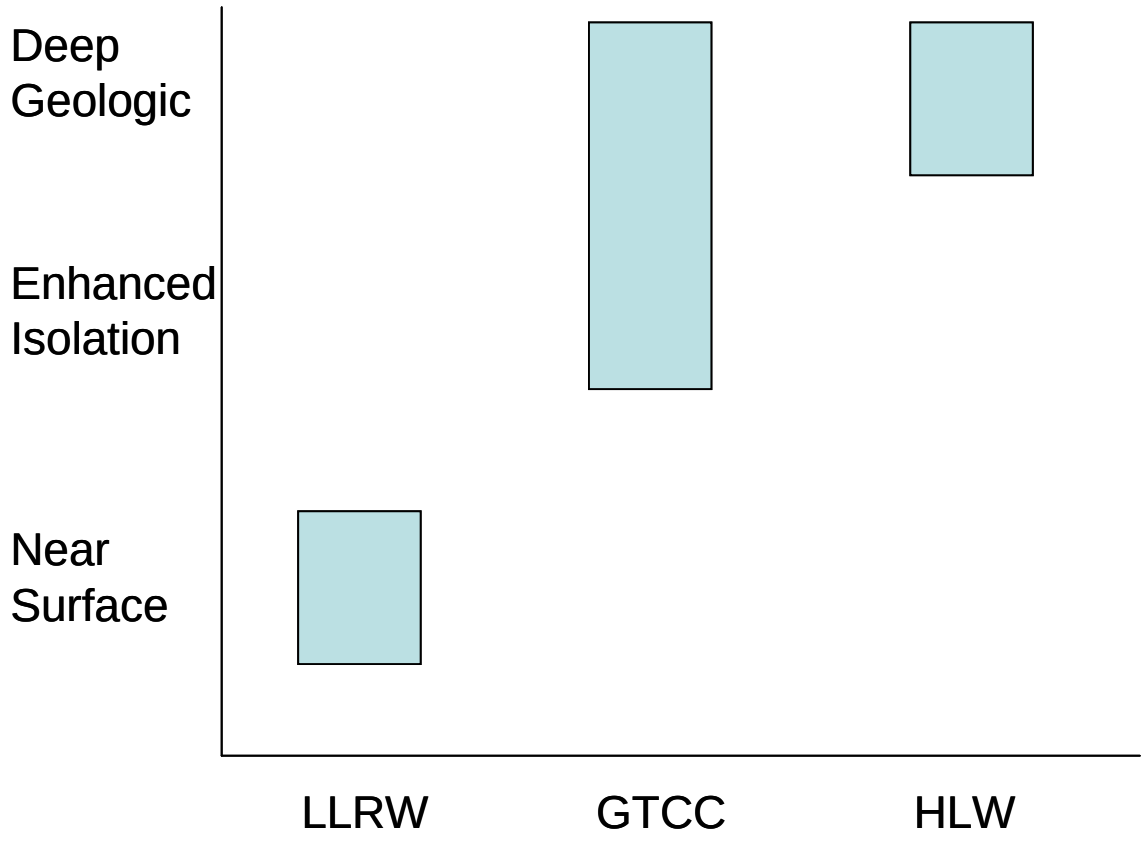


Figure 1. Disposal System Classes

Near Surface: Near surface disposal systems are the range of disposal facilities located in or within the first 30 meters of the earth's surface, consistent with the definition at 10 CFR 61.2. This definition also includes above ground disposal facilities. Near surface facilities are assumed to be capable of disposing only LLRW.

Enhanced Isolation: Enhanced isolation disposal systems are the range of disposal facilities that offer a greater degree of isolation than is offered by near surface systems, but not the degree of

isolation offered by deep geologic facilities. Enhanced isolation disposal systems include both enhanced near surface disposal facilities and facilities located at greater depths that are being considered by the U.S. Department of Energy in it's Environmental Impact Statement for the Disposal of Greater-Than-Class-C Low-Level Radioactive Waste [Ref. 5]. Enhanced isolation disposal systems would be used to dispose of GTCC wastes that would not require a higher degree of isolation offered by deep geologic disposal.

Deep Geologic: Deep geologic disposal systems are the range of disposal facilities located hundreds of meters below the earth's surface. Deep geologic disposal is the internationally accepted method for disposing HLW (and SNF). Deep geologic disposal may also be needed to dispose of GTCC wastes that require a higher degree of isolation. Deep geologic disposal of GTCC is being considered by the U.S. Department of Energy in it's Environmental Impact Statement for the Disposal of Greater-Than-Class-C Low-Level Radioactive Waste.

The potential disposal pathways for GNEP wastes are shown in Table 3 through 6 for the COEX, UREX, and ECHEM processes under the current policy/regulatory framework. Tables 3 and 4 show that under the current policy/regulatory framework the vast majority of the waste generated would be classified as HLW and require deep geologic disposal. Disposal of only cladding hulls, hardware, tritium, and krypton waste streams via disposal pathways other than deep geologic disposal would be possible.

Table 5 shows the disposal pathways for the COEX process under the assumed revised policy/regulatory framework. Again, the vast majority of the waste generated would be classified as HLW due to the high radionuclide content that would be in the waste (high radioactivity and significant heat generation for several centuries). The only benefit offered by the revised regulation is the possibility of being able to classify the tritium and krypton waste streams as exempt after a period of decay storage.

The potential disposal pathways for wastes generated by the UREX and ECHEM processes under the assumed policy/regulatory framework are shown in Table 6. Recovering and recycling minor actinides and the separation of Cs/Sr allows considerable flexibility in managing wastes. While several of the waste streams would likely still be classified as HLW, requiring deep geologic disposal, other waste forms (e.g., lanthanides and fission products) would not likely have to be disposed as HLW. Decay storage would also allow flexibility in managing the heat generating Cs/Sr waste form and could also potentially allow the tritium and krypton waste streams as exempt.

The most important factor in determining the disposal pathway for the wastes generated by the UREX and ECHEM processes under the assumed policy/regulatory framework would be the risk to the public associated with disposing wastes in a specific facility as opposed to one that offers a greater degree of isolation (i.e., and enhanced isolation facility vs. a deep geologic facility). Table 6 discusses those waste streams where this would play a role. It may be possible that an increased degree of isolation may be required for a given classification of waste. For example, deep geologic disposal may be the preferred disposal pathway for some wastes that could be classified as GTCC.

Additional analyses will be required, should policies and regulations regarding the definition of HLW be changed, to determine the preferred disposal pathways for wastes generated by the UREX and ECHEM processes. Additional information collected from the research and development program would support these analyses.

Table 3. Possible Disposition Pathways for Wastes Generated From COEX Processing, Current Policy/Regulatory Framework

Waste Stream	Potential Waste Form(s)	Classification	Potential Disposal Options			Discussion
			Deep Geologic	Enhanced Isolation	Near-Surface	
Transuranic and Fission Products	Borosilicate Glass	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Metals—Cladding/ Hardware	Compacted metal. Metal ingot if cost effective.	Classification would depend on segregation and clean-up. Likely GTCC, but fraction could be HLW and a fraction could be LLRW.	X	X	X	Radionuclide concentrations would dictate disposal system option. Significant clean-up and/or segregation of the waste stream could result in classification as LLRW and near-surface disposal being feasible for perhaps some of this waste stream. High concentrations of key radionuclides through little/no clean-up could require classification as HLW, requiring deep geologic disposal. GTCC classification may be possibly with clean-up with enhanced isolation disposal being feasible.
Tritium	Grouted tritiated water (HTO).	Likely LLRW, exempt with sufficient decay storage			X	Decay storage (^{3}H half-life of 12.3 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classification and near-surface disposal would be required.
Iodine	Grouted silver zeolite.	Likely HLW, potentially GTCC	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Carbon-14	Grouted $Na/CaCO_3$.	Likely GTCC, possibly HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Kr	Pressurized gas cylinder w/wo Xe.	Likely LLRW, exempt with sufficient decay storage			X	Decay storage (^{85}Kr half-life of 10.7 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classification (Class A) and near-surface disposal would be required (with solidification likely required).

Table 4. Possible Disposition Pathways for Wastes Generated From UREX and Echem Processing, Current Policy/Regulatory Framework

Waste Stream	Potential Waste Form(s)		Classification	Potential Disposal Options			Discussion
	UREX	Echem		Deep Geologic	Enhanced Isolation	Near-Surface	
Tc	Metal Alloy, possibly containing UDS and transition metal FP. Alloy may require Zr/Fe, which could come from cladding and hardware.	Metal Alloy containing UDS and transition metal FP. Alloy may contain cladding, and may require supplemental Zr or Fe, which could come from additional cladding and hardware.	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Cs/Sr	Glass or Ceramic, process design should consider ramifications of high heat, high radioactivity, powder handling should be avoided.	Glass-bonded sodalite, regardless of whether Cs/Sr and balance of FP are removed from salt.	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Ln	Glass— borosilicate glass if segregated as separate Ln stream. Ln/FP borosilicate glass if Ln and FP streams are combined.	Glass-bonded sodalite, regardless of whether Cs/Sr and balance of FP are removed from salt.	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
FP	Metal alloy potentially combined with Tc and UDS. Borosilicate glass if combined with lanthanides.	Glass bonded sodalite, regardless of whether or not Cs/Sr and balance of FP are removed from salt.	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.

Table 4. Possible Disposition Pathways for Wastes Generated From UREX and Echem Processing, Current Policy/Regulatory Framework (continued)

Waste Stream	Potential Waste Form(s)		Classification	Potential Disposal Options			Discussion
	UREX	Echem		Deep Geologic	Enhanced Isolation	Near-Surface	
Metals—Cladding/ Hardware	Compacted metal. Metal ingot if cost effective.	Compacted metal. Metal ingot if cost effective.	Classification would depend on segregation and clean-up. Likely GTCC, but fraction could be HLW and a fraction could be LLRW.	X	X	X	Radionuclide concentrations would dictate disposal system option. Significant clean-up and/or segregation of the waste stream could result in classification as LLRW and near-surface disposal being feasible for perhaps some of this waste stream. High concentrations of key radionuclides through little/no clean-up could require classification as HLW, requiring deep geologic disposal. GTCC classification may be possible with clean-up with enhanced isolation disposal being feasible.
Tritium	Grouted tritiated water (HTO).	Grouted tritiated water (HTO).	Likely LLRW, exempt with sufficient decay storage			X	Decay storage (^{3}H half-life of 12.3 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classification and near-surface disposal would be required.
Iodine	Grouted silver zeolite.	Glass-bonded sodalite w/Ln/FP.	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Carbon-14	Grouted $Na/CaCO_3$.	Glass-bonded sodalite w/Ln/FP.	HLW	X			Definition of HLW in the NWPA: the highly radioactive material resulting from the reprocessing of spent nuclear fuel.
Kr	Pressurized gas cylinder w/wo Xe.	Pressurized gas cylinder w/wo Xe.	Likely LLRW, exempt with sufficient decay storage			X	Decay storage (^{85}Kr half-life of 10.7 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classification (Class A) and near-surface disposal would be required (with solidification likely required).

Table 5. Possible Disposition Pathways for Wastes Generated From COEX Processing, Assumed Revised Policy/Regulatory Framework

Waste Stream	Potential Waste Form(s)	Classification	Potential Disposal Options				Discussion
			Deep Geologic	Enhanced Isolation	Near-Surface	Exempt	
Transuranic and Fission Products	Borosilicate Glass	HLW	X				Contains all fission products and minor actinides, except for Pu. Highly radioactive for a very long period of time and generates significant heat for several centuries.
Metals—Cladding/ Hardware	Compacted metal. Metal ingot if cost effective.	Classification would depend on segregation and clean-up. Likely GTCC, but fraction could be HLW and a fraction could be LLRW.	X	X	X		Radionuclide concentrations would dictate disposal system option. Significant clean-up and/or segregation of the waste stream could result in classification as LLRW and near-surface disposal being feasible for perhaps some of this waste stream. High concentrations of key radionuclides through little/no clean-up could require classification as HLW, requiring deep geologic disposal. GTCC classification is likely with clean-up with enhanced isolation disposal being feasible.
Tritium	Grouted tritiated water (HTO).	Likely LLRW, exempt with sufficient decay storage			X	X	Decay storage (^{3}H half-life of 12.3 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classifation and near-surface disposal would be required.
Iodine	Grouted silver zeolite.	Likely HLW, potentially GTCC	X	X			^{129}I is a long-lived fission product (half-life of 16,000,000 years) and is a dominant radionuclide in repository performance assessments in various geologic environments (e.g., unsaturated tuff, granite, clay, salt). Waste form concentrations would be well in excess of Class C LLRW limits, perhaps high enough to result in HLW classification. Significant quantities of ^{129}I could result in an enhanced isolation disposal facility being unable to meet individual protection requirements and if so, disposal in such facilities would not be feasible.
Carbon-14	Grouted Na/$CaCO_3$.	Likely GTCC, possibly HLW	X	X			^{14}C is a moderately long-lived (half-life of 5,730 yrs) activitation product and has been shown to be an important radionuclide in oxidizing environments (e.g., tuff). Waste form would likely classify as GTCC. High concentrations may warrant deep geologic disposal as either GTCC or HLW.
Kr	Pressurized gas cylinder w/wo Xe.	Likely LLRW, exempt with sufficient decay storage			X	X	Decay storage (^{85}Kr half-life of 10.7 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classification (Class A) and near-surface disposal would be required (with solidification likely required).

Table 6. Possible Disposition Pathways for Wastes Generated From UREX and Echem Processing, Assumed Revised Policy/Regulatory Framework

Waste Stream	Potential Waste Form(s)		Classification	Potential Disposal Options				Discussion
	UREX	Echem		Deep Geologic	Enhanced Isolation	Near-Surface	Exempt	
Tc	Metal Alloy, possibly containing UDS and transition metal FP. Alloy may require Zr/Fe, which could come from cladding and hardware.	Metal Alloy containing UDS and transition metal FP. Alloy may contain cladding, and may require supplemental Zr or Fe, which could come from additional cladding and hardware.	Likely HLW, potentially GTCC	X	X			^{99}Tc is a long-lived fission product (half-life of 213,000 years) and has been shown to be an important radionuclide in repository performance assessments in various geologic environments (e.g., unsaturated tuff, clay). Waste form concentrations would be well in excess of Class C LLRW limits, likely high enough to result in HLW classification. Significant quantities of ^{99}Tc could result in an enhanced isolation disposal facility being unable to meet individual protection requirements and if so, disposal in such facilities would not be feasible.
Cs/Sr	Glass or Ceramic, process design should consider ramifications of high heat, high radioactivity, powder handling should be avoided.	Glass-bonded sodalite, regardless of whether Cs/Sr and balance of FP are removed from salt.	HLW, GTCC, or LLW depending on period of decay storage	X	X	X		The period of decay storage would dictate classification and possible disposal paths. ^{135}Cs has been shown to be an important radionuclide in repository performance assessments in various geologic media (unsaturated tuff, granite, salt). Limited or no decay storage would likely render this waste form as HLW due to large heat generation rates requiring geologic disposal. Increased decay storage would reduce both heat ouput such that the waste form could be classified as GTCC when considering heat only. Radioactivity (individual protection requirement) would discriminate between deep geologic and enhanced isolation disposal (as either HLW or GTCC). Very long decay storage would further reduce activity such that it could potentially be classified as LLRW based on ^{137}Cs concentration. However the concentration and total inventory of ^{135}Cs to be disposed could either limit the amount that could be disposed in a near-surface facility (individual protection requirement) or require additional isolation, perhapos being disposed in an enhanced isolation facility.

Table 6. Possible Disposition Pathways for Wastes Generated From UREX and Echem Processing, Assumed Revised Policy/Regulatory Framework (continued)

Waste Stream	Potential Waste Form(s)		Classification	Potential Disposal Options				Discussion
	UREX	Echem		Deep Geologic	Enhanced Isolation	Near-Surface	Exempt	
Ln	Glass—borosilicate glass if segregated as separate Ln stream. Ln/FP borosilicate glass if Ln and FP streams are combined.	Glass-bonded sodalite, regardless of whether Cs/Sr and balance of FP are removed from salt.	Likely GTCC, quantity of residual transuranic and any combination with I or ^{14}C waste streams could lead to HLW classification	X	X			This waste form would contain no significant heat generating radionuclides, but would contain residual alpha-emitting transuranic radionuclides. The waste could likely be classified as GTCC. This waste form would not contain radionuclides that contribute significantly to risk (individual protection requirements) associated with deep geologic disposal and enhanced isolation disposal may be feasible. Combination of this waste stream with Iodine and/or ^{14}C waste stream (Echem) could affect classification and possible disposal pathway.
FP	Metal alloy potentially combined with Tc and UDS. Borosilicate glass if combined with lanthanides.	Glass bonded sodalite, regardless of whether or not Cs/Sr and balance of FP are removed from salt.	Likely GTCC, quantity of residual transuranic and any combination with I or ^{14}C waste streams could lead to HLW classification	X	X			This waste form would contain no significant heat generating radionuclides, but would contain residual alpha-emitting transuranic radionuclides. The waste could likely be classified as GTCC. This waste form would not contain radionuclides that contribute significantly to risk (individual protection requirements) associated with deep geologic disposal and enhanced isolation disposal may be feasible. Combination of this waste stream with Iodine and/or ^{14}C waste stream (Echem) could affect classification and possible disposal pathway.
UDS	Metal alloy potentially combined with Tc and FP.	Metal alloy containing Tc and transition metal FP. Matrix may contain cladding, and supplemental Zr/Fe could come from additional cladding and hardware.	Likely HLW, potentially GTCC	X	X			UDS will contain ^{99}Tc which is a long-lived fission product (half-life of 213,000 years) and has been shown to be an important radionuclide in repository performance assessments in various geologic environments (e.g., unsaturated tuff, clay). Waste form concentrations would be well in excess of Class C LLRW limits, likely high enough to result in HLW classification. Significant quantities of ^{99}Tc could result in an enhanced isolation disposal facility being unable to meet individual protection requirements and if so, disposal in such facilities would not be feasible.

Table 6. Possible Disposition Pathways for Wastes Generated From UREX and Echem Processing, Assumed Revised Policy/Regulatory Framework (continued)

Waste Stream	Potential Waste Form(s)		Classification	Potential Disposal Options				Discussion
	UREX	Echem		Deep Geologic	Enhanced Isolation	Near-Surface	Exempt	
Metals—Cladding/ Hardware	Compacted metal. Metal ingot if cost effective.	Compacted metal. Metal ingot if cost effective.	Classification would depend on segregation and clean-up. Likely GTCC, but fraction could be HLW and a fraction could be LLRW.	X	X	X		Radionuclide concentrations would dictate disposal system option. Significant clean-up and/or segregation of the waste stream could result in classification as LLRW and near-surface disposal being feasible for perhaps some of this waste stream. High concentrations of key radionuclides through little/no clean-up could require classification as HLW, requiring deep geologic disposal. GTCC classification may be possible with clean-up with enhanced isolation disposal being feasible.
Tritium	Grouted tritiated water (HTO).	Grouted tritiated water (HTO).	Likely LLRW, exempt with sufficient decay storage			X	X	Decay storage (^{3}H half-life of 12.3 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classifation and near-surface disposal would be required.
Iodine	Grouted silver zeolite.	Glass-bonded sodalite w/Ln/FP.	Likely HLW, potentially GTCC	X	X			^{129}I is a long-lived fission product (half-life of 16,000,000 years) and is a dominant radionuclide in repository performance assessments in various geologic environments (e.g., unsaturated tuff, granite, clay, salt). Waste form concentrations would be well in excess of Class C LLRW limits, perhaps high enough to result in HLW classification. Significant quantities of ^{129}I could result in an enhanced isolation disposal facility being unable to meet individual protection requirements and if so, disposal in such facilities would not be feasible.
Carbon-14	Grouted $Na/CaCO_3$.	Glass-bonded sodalite w/Ln/FP.	Likely GTCC, possibly HLW	X	X			^{14}C is a moderately long-lived (half-life of 5,730 yrs) activitation product and has been shown to be an important radionuclide in oxidizing environments (e.g., tuff). Waste form would likely classify as GTCC. High concentrations may warrant deep geologic disposal as either GTCC or HLW.

Table 6. Possible Disposition Pathways for Wastes Generated From UREX and Echem Processing, Assumed Revised Policy/Regulatory Framework (continued)

Waste Stream	Potential Waste Form(s)		Classification	Potential Disposal Options				Discussion
	UREX	Echem		Deep Geologic	Enhanced Isolation	Near-Surface	Exempt	
Kr	Pressurized gas cylinder w/wo Xe.	Pressurized gas cylinder w/wo Xe.	Likely LLRW, exempt with sufficient decay storage			X	X	Decay storage (^{85}Kr half-life of 10.7 yrs) could result in this waste being exempt. Otherwise, concentrations would result in LLRW classification (Class A) and near-surface disposal would be required (with solidification likely required).

3 Disposal Systems

This section summarizes disposal facilities within each of the systems (deep geologic, enhanced isolation, near surface). The purpose of this section is to identify the types of disposal facilities that potentially could be used for disposing wastes generated under the GNEP. This summary is provided at a survey-level and additional information can be found in the references cited.

3.1 Deep Geologic Disposal Facilities

Geologic disposal of HLW and SNF is internationally accepted. In 2001 the National Research Council reaffirmed their position regarding geologic disposal of HLW, stating [Ref. 6]:

> "Geological disposal, the approach recommended in previous National Research Council (NRC) reports and by many other national and international scientific bodies, is the only available alternative that does not require ongoing control and resource expenditures by future generations. The science supporting this alternative has been developed by intensive work over the past 25 years. The view repeatedly expressed by a large fraction of the scientific and technical community is that geological disposal, correctly managed, is a safe approach to long-term management of HLW and that it best satisfies the ethical goal of minimizing burdens on future generations. Nevertheless, uncertainties remain, and some scientists feel that it is premature to commit fully to disposal. The biggest challenges to initiating geological disposition, however, are societal: there is a clear lack of public confidence and support in many countries for proceeding with siting and construction of geological repositories."

The Nuclear Waste Policy Act defines the term disposal [Ref. 7] as the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste. However, a key aspect of deep geologic disposal becomingin more important internationally is the concept of retrievablity with the context of stepwise decision making.

In 2004 the OECD Nuclear Energy Agency (NEA) published *Stepwise Approach to Decision Making for Long-term Radioactive Waste Management - Experience, Issues and Guiding Principles* [Ref. 8]. This report discusses and recommends the use of a stepwise decision making process in the management of radioactive waste. Although the report focuses primarily on ultimate disposition in geologic repositories, its findings and recommendations are relevant to the disposition of wastes generated by the GNEP. Key points, taken from this report, are presented below.

- Radioactive waste management involves both technical and societal decision making.
- The key feature of the stepwise concept is development by steps or stages that are reversible, within the limits of practicability.
- A stepwise approach provides reassurance that decisions can be reversed if experience shows them to have adverse or unwanted effects.
- A stepwise approach to decision making has thus come to the fore as being of value in advancing long-term radioactive waste management solutions in a societally acceptable manner.

Reversibility denotes the fact that fallback positions are incorporated in the long-term waste management policy, as well as in the actual technical program. Reversibility is meant to help a facility program respond flexibly to:

- new technical information regarding the site and design;
- new technological developments relevant to radioactive waste management;
- changes in economic, social and political conditions and acceptance; and
- changes in regulatory guidance and its interpretation or even, possibly, in basic safety standards.

Reversibility is assured by considering and incorporating fallback positions at any given step in the development program of a waste management facility. This contributes both to technical confidence in the ability to manage the waste safely and, also, to confidence in wider audiences that an irreversible decision is not being made. Reversibility should not be seen as a lack of confidence in ultimate safety of a waste management option, but rather as a desire to make optimum use of available options and design alternatives.

Some facility concepts for deep geologic disposal and certain geologic media are more amendable for implementing a stepwise decision making process that includes retrievability than others. The retrievability of wastes is virtually impossible in some concepts. Retrievability aspects are discussed below for each concept within the deep geologic disposal system.

3.1.1 Mined Geologic Repository

Every organization actively pursuing the disposal of SNF or HLW is investigating the disposal of these wastes in mined geologic repositories. A mined geologic repository is simply that, a mined facility for the disposal of wastes located hundreds of meters beneath the earth's surface. They consist of both engineered and natural barriers that together serve to prevent or minimize the movement of radionuclides to a point where they can affect the population. It is recognized that a properly sited and constructed repository with passive engineered and natural barriers will provide adequate protection of public health and safety during the hazardous lifetime of the wastes without requiring additional human action.

Several geologic media have been considered including salt, unsaturated tuff, and saturated basalt, shale, granite, argillite, and clay. Repository designs differ based on the quantities and types of waste disposed (SNF vs. HLW) and the geologic media in which the repository would be constructed. In general, they consist of access shafts or ramps and rooms, tunnels, or galleries for disposing of wastes. Design concepts for several repositories under development are shown in Figure 2.

The development of a repository can be broken out into 4 phases.

- *Site Characterization, Preliminary Design, Licensing*: The activities required to characterize the site, develop a preliminary design, and to develop the safety case to submit to the regulator to obtain authorization to construct the facility and to receive waste
- *Detailed Design, Surface Facility Construction, Initial Subsurface Facility Construction*: Detailed facility design followed by construction of the surface waste handling facilities and the initial subsurface facilities for waste emplacement
- *Subsurface Facility Construction and Emplacement*: Construction of additional sub surface disposal facilities in parallel with emplacement operations
- *Monitoring and Ventilation*: Ventilation of the subsurface facility (active and natural) to allow for thermal decay and monitoring
- *Closure*: Sealing, backfilling (depending on the repository design), and repository closure

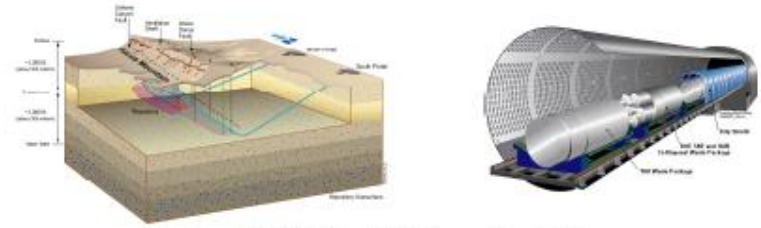
These phases are for the most part independent, but there is some overlap. For example, in a two-step licensing approach, as is the case for the proposed Yucca Mountain repository, licensing would continue into the detailed design, surface facility construction, and initial subsurface facility construction phase. Some steps also may not be included. For example, some designs may not utilize backfill or the facility

may be backfilled immediately after the waste is emplaced. In addition, monitoring and ventilation may not be included in some repository concepts as immediate closure may be desired.

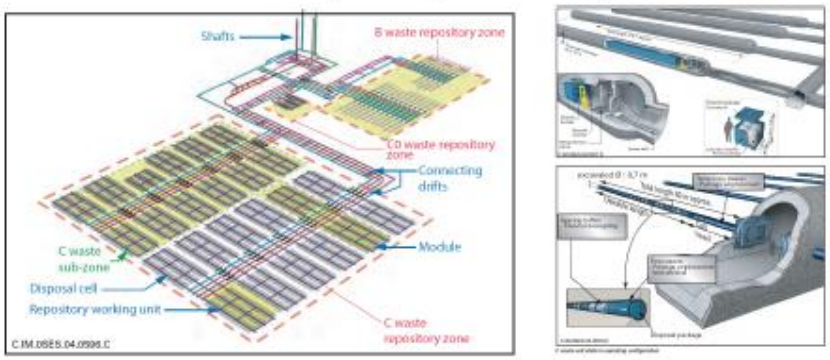
The ability to retrieve wastes depends on the geologic media and the operational phase. Although repositories are designed primarily to dispose wastes, typically the disposed wastes can be are retrieved until backfill is placed, the repository is sealed, or both. This is not to say that repositories are designed such that the waste can be retrieved, but rather retrievability is possible. Retrieving wastes disposed in salt may be more difficult due to salts propensity to creep, which would be accelerated in the presence of heat generating wastes.

3.1.2 Deep Borehole Disposal

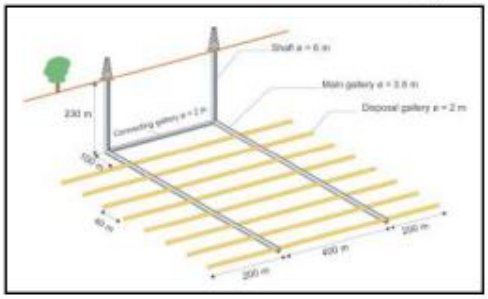
The disposal of radioactive wastes in deep boreholes is not a new concept, but only began to receive consideration during the 1990s. In the deep borehole concept waste would be emplaced in the lower part of one or more deep boreholes drilled in tectonically, hydrologically, thermally and geochemically stable rock formations. Once the emplacement zone of the borehole is filled with materials, the "isolation zone" extending from the top of the emplacement zone to the ground surface is filled and sealed with appropriate materials. A diagram of a deep bore hole disposal facility is shown in Figure 3.



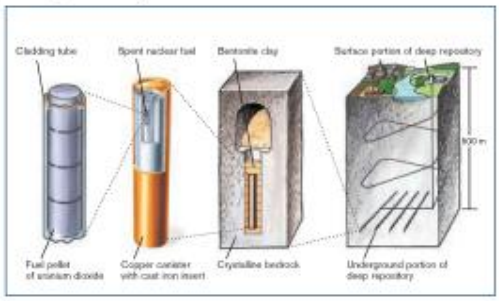
Unsaturated Tuff (U.S. DOE, Yucca Mountain)



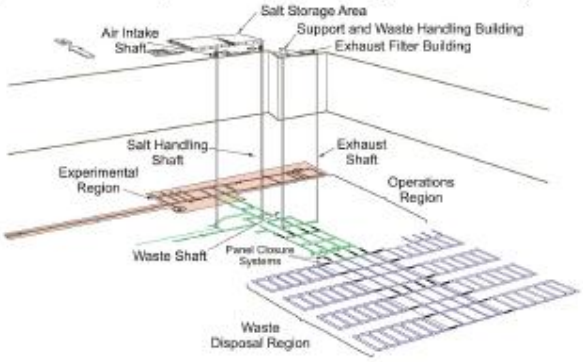
Argillite (France, ANDRA)



Clay (Belgium, ONDRAF/NIRAS)



Crystalline Rock (Granite, SKB)



Bedded Salt (U.S. DOE, WIPP)

Figure 2: Mined Geologic Repository Concepts [Refs. 9-13]

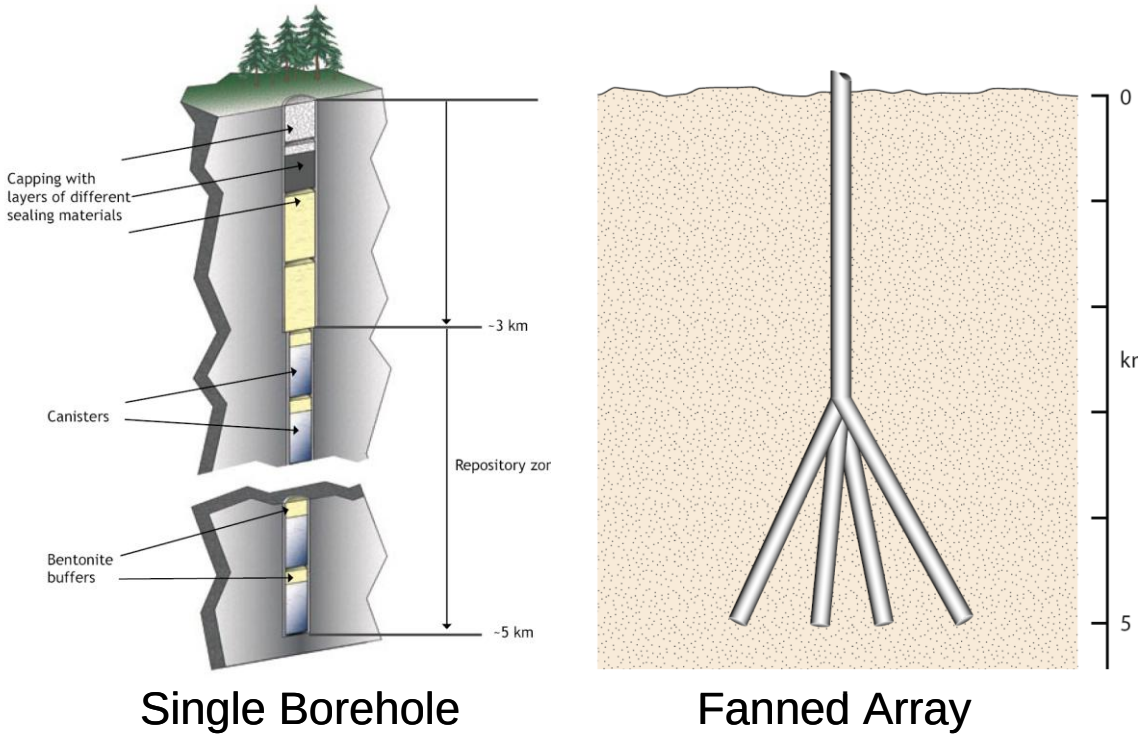


Figure 3. Deep Borehole Concept [Ref. 14]

At emplacement depths, the groundwater is expected to be relatively stagnant, highly saline, hot (75-150 °C), and under high pressure. In deep boreholes there is a large barrier to transport posed by the isolation zone because of its low permeability and high sorptivity, the stability and low-solubility of the disposal form, and high salinity and the lack of driving forces for fluid flow. Thus the disposed material is expected to remain, for all practical purposes, permanently isolated from the biosphere.

Two concepts have been proposed for a deep borehole disposal facility [Refs. 14 and 15]. The first is a low temperature concept where wastes are disposed at low concentrations such that the heat released into the rock is limited. The bedrock surrounding the boreholes will be impacted as little as possible so as to maintain a stable ground water density stratification. A second concept is a high temperature facility where the waste is disposed at larger concentrations with boreholes placed close together. In this concept the heat generated would partially melt immediately adjacent rock, purging the area surrounding the boreholes of water and gas. As the waste cools, the rock would solidify into dry, newly crystallized rock.

Rather than relying on a combination of engineered and natural barriers to protect the public as does a mined geologic repository, deep boreholes rely on the natural conditions of the site as the only isolation barrier. However, in a high temperature concept the waste package must remain intact during the high temperature stage. Lifetimes on the order of 10,000 years are required. Copper or noble metal alloys may be the only metallic materials that could survive the high temperatures and pressures for the requisite time. Mineral-based waste packages may be needed.

As discussed in *Disposition of Excess Weapon Plutonium in Deep Boreholes, Site Selection Handbook* [Ref. 16], ideally, a deep borehole site within the United States would consist of a combination of:

- cyrstaline rock at the surface or within 1 km of the surface,

- a region that is tectonically stable,
- an area located away from population centers, and
- a region that is not near international borders (~200 km).

Cratons are part of the earth's continental crust that are stable and have been little deformed for a prolonged period and a pluton is a large mass of igneous material that has intruded into the shallow earth crust. The center of a pluton within a craton that is either exposed or overlaid by a relatively thin layer of sedimentary material would likely be preferred sites for a deep borehole disposal facility.

As also discussed in *Disposition of Excess Weapon Plutonium in Deep Boreholes, Site Selection Handbook* [Ref. 16], about 90 percent of the United States is underlain by Precambrian rock (>540 million years old) which make up the continental crust with large areas of this "basement" covered by less than 1 kilometer of sedimentary and volcanic rock. The Midcontinent region (between the Rock and Appalachian mountains) has an approximate area of at least 2,600,000 km^2 of accessible Precambrian basement within 1 km of the surface and could be a large resource for siting a deep borehole disposal facility. Within this region, the Canadian Shield is a large tectonically stable area of "basement" rocks that has been exposed by glaciation of a craton (stable continental mass). The Canadian Shield extends into the northern United States where it is either exposed or covered by a thin layer of sedimentary cover. Plutonic rocks within the Canadian Shield area may be ideal sites because they are relatively uniform in nature.

Criteria for developing a deep borehole disposal facility have been proposed in *Final Deposition of High-level Nuclear Waste in Very Deep Boreholes: An Evaluation Based on Recent Research of Bedrock Conditions at Great Depths* [Ref. 14]. These are:

1. the existence of a sufficiently large area at a depth of 3 to 5 km having groundwater, the density-stratification of which is stable;
2. the availability of reliable technology for measurements and analyses that can localize areas at -3 to -5 km having groundwater, the density-stratification of which is stable;
3. sufficient knowledge of geodynamic and hydrogeological conditions as to permit the identification of areas at depths of 3 to 5 km;
4. the availability of technology for the precision drilling required for both exploration and deposition;
5. the ability to deposit filled canisters and, during the period of deposition, to retrieve canisters in order to exchange them or to test materials and technological solutions;
6. the feasibility of drilling boreholes, depositing the canisters, and sealing all of the boreholes without corrupting the long-term stability of the density-stratification of the groundwater around the repository;
7. the feasibility of storing high-level radioactive waste in canisters for extremely long periods of time so that neither the heat nor the radioactivity generated by the decay process corrupts the stability of the density-stratification of the groundwater around the repository; and
8. the selection of drilling equipment, canisters and sealing materials with a view to avoiding chemical reactions that might give rise to gases in the repository area.

In 1996, the DOE considered deep borehole disposal as a method for disposing surplus weapons grade plutonium [Refs. 17 and 18]. Although this method was not chosen for managing surplus weapons grade plutonium, the concept was found to be feasible. As stated in *Technical Summary Report for Weapons-Useable Plutonium Disposition* [Ref. 17]:

> "While no deep borehole disposal facilities for plutonium disposition have ever been developed, many of the technologies needed for this alternative are quite mature; and the basic concept has been considered previously for waste disposal."
>
> "It is believed that suitable rock formations can be found in a variety of areas, that they can be adequately characterized, and that the long term evolution of processes can be predicted to assure long term isolation and safety."
>
> "Siting guidelines and procedures is the largest area of uncertainty. Site suitability guidelines consistent with the mission and safety concept of deep borehole disposition will require development. … a regulatory framework to address this deep borehole disposal does not currently exist. Therefore, regulatory uncertainty was identified as a risk that affects the viability of deep borehole disposition. However, preliminary discussions with licensing experts indicate that a licensing regime can be developed, given sufficient time and a mandate."
>
> "The equipment required to implement the deep borehole alternatives are adaptations of equipment designed and used for nuclear weapons testing, geological studies, and the petroleum and gas drilling industries. The equipment requirements with respect to environmental safety and quality are within current capability or are viable extrapolations from existing mechanical engineering designs. An integration and demonstration of the equipment will be required, and the systems engineering must be performed. Notwithstanding, the mechanical design is not expected to be a controlling technical risk for these alternatives."

In 2003, the Massachusetts Institute of Technology released a report entitled *The Future of Nuclear Power, An Interdisciplinary MIT Study* that discussed deep borehole disposal an alternative to mined geologic repositories [Ref. 19]. The report identified several obstacles as listed below.

- A new set of standards and regulations would have to be developed;
- The difficulty of retrieving waste from boreholes would be difficult should a problem develop;
- Satisfying current U.S. regulations that require a period of several decades during which the waste must be retrievable would be difficult and expensive, but not impossible, in a deep borehole facility;
- The knowledge of in situ conditions at great depth would never be as comprehensive as in a shallower mined geologic repository environment;
- Recovery from accidents during waste emplacement would likely be more difficult than in a mined geologic repository; and
- It is difficult to predict the impact on public opinion of a shift in siting strategy from a large central repository to perhaps several widely dispersed boreholes.

Despite these obstacles, the report authors state:

> "Despite these obstacles, we view the deep borehole disposal approach as a promising extension of geological disposal, with greater siting flexibility and the potential to reduce the already very low risk of long-term radiation exposure to still lower levels without incurring significant additional costs."

3.1.3 Other Deep Geologic Disposal Concepts

Other concepts have been proposed for deep geologic disposal systems. However, some have not been fully evaluated and are very conceptual. They are presented here both for completeness and to introduce the concepts for potential future evaluation.

International Repositories - High Isolation Sites

Preliminary efforts in siting an international mined geologic repository have focused on finding what is termed a high isolation site. In order to choose a site for an international repository, an organization called Pangea proposed finding a site that would fulfill the safety requirements of national repository programs, but would also be as simple as possible such that the safety case could be demonstrated with the most transparency. This resulted in Pangea identifying a set of attributes where a high isolation site would have most, but not necessarily all, of them [Ref. 20]. The attributes are:

- Stable geology (needed because of the extremely long isolation times required)
- Flat topography (reduces driving forces for groundwater movement)
- Near-horizontal sedimentary strata (simpler to investigate and characterize)
- Stable, arid climate with negligible erosion (eases problem of extrapolation into the future)
- Low permeability host rock (reduces groundwater movements)
- Old and saline groundwater (indicates negligible groundwater movement; and non-potable water)
- Stratified salinity (counteracts thermal buoyancy effects)
- Reducing geochemical conditions (reduces solubilities of radionuclides)
- Absence of complex karst systems (simplifies hydrogeologic modeling)
- Low population density (reduces intrusion risks)
- No significant resource conflicts (reduces intrusion risks)

Pangea identified regions where potential high isolation sites might exist in Western and Southern Australia, Argentina, Southern Africa, and China [Ref. 20]. Pangea decided to focus its limited resources on Western Australia. However, the premature release of internal documents resulted in political and public resistance. At present, no feasibility results are available and Pangea has essentially ceased operations. Efforts to advance an international repository continue to be advanced by the Association for Regional and International Underground Storage (ARIUS) [Ref. 21], although they do not specifically promote a high-isolation site concept.

Deep Rock Melting

The deep rock-melting concept involves using the decay heat from the waste to first melt the adjacent rock, and perhaps the waste from itself, that when cooled will produce a solid mass that either incorporates or encases the waste. The waste would be disposed in either a shaft or excavated cavity at depth of 2-5 kilometers. The high temperature deep borehole concept discussed above is an example of a deep rock melting concept.

The technique would only be applicable to wastes that would generate significant amounts of heat. Several of the waste streams generated through the processes being considered under the GNEP would not be amendable to this option because they do not generate significant quantities of heat. The vitrified HLW from COEX processing or the Cs/Sr waste form with little decay could potentially be disposed using this approach.

As discussed in a NIREX report entitled *Description of Long-term Management Options for Radioactive Waste Investigated Internationally* [Ref. 22[a]], in the late 1970's and early 1980's, a deep rock melting concept was taken to the engineering design stage. The design concept involved a shaft or borehole which led to an excavated cavity at a depth of 2-5 km. The designers estimated that the waste would be immobilized in a volume of rock one thousand times larger than the original volume of the waste.

A variation of the deep rock melting concept is the salt-diver repository where the high-heat generation rates of the waste is postulated to allow disposal at depths up to 10 km underground in salt domes [Ref. 23]. The wastes are packaged into moderately large containers called salt divers that are placed in a salt dome. The high-density salt-diver heat source sinks by heating the salt under the WP until the salt becomes plastic and the salt diver then sinks to the bottom of the salt dome.

There are several obstacles associated with deep rock melting disposal concepts (and variants).

- Establishment of standards and regulations
- Knowledge of in situ conditions at depth
- Understanding of high temperature processes at depth (i.e., rock melting, cooling and re-crystallization; heating and movement of waste packages through the plastic salt in the salt diver concept).
- Ability to predict characteristics following cooling of the waste in order to conduct long-term safety assessments of the disposal facility for demonstrating compliance with regulations.
- Wastes disposed in these concepts would not be retrievable.

Direct Injection

The NIREX report entitled *Description of Long-term Management Options for Radioactive Waste Investigated Internationally* [Ref. 22[b]] discusses the direct injection approach where liquid radioactive waste is directly injected into a layer of rock deep underground that has been chosen because of its suitable characteristics to trap the waste. The NIREX report identified a number of geological pre-requisites that are required.

- There must be a layer of rock, the injection layer, with sufficient porosity to accommodate the waste and with sufficient permeability to allow easy injection (i.e. act like a sponge).
- Above and below the injection layer there must be impermeable layers above and below the injection layer that act as a natural seal.
- Additional benefits could be provided from geological features that limit horizontal or vertical migration such as injection into layers of rock containing natural brine groundwater that is stratified.

Direct injection could, in principle, be used on any type of radioactive waste that can be transformed into a solution or slurry. Slurries containing a cement grout that would set as a solid when underground could also be used to help minimize movement of radioactive waste. This would require further processing of solidified wastes at a deep injection facility because it is not likely that it would be permissible to transport GNEP wastes in liquid or slurry form. However deep injection may be possible if a processing facility were located at a site where deep injection would be feasible.

[a] Section 5.5of referenced report.

[b] Section 5.6 of referenced report.

The NIREX report points out that Russia injected some tens of million cubic meters (by 2002) of low, intermediate, and high level of radioactive waste into porous sandstones capped by clay at depths of 400 meters and into sandstones and limestones at depths of 1,400 meters [Ref. 22].

As discussed by the National Academy of Science in *Disposition of High-Level Waste and Spent Nuclear Fuel: the Continuing Societal and Technical Challenges* [Ref. 24[c]]:

> "The United States practiced direct injection of low-level liquid waste grouts under high pressure into a shale formation beneath the Oak Ridge, Tennessee site in the early 1970s. This process was abandoned due to uncertainties about how the grout flowed within the fractured shale. In 1972, an NRC study found the option of disposing of HLW at the U.S. Savannah River Site directly into crystalline bedrock beneath the site to be technically feasible. However the report cautioned that public approval for this option would be problematic."
>
> "For many years, the former Soviet Union injected intermediate-level liquid waste into the subsurface at sites such as Krasnoyarsk, Tomsk, and Dimitrograd. In these cases, the waste appears to have been contained between geological strata as intended. However, the approach is being phased out because it is not considered to be in line with better practices that include solidifying and packaging the waste."

Although possibly feasible, it is unlikely that direct injection of GNEP wastes is a viable option because it involves waste forms that are not in solid form and no waste packaging. In addition, the approach is unlikely to be accepted by the public.

3.2 Enhanced Isolation Disposal Facilities

This section summarized concepts for enhanced isolation disposal facilities used primarily to dispose intermediate level waste. Most of the concepts presented are either being used, are under consideration, or have been considered in the past by various organizations internationally.

3.2.1 Geologic Repositories

Geologic repositories were discussed above, primarily focused on the disposal of spent nuclear fuel or high level nuclear waste. However, several organizations are either using or intend to use geologic repositories for the disposal of intermediate and in some cases low level wastes. Several of these facilities, either operational or conceptual, are discussed herein. While several facilities are essentially equivalent to deep geologic facilities in terms of depth, and will in fact dispose HLW, others are at much shallower depths. For completeness both deep and shallow geologic facilities are summarized.

Switzerland

NAGRA intends to dispose of both low- and intermediate level wastes in geologic repositories per Switzerland's Nuclear Energy Act and site selection is underway [Ref. 25] . In the past, a facility constructed horizontally into a hillside has been considered [Ref. 26]. Current activities indicate that the

[c] Chapter 7, Page 124 of referenced report

Opalinus clay formation may be suitable for locating a spent nuclear fuel, HLW, and intermediate-level waste repository [Ref. 27].

Finland

Finland disposes low- and intermediate-level wastes from the operation of the Olkiluoto and Loviisa nuclear power plants in geologic repositories constructed at a depth of 70 - 100 meters in crystalline bedrock at each plant site [Ref. 28]. A diagram of the VLJ repository at the Olkilouto site is shown in Figure 4.

Sweden

Since 1988 Swedish low and intermediate waste has been disposed in the SFR repository, located at the Forsmark nuclear power plant [Ref. 30]. The SFR repository is constructed in crystalline bedrock, 60 meters under the Baltic Sea and consists of five different chambers; four "simple" caverns for low-level waste and one concrete silo surrounded by a clay buffer for intermediate level waste [Ref. 31]. A diagram of the SFR repository is shown in Figure 5.

Canada

Ontario Power Generation intends to dispose intermediate level waste in a deep geologic facility in Ontario on the site of the Bruce nuclear power plant [Ref. 33]. The facility would be located 660 meters below the surface beneath thick layers of limestone and shale rock that have remained stable. It is estimated that 160,000 m^3 of low and intermediate level waste will be disposed in the facility. An artist's rendition of the facility is shown in Figure 6.

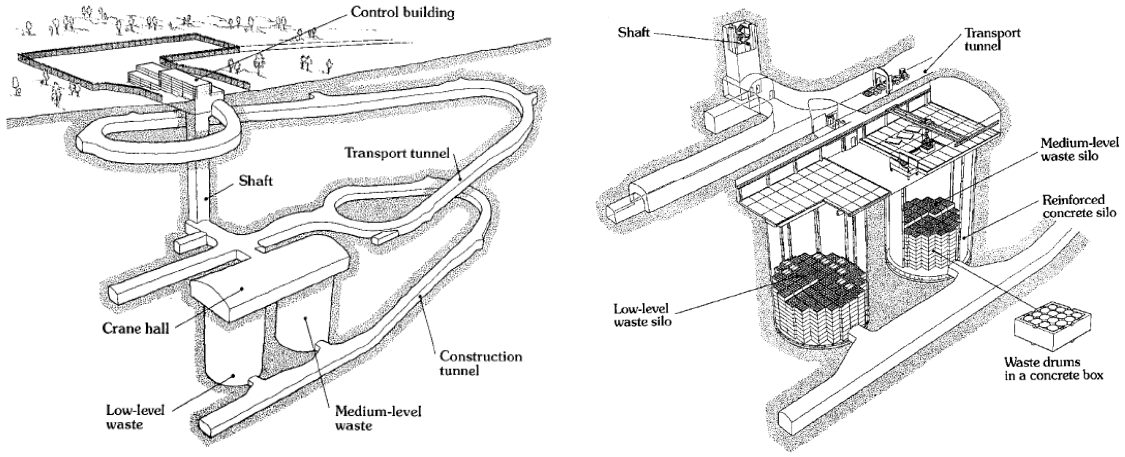


Figure 4. VLJ Repository at Finland's Olkilouto Nuclear Power Plant Site [Ref. 29]

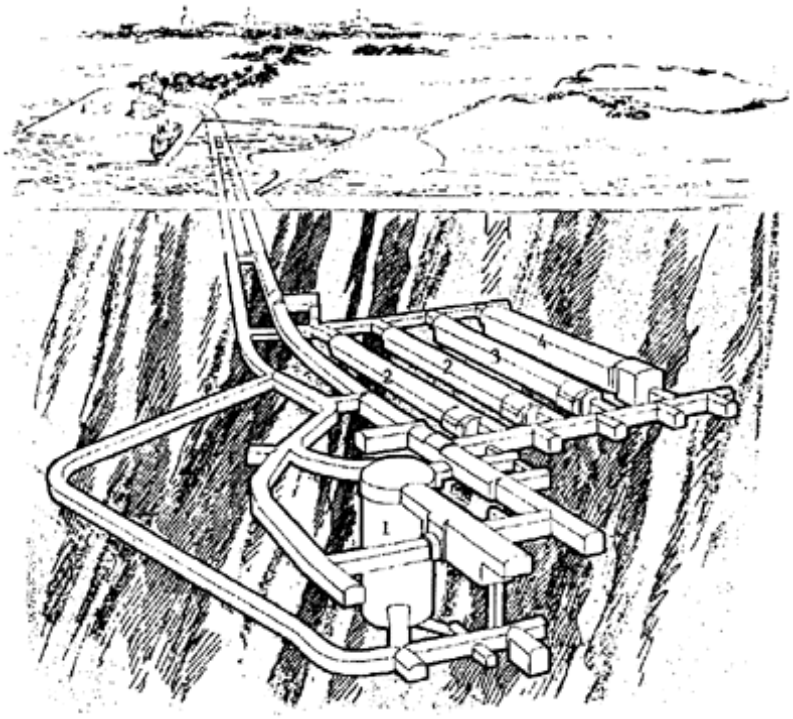


Figure 5: Diagram of the SFR Repository [Ref. 32]

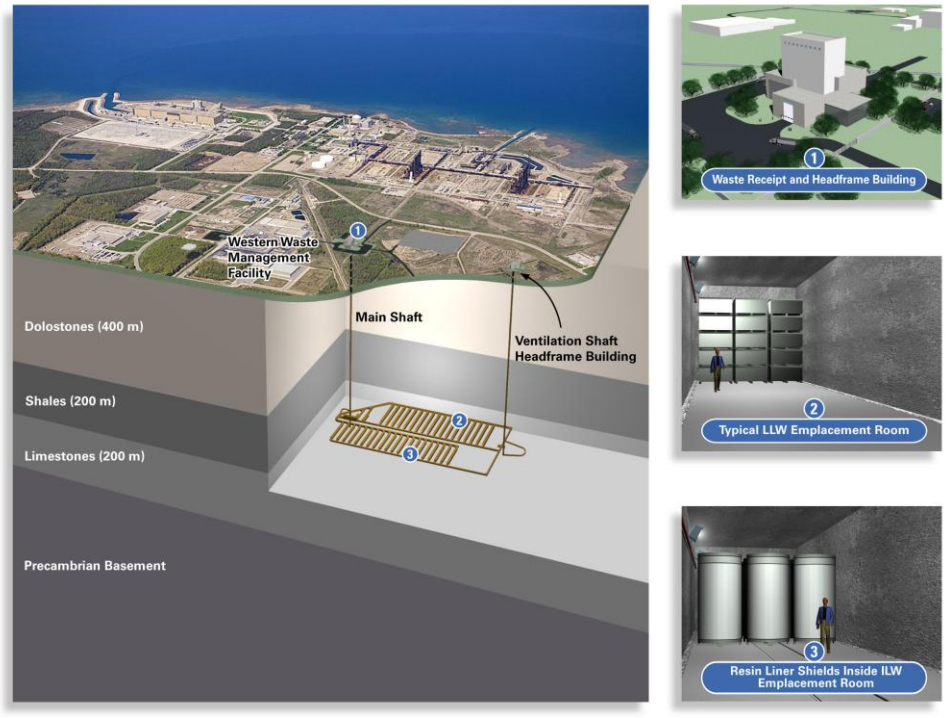


Figure 6. Artist Rendition of OPG's Deep Geologic Disopsal Facility [Ref. 33]

United Kingdom

The United Kingdom's Committee on Radioactive Waste Management (CoRWM) has recommended that all long-lived radioactive waste be disposed in a geologic repository [Ref. 34]. This includes HLW, intermediate-level, and some LLW that does not meet the acceptance criteria for near surface disposal. This recommendation and others made by CoRWM regarding the process for siting and developing such a proposal was accepted by the U.K. government [Ref. 35].

3.2.2 Conversion of Mines to Geologic Repositories

Germany has concentrated on converting previously excavated mines into geologic repositories for isolating intermediate level wastes (non-heat generating) [Ref. 36]. Low and intermediate level wastes were disposed in the Asse (former West Germany) and Morsleben (former East Germany). Operations at Asse, as an experimental repository, were halted in 1978. In 1998 waste emplacement at Morsleben was suspended and will not be resumed. The site continues to be monitored/maintained, and closure activities are scheduled to begin in 2011.

Germany currently plans on using a former iron mine, the Konrad mine, for the disposal of non-heat generating wastes. In 2002, the Konrad mine was license to be converted to a repository that could dispose of up to 303,000 m^3 of radioactive waste. The licensing decision was appealed and the legal process was completed in April of 2007. Wastes will be disposed at depths ranging from 800 to 1,300 meters.

3.2.3 Intermediate Depth Borehole

On July 23, 2007 the U.S. Department of Energy issued a notice of intent (NOI) to prepare an environmental impact statement for the disposal of GTCC LLRW [Ref. 37]. In this NOI, the DOE proposes to construct and operate a new facility or facilities, or use an existing facility, for the disposal of GTCC LLW and GTCC-like waste. One of the concepts that will be evaluated is intermediate depth boreholes, shown schematically in Figure 7. The concept involves the construction of a deep borehole (deeper than 30 meters) in the ground. The wastes are then placed in the borehole up to about 30 meters from the surface, and the remaining space is filled with clean soil.

From 1984 through 1989, the U.S. DOE emplaced high activity LLRW and some transuranic wastes in thirteen intermediate depth boreholes, called greater confinement boreholes at the Nevada Test Site [Ref. 39]. Three meter diameter boreholes were constructed to a depth of 36 meters with the bottom 15 meters being used to dispose wastes. The boreholes were located in unsaturated alluvium with the bottom of each borehole approximately 200 meters above the water table.

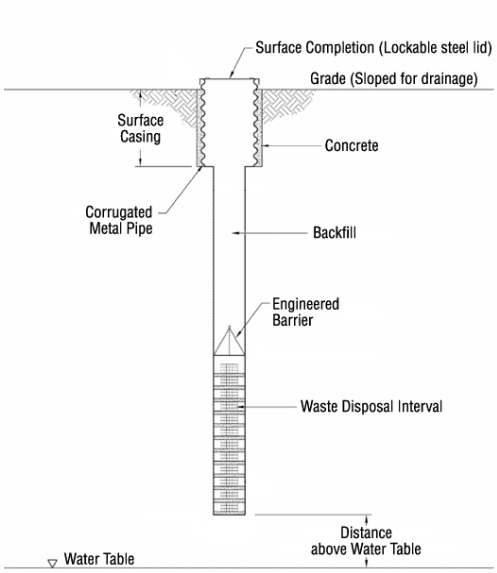


Figure 7. Conceptual Drawing of Intermediate Depth Borehole [Ref. 38]

3.2.4 Enhanced Near Surface Disposal

The U.S. DOE also plans to evaluate the disposal of GTCC LLW and GTCC-like waste in what is termed an enhanced near surface disposal facility. This involves the placement of the wastes in engineered trenches, vaults, or other similar facilities. The containment characteristics of these disposal facilities are enhanced by incorporating features such as barriers, deeper depth to disposal, and enhanced waste packaging. A schematic diagram of a conceptual enhanced near-surface disposal facility being considered is shown in Figure 8.

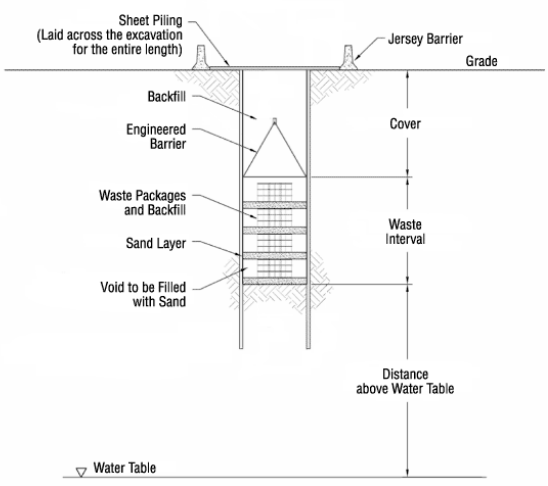


Figure 8. Conceptual Drawing of an Enhanced Near Surface Disposal Facility [Ref. 38]

3.3 Near Surface Disposal Facilities

There are different types of near surface disposal facilities that are being used to dispose primarily of short-lived, LLRW. These include trench facilities, trench facilities with disposal vaults, and above grade disposal vaults. In 2001, the IAEA released a report entitled Technical Considerations in the Design of Near Surface Disposal Facilities for Radioactive Waste [Ref. 40]. This report described technical guidance and information regarding the design objectives and design requirements for near surface disposal systems. It also described several facilities that are currently in operation. In particular, this report discussed the use of engineered barriers in addition to the natural features of the site to isolate wastes.

For LLRW containing short lived radionuclies, the IAEA states that "disposal in trenches with simple engineered barriers might be appropriate, provided that the migration of radionuclides is at an acceptable rate as determined by evaluation of the engineering used." The IAEA also states that "for disposal of LILW [long-lived low or intermediate level waste] with higher levels of radioactivity and/or long lived radionuclides more engineered disposal facilities might be needed [i.e., vaults]."

An example of an above ground vault design facility is the Centre de l'Aube facility in France [Ref. 41], shown in Figure 9. The vaults are designed to isolate the waste from groundwater and to have mechanical integrity for 300 years. The base of the vaults is located above the water table. Vault that dispose durable waste packages are backfilled with gravel and those that dispose less durable waste packages are backfilled with concrete. Each vault is closed with a concrete slab when full. Final closure will involve construction of a sloped engineered cover comprised of several layers of drainage material and clay with a final vegetation cover.

The Barnwell South Carolina site disposes Class A, B, and C LLRW in vaults located in trenches [Ref. 42]. All LLRW waste containers are disposed within concrete vaults that are placed in a trench. Different trench designs, are used based on the classification of the waste. The Class A trench is approximately 1000 feet long, 300 feet wide, and 30 feet deep. The Class B/C trench is 600 feet long, 50 feet wide, and 20 feet deep. Slit trenches (300 feet long, 10 feet wide, and 20 feet deep) are used to dispose higher concentration so as to minimize exposures. The bottom of each trench is located a minimum of five feet above the maximum historically measured water table elevation in the vicinity of the trench. When a vault is full, the space between the vaults is backfilled with clay. Engineered covers are constructed over the backfilled vaults as the trenches fill. The engineered cover consists of a minimum 1 foot thick clay layer, a geosynthetic clay liner, a high density polyethylene liner, a sand drain layer, and a vegetated topsoil cover.

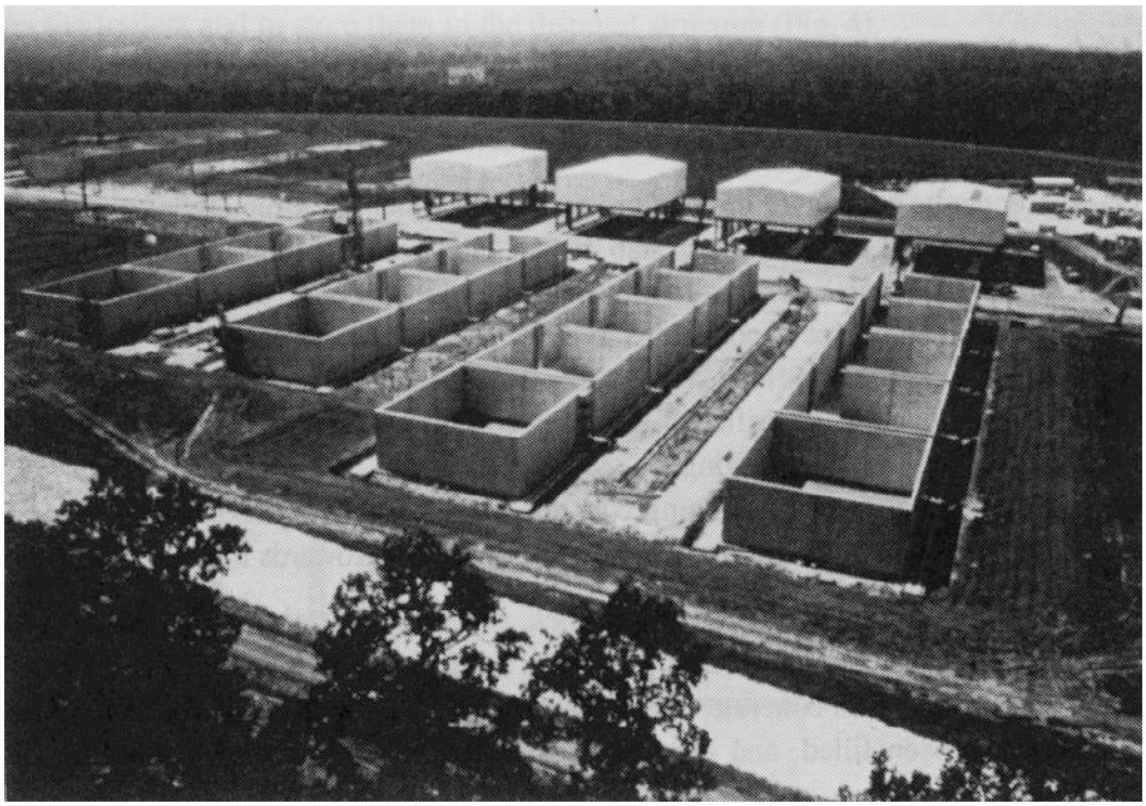


Figure 9. Disposal Vaults at the Centre de l'Aube Facility in France [Ref. 41]

The Richland LLRW disposal facility is a trench design [Ref. 43]. Unstable Class A wastes are segregated from Class B and C wastes and is put directly into trenches. Class B and C wastes are typically placed in high integrity containers or in engineered concrete barriers to achieve required stability and disposed. The trenches are typically 45 feet deep, 850 feet long, and 150 feet wide. An engineered cover is placed on the trenches as they are filled. A conceptual drawing of the facility and one of the covers considered in the facility's EIS [Ref. 48] is shown in Figure 10.

3.4 Storage Facilities

While storage facilities are not intended to be disposal facilities, decay storage could potentially play a significant role in the management of GNEP wastes. As shown in Table 6 above, the ultimate disposition pathway for some of the wastes that could potentially be generated under the GNEP would be enabled only through decay storage. Examples include the decay storage of tritium, krypton, and the heat generating Cs/Sr waste stream. Decay storage on the order of a few decades at most would enable some disposal pathways while very long term storage would be required to enable other pathways.

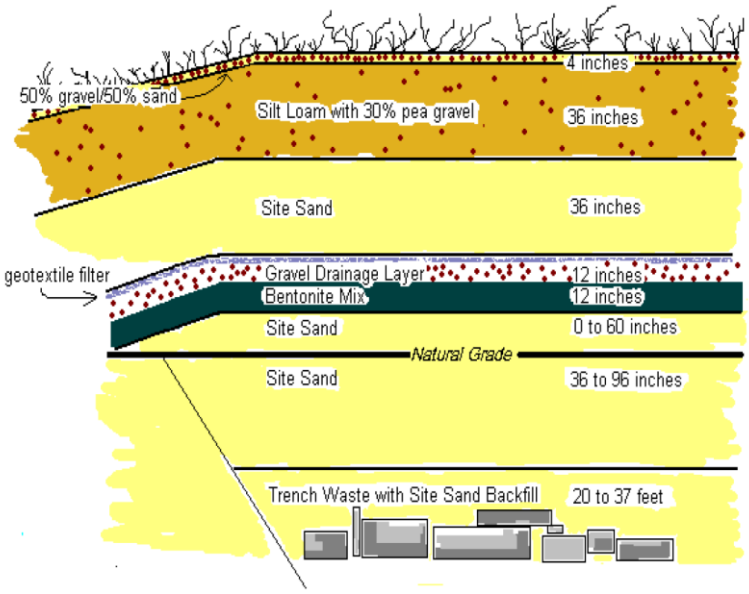


Figure 10. Conceptual Drawing of Richland LLRW Disposal Facility [Ref. 44]

Radioactive wastes have been stored in the United States for a number of years. However, this has not necessarily been by choice, but rather by the fact that a disposal pathway was not available. Although long-term decay storage has occurred, resulting in reductions in activity and heat generation rates, it has not been an integral part of the waste management system by choice. Rather, it was a necessity.

This section does not systematically evaluate options and alternatives for long-term storage, but rather identifies issues regarding long-term storage that must ultimately be considered in a future evaluation. Statements made in reports by the IAEA and the OECD/NEA point out that storage is a necessary part of managing wastes and has been done successfully. Neither organization necessarily precludes consideration of long-term storage. However, they raise issues primarily regarding long-term control and inter-generational equity.

In 2003 the IAEA published a position paper of international experts regarding the long term storage of radioactive waste [Ref. 45]. The paper concluded that:

- Storage is a necessary phase in safely managing most types of radioactive waste. It can allow radiation levels and heat generation rates to decay to manageable levels and is a necessary part of waste treatment and conditioning programs. Storage has been carried out safely over the past decades and there is a high degree of confidences that it can be continued safely for limited periods of time.

- Perpetual storage is not considered to be either feasible or acceptable. Long term safety also requires that future societies will be in a position to exercise active control over these materials and maintain effective transfer of responsibility, knowledge and information from generation to generation. Long term storage is only sustainable if future societies can maintain these responsibilities. Active controls cannot be guaranteed in perpetuity because there is no guarantee that the necessary societal infrastructure can be maintained in perpetuity.

- Storage and disposal are complementary rather than competing activities and both are needed. Strategies for storage and disposal need careful consideration in light of the many issues involved.

In 2006 the OECD Nuclear Energy Agency published a report entitled *The Roles of Storage in the Management of Long-lived Radioactive Waste* [Ref. 46]. This report examined the role that storage plays, or might play, in OECD member countries and draws conclusions on these roles. The findings and conclusions of this report are summarized below.

- Storage of radioactive waste is valuable for:
  - Decay storage – allowing levels of radioactivity and heat to decline before the next step or process in the waste management strategy.
  - Buffer storage – to provide stock for an ongoing process, transportation step, or disposal.
  - Interim storage – waiting for a waste management step to be deployed or while waiting for a decision to be made on the next step.
  - Strategic storage – for materials that may have a potential future use.

- Storage has been done safely and securely for the past several decades. Storage could continue for decades given proper controls, supervision, and maintenance

- Extensive experience and technical knowledge related to storage exists. Storage is firmly regulated.

- Various motives have been put forward for extending storage on the order of 100 years:
  - Practical reasons such as allowing more waste to accumulate before disposal (economic)
  - Strategic reasons such as the possible deployment of regional or multi-national disposal solutions.

- It is not clear that storage facilities are necessarily more acceptable than disposal facilities in general.

- The technical challenges associated with storage increase as the timeframe for storage increases. These include technical challenges, such as facility design and maintenance, and societal challenges, such as maintaining institutional control and funding.

- An "open" solution with indefinite storage is not sustainable because it "implies unquantifed impacts and uses of resources (intergenerational equity – passes the problem to future generations). Storage also cannot be an endpoint in a radioactive waste management strategy. It is important to define the time period for which storage is expected to enable design, cost estimating, licensing, and to indicate the organizational commitment required in the future.

4 Disposal Capacity and Current Status

This section discusses the current status of existing and planned disposal capacity for HLW and LLRW in the United States.

4.1 SNF/HLW (Yucca Mt.)

The development of a repository at Yucca Mountain is proceeding in accordance with the Nuclear Waste Policy Act. The current design of the proposed repository emplaces 63,000 MTHM of commercial spent nuclear fuel and 7,000 MTHM-equivalent of Department of Energy-owned spent nuclear fuel and high level nuclear waste. Efforts are underway by the U.S. DOE Office of Civilian Radioactive Waste Management (OCRWM) to obtain a construction authorization for the Yucca Mountain Repository. OCRWM completed the pre-closure and post-closure safety analyses that demonstrate compliance with the performance objectives at 10 CFR 63. These safety analyses were included in a license application for construction of the repository that was submitted to the U.S. Nuclear Regulatory Commission (NRC) in June of 2008 [Ref. 47].

The best-achievable schedule for initiating operations at the repository is March 2017 and is predicated on adequate funding and NRC's review of the license application within the timelines contained in the NWPA [Ref. 48]. For the past couple of years funding has been authorized at levels less than OCRMW has requested, potentially resulting in further delays to when a repository at Yucca Mountain would become operational.

In fiscal year 2008 OCRWM plans to perform necessary analysis and deliver the report to Congress required by the NWPA on the need for a second repository [Ref. 49]. Several bills have been introduced into Congress to repeal the 70,000 MTHM statutory limit on the amount of waste that could be disposed in a repository at Yucca Mountain. Most recently, a bill entitled the Nuclear Waste Access to Yucca Act, was introduced to the U.S. Senate on May 23, 2007 [Ref. 50]. That bill was referred to the Committee on Energy and Natural Resources where it remains.

4.2 TRU: WIPP

The DOE is operating the Waste Isolation Pilot Plant (WIPP) for the disposal of defense-related transuranic waste. The WIPP is located in a bedded salt formation in southeast New Mexico. The WIPP Land Withdrawal Act [Ref. 51] established the site for the disposal of defense-related transuranic wastes only and established the EPA as the regulatory authority. In accordance with the WIPP Land Withdrawal act, the WIPP is regulated by the EPA under:

- 40 CFR 191: Environmental radiation protection standards for management and disposal of spent nuclear fuel, high-level and transuranic radioactive wastes
- 40 CFR 194: Criteria for the certification and re-certification of the Waste Isolation Pilot Plant's compliance with the 40 CFR Part 191 disposal regulations.

4.3 LLRW

Both existing and planned commercial and federal LLRW capacity is discussed in this section.

4.3.1 Commercial

Most States have entered into compacts as shown in Figure 11. To-date, no new disposal facilities have been built. In 2004 the General Accounting Office provided an overview of the three existing low-level waste disposal facilities in the U.S [Ref. 52]. This overview is further summarized below.

- **EnergySolutions Barnwell Operations, located in Barnwell, South Carolina**
  Currently, Barnwell accepts waste from all U.S. generators except those in the

Rocky Mountain and Northwest Compacts. Barnwell is licensed by the State of South Carolina to receive wastes in Classes A-C.

As of 2004, about 102 acres of the 235-acre site had been filled, with about 13 acres left for disposal. According to the operator, there were about 2.7 million cubic feet of space remaining in 2004. The vast majority of this remaining space, about 2.2 million cubic feet, has been set aside for the decommissioning of the 12 nuclear power plants in the three state compact region. The decommissioning waste is anticipated at about 12,000 cubic feet per facility annually, beginning around 2031 and lasting for about 20 years.

The Barnwell disposal facility is planned for closure to out-of-compact waste by mid- 2008, accepting only waste from the Atlantic compact states (Connecticut, New Jersey, and South Carolina). In 2001, the South Carolina legislature imposed volume caps on the amount of waste that could be accepted at Barnwell. Between 2001 and 2008, the facility is allowed to accept decreasing levels of waste until it reaches a steady state level of 35,000 cubic feet in 2008.

On February 15, 2007, a bill was introduced in the SC House to allow out of compact waste to be accepted for disposal at the Barnwell site after July 1, 2008 [Ref. 53]. That bill was referred to the South Carolina House Committee on Agriculture, Natural Resources and Environmental Affairs on February 15, 2007 [Ref. 54] and on March 28th the bill did not receive a favorable recommendation [Ref. 55]. The bill was rejected in March 2007 [Ref. 56].

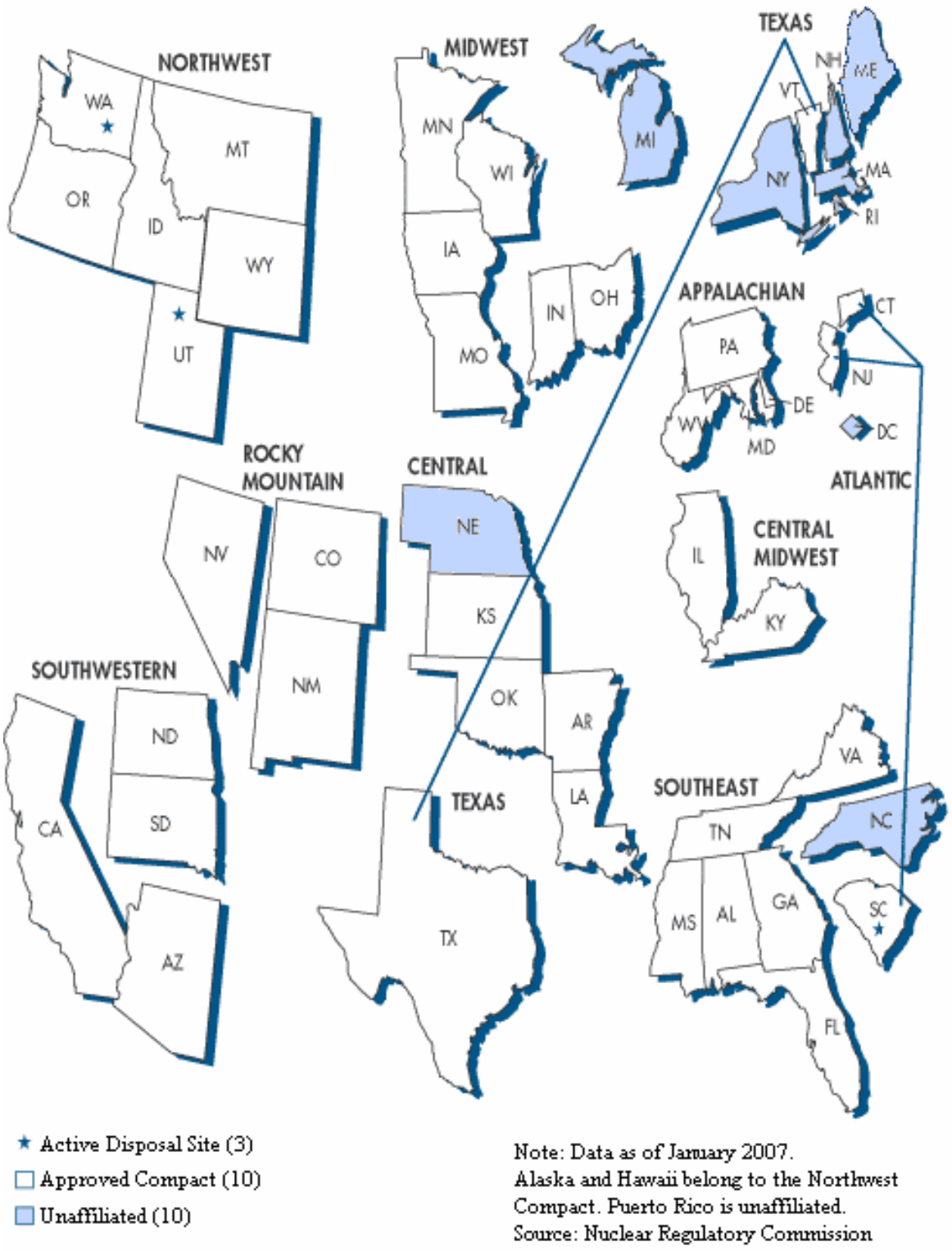


Figure 11. Low-Level Radioactive Waste Disposal Compacts

- **U.S. Ecology, located in Richland, Washington**
  Richland accepts waste from the Northwest and Rocky Mountain Compacts. Richland is licensed by the State of Washington to receive wastes in Classes A-C.

  As of 2004, the Richland facility had much unused capacity to accept LLRW. According to state regulators and company officials, the remaining capacity at Richland is approximately 21 million cubic feet. As of 2004 the facility has disposed of approximately 13.9 million cubic feet of LLRW in 20 trenches. About 95 percent of the waste received is class A. There has been a significant decline in disposal volumes since 1993, when the Northwest Compact placed restrictions on the origin of the waste that the Richland disposal facility could accept.

- **EnergySolutions Clive Operations, located in Clive, Utah**
  Clive accepts waste from all regions of the United States. Clive is licensed by the State of Utah for Class A waste only.

  The disposal site has the capacity for more than 20 years of disposal under its current license. According to EnergySolutions, at the beginning of March 2004 the disposal facility had 58.9 million cubic feet of class A waste. It is anticipated that the disposal facility will accommodate more than 20 years of waste for several reasons, such as a reduction in the annual disposal of waste at the Clive facility.

On August 4, 2004, Waste Control Specialists LLC submitted to the Texas Commission on Environmental Quality an application for a license to authorize near-surface land disposal of low-level radioactive waste [Ref. 57]. This license application seeks authorization to construct and operate in Andrews County a facility that will receive both compact states' waste and federal waste for disposal.

The license application [Ref. 58] considers a total of 80,000 m^3 of Class A (70,000 m^3), Class B (6,900 m^3), and Class C (900 m^3) wastes generated within the Texas compact being disposed in a facility dedicated to disposing commercially generated wastes. The application considers a total of 1,600,000 m^3 of Class A (1,000,000 m^3), mixed Class A (100,000 m^3), Class B/C (445,000 m^3), and mixed Class B/C (6400 m^3) being disposed in facilities dedicated to disposing waste primarily U.S. Department of Energy sites (not confined to those within the Texas compact).

Federal

In 2000, the DOE released Revision 2 of the Low-Level Waste Disposal Capacity Report [Ref. 59]. The purpose of this report was to assess whether DOE disposal facilities have sufficient volumetric and radiological capacity to accommodate the low-level waste (LLW) and mixed low-level waste (MLLW) that the DOE expects to dispose at these facilities. The report concluded that the DOE has sufficient complex-wide volumetric capacity for LLW disposal through 2070 and that the radiological capacity also appears to be sufficient

Table 7 shows the estimated volume and projected disposition of DOE's LLW and MLLW though 2070. It can be seen that the vast majority of the LLW and MLLW that will be disposed is CERCLA wastes. A distinction is made between DOE's waste operations facilities and CERCLA disposal facilities because the CERCLA facilities can only receive waste generated from on-site environmental restoration activities.

In contrast, the waste operations facilities can receive waste from both environmental restoration and other activities. DOE also plans to utilize commercial disposal facilities for a significant amount of LLW and MLLW.

Table 8 shows the estimated volume of LLW that will be disposed in and the capacity of each DOE waste operations disposal facility. Two of the disposal facilities shown in Table 8, the Hanford and Nevada Test Site, are regional facilities. The other facilities are for disposal of on-site wastes only. Table 8 shows that there is a significant amount of excess capacity at the regional disposal facilities (Hanford and the Nevada Test Sites) and at the Los Alamos National Laboratory waste operations disposal facility. In total, there is over 5.5 million cubic meters of excess capacity at these facilities.

4.4 GTCC

On July 23, 2007 the U.S. Department of Energy issued a notice of intent (NOI) to prepare an environmental impact statement for the disposal of GTCC LLRW [Ref. 37]. In this NOI, the DOE proposes to construct and operate a new facility or facilities, or use an existing facility, for the disposal of GTCC LLW and GTCC-like waste. As stated in the NOI, the GTCC LLRW is to be disposed of in a facility licensed and determined to be adequate by the NRC. The NOI states that several alternatives are under consideration. These alternatives are:

Table 7. Estimated Volume and Projected Disposition of DOE's LLW and MLLW through 2070 [Ref. 59]

Projected Disposition	Estimated Volume (m^3)		Totals[a]
	LLW	MLLW	
Waste Operations Disposal Facilities	1,200,000	63,000	1,200,000
Existing/Approved Environmental Restoration CERCLA[b] Disposal Facilities	7,500,000	200,000	7,700,000
Planned Environmental Restoration CERCLA[b] Facilities	170,000	37,000	200,000
To Be Determined	280,000	5,100	280,000
Commercial Disposal	1,000,000	150,000	1,200,000
Totals[a]	10,100,000	450,000	10,600,000

[a]because of rounding, some totals may not equal the sum of their components.
[b]CERCLA - Comprehensive Environmental Response, Compensation, and Liability Act
Source: Low-Level Waste Disposal Capacity Report, Revision 2, Table ES-1.

Table 8. Estimated Volume and Capacity of DOE's Waste Operations Disposal Facilities [Ref. 59]

Waste Operations Disposal Facility	Projected Volume (m^3)	Capacity (m^3)	Excess Capacity (m^3)
Regional Disposal Facilities			
200 Area Burial Grounds	380,000	2,000,000	1,620,000
NTS Areas 3 and 5 Radioactive Waste Management Site	1,100,000	3,700,000	2,600,000
Total Regional	1,480,000	5,700,000	4,220,000

On-Site Disposal Facilities			
INL Radioactive Waste Management Complex	51,000	97,000	46,000
LANL Technical Area-54 Area G	320,000	1,600,000	1,280,000
ORR Interim Waste Management Facility	5,400	5,400	0
SRS Low Activity Waste Vaults	48,000	61,000	13,000
SRS Intermediate Level Waste Vaults	6,900	14,600	7,700
SRS E-Area Trenches	64,000	170,000	106,000
Total	495,300	1,948,000	1,452,700
Source: Low-Level Waste Disposal Capacity Report, Revision 2, Tables 2-6 and 2-8			

- disposal in geologic repositories at Yucca Mountain and at the Waste Isolation Pilot Plant (WIPP);
- disposal at a new Enhanced Near-Surface Faclity at the Idaho National Laboratory, Los Alamos National Laboratory, Savannah River Site, Nevada Test Site, Oak Ridge Reservation, Hanford Site, or in the vicinity of the WIPP; and
- disposal at a new Intermediate Depth Borehole Faclity at the Idaho National Laboratory, Los Alamos National Laboratory, Savannah River Site, Nevada Test Site, Oak Ridge Reservation, Hanford Site, or in the vicinity of the WIPP.

In the NOI the DOE proposes to evaluate alternatives for the disposal of both GTCC LLRW and DOE GTCC-like waste in this EIS. The DOE owns and generates LLRW and transuranic radioactive waste with characteristics similar to GTCC LLRW and that may not have a path to disposal and calls this DOE-GTCC like wastes.

The estimated quantities of waste that will be analyzed are 2,600 m^3 of GTCC-LLRW and 3,000 m^3 of DOE-GTCC like waste [Ref. 60]. The estimate included quantities of GTCC LLW and DOE GTCC-like waste in storage and the projected through 2035. Nuclear utility GTCC projections are made to 2062 and considered the GTCC that would arise from the decommissioning of the 104 nuclear reactors currently operating and from 18 decommissioned reactors. The quantity of DOE-GTCC like waste was estimated by reviewing DOE databases and other documented information sources. In addition, DOE issued a complex-wide data call in August 2005 to obtain additional information on stored and projected DOE GTCC-like waste through 2035 and beyond.

The NOI states that based on the EIS analysis, DOE expects to make a decision on the method(s) and location(s) for disposing of GTCC LLW and DOE GTCC-like waste. A combination of disposal methods and locations may be appropriate based on the characteristics of the waste and other factors.

5 Policy and Regulatory Framework Analysis

This section discusses the policy and regulatory framework regarding the disposal of HLW (and SNF) and LLRW (including GTCC). The current policy and regulatory framework, issues associated with each disposal pathway, and potential options and alternatives are discussed for each. One of the most important aspects of the regulatory framework is the waste classification or definition, in particular for

HLW. This was discussed above and is not repeated in this section. Nevertheless, the waste classification/definition should be considered in any revision to HLW disposal policy and regulations.

5.1 HLW

The disposal of HLW generated under the GNEP has historically focused on its disposal in a repository at Yucca Mountain. However, current policy in the United States does not support this disposal pathway. While disposal in a repository at Yucca Mountain or in an alternative repository may be feasible, both the policy and/or regulatory framework would have to be revised and/or established to support the disposal path chosen for HLW. This section discusses the policy and regulatory framework associated with HLW disposal and discusses alternatives that could be considered in the future.

5.1.1 HLW Policy and Regulatory Framework

National policy for managing and disposing spent nuclear fuel and high level nuclear waste is established in the Nuclear Waste Policy Act (NWPA) [Ref. 61]. In 1982 the NWPA [Ref. 62] established a process for the nomination of at least five sites suitable for site characterization and the recommendation of three of those sites to be characterized for a first repository and a second site. This ultimately led to the selection of the Deaf Smith County (bedded salt), Hanford (basalt), and Yucca Mountain (tuff) sites for characterization for the first repository site. In 1987 the NWPA was amended to terminate site characterization activities at all other candidate sites other than the Yucca Mountain Site (Subtitle E). In addition, activities related to a second repository site were also terminated.

The NWPA "prohibit[s] the emplacement in the first repository of a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel until such time as a second repository is in operation" (Sec. 114d). The NWPA states (Sec. 161) that site-specific activities with respect to a second repository may not be conducted unless Congress has specifically authorized and appropriated funds for such activities. Further, the NWPA directs the Secretary of Energy to report to the President and to Congress between January 2007 and January 2010 on the need for a second repository.

The NWPA directed the Environmental Protection Agency (EPA) (Sec. 121a) to "promulgate generally applicable standards for protection of the general environment from offsite releases from radioactive material in repositories." The NWPA also directed the Nuclear Regulatory Commission (NRC) (Sec. 121b) to promulgate technical requirements and criteria that will apply in approving or disapproving 1) applications for authorization to construct repositories; 2) applications for licenses to receive and possess spent nuclear fuel and high-level radioactive waste in such repositories; and 3) applications for authorization for closure and decommissioning of such repositories. The technical requirements and criteria promulgated by the NRC shall not be inconsistent with any comparable standards promulgated by the EPA.

In 1981 the NRC issued 10 CFR 60 [Ref. 63] pursuant to the authority established in the Energy Reorganization Act of 1974. This version of 10 CFR 60 set forth the requirements applicable to the DOE for submitting an application for a license and specified the procedures which the NRC would follow in considering the application. In 1983 the licensing procedures in 10 CFR 60 were supplemented with technical criteria established in Subpart E [Ref. 64]. These technical criteria include:

- 10 CFR 60.112: "The geologic setting shall be selected and the engineered barrier system and the shafts, boreholes and their seals shall be designed to assure that releases of radioactive materials to the accessible environment following permanent closure conform to such generally applicable

environmental standards for radioactivity as may have been established by the Environmental Protection Agency with respect to both anticipated processes and events and unanticipated processes and events." Note that the EPA generally applicable environmental standards had not yet been established.

- 10 CFR 60.113(a)(1)(ii)(A): "Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account the factors specified in § 60.113(b) provided, that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository."
- 10 CFR 60.113(a)(1)(ii)(B): "The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Commission; provided, that this requirement does not apply to any radionuclide which is released at a rate less than 0.1% of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay."
- 10 CFR 60.113(a)(2): "The geologic repository shall be located so that pre-waste-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment shall be at least 1,000 years or such other travel time as may be approved or specified by the Commission."

Following the establishment of the NWPA the EPA promulgated 40 CFR 191 in 1985 [Ref. 65] that established the generally applicable environmental standards for the management and disposal of spent nuclear fuel, high-level radioactive wastes, and transuranic radioactive wastes. These generally applicable requirements were invoked by the NRC regulations through the requirements at 10 CFR 63.112. In particular, 40 CFR 191 established in Subpart B:

- the long-term containment requirements that limit projected releases of radioactivity to the accessible environment for 10,000 years after disposal;
- annual limitations on individual members of the public in the accessible environment at 25 mrem/yr to the whole body or 75 mrem to any critical organ for 1,000 years after disposal; and
- a set of ground water protection requirements that limit radionuclide concentrations in ground water for 1,000 years after disposal.

The NRC subsequently revised 10 CFR 60 in 1986 [Ref. 66] to conform the licensing procedures to the provisions of the NWPA. However, the technical criteria established in 10 CFR 60 Subpart E, in particular those at 10 CFR 112 and 113 were unchanged. Thus, the NRC regulations in 10 CFR 60 included the EPA containment, individual protection requirements, and ground water requirements under 10 CFR 60.112 along with additional performance objectives at 10 CFR 60.113.

In 1993 the EPA revised 40 CFR 191 [Ref. 67] in response to legal rulings [Ref. 68] and the WIPP Land Withdrawal Act [Ref. 51]. This revision resulted in changes to the individual protection and ground water protection requirements. The compliance period for both the individual and ground water protection requirements were extended from 1,000 to 10,000 years after disposal. In addition, the individual protection requirement was changed to annual committed effective dose received through all potential pathways from the disposal system with a limit of 15 mrem. The ground water protection requirements invoked the limits specified at 40 CFR 141 (as they existed on January 19th, 1994). The technical requirements in 10 CFR 60 Subpart E were not changed, thus these revised EPA requirements were included in 10 CFR 60 through the requirements at 10 CFR 63.112.

The Energy Policy Act of 1992 (EnPA) directed the EPA to, "based upon and consistent with the findings and recommendations of the National Academy of Sciences, promulgate, by rule, public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site" and directed that "such standards shall prescribe the maximum annual effective dose equivalent to individual members of the public from releases to the accessible environment from radioactive materials stored or disposed of in the repository."

The National Academy of Sciences completed their study entitled *Technical Bases for Yucca Mountain Standards* in 1995 [Ref. 69]. In 2001 the EPA promulgated 40 CFR 197 [Ref. 70] establishing a limit on exposures to individual members of the public and a set of ground water protection requirements for 10,000 years after disposal. In contrast to 40 CFR 191, 40 CFR 197 does not include the long-term containment requirements. The individual protection limit remained at 15 mrem per year, but is to be determined through performance assessment, which is rigorously defined in 40 CFR 197, for a reasonably maximally exposed individual (RMEI). 40 CFR 197 also included ground water protection requirements, but established limits on combined radium-226 and radium-228, gross alpha activity, and combined beta and photon emitting radionuclides rather than invoking 40 CFR 141.

In 2001 the NRC subsequently revised 10 CFR 60 and promulgated final rules at 10 CFR 63 [Ref. 71]. The NRC left its existing, generic regulations at 10 CFR 60 in place, changing only to state that they do not apply, nor may they be the subject of litigation, in any NRC licensing proceeding for a repository at Yucca Mountain. The performance requirements established in 10 CFR 63 included both individual and ground water protection standards applicable for a 10,000 year compliance period that are consistent with 40 CFR 197 and defined performance assessment as the methodology for demonstrating compliance

In 2004 the U.S. District Court of Appeals for the District of Columbia Circuit vacated portions of 40 CFR 197 and 10 CFR 63 that addressed the period of time for which compliance must be demonstrated, ruling that the time frame for regulatory compliance was not "based upon and consistent with" the findings and recommendations of the National Academy of Sciences. In response, the EPA has issued proposed rules [Ref. 72] that consider a 1,000,000 year compliance period with different individual protection standards that cover a 10,000 year period after disposal (15 mrem annual dose to the RMEI), and for the period thereafter (350 mrem annual dose to the RMEI). The NRC also issued proposed rules [Ref. 73] that implement the EPA's proposal at 40 CFR 197. Final rules have yet to be promulgated.

5.1.2 Future Changes to the HLW Policy and Regulatory Framework

The establishment of National policy regarding the disposal of spent nuclear fuel (SNF) and high level waste (HLW) and the corresponding regulatory framework has been complex and driven by many factors. In summary, the NWPA establishes the Nation's policy for disposing SNF and HLW and directed site characterization activities to focus on the Yucca Mountain Site. Site characterization activities completed when the Yucca Mountain Site was recommended to Congress [Ref. 74] and approved [Ref. 75]. The Yucca Mountain Project is currently developing a license application for authorization to construct a repository and plans to submit it by June 2008.

The NWPA directs the NRC to submit a report to Congress describing "any Commission actions regarding the granting or denial of such authorization" [section 114.(c)(3)]. As discussed above, the NWPA also requires that the Secretary of Energy report to the President and Congress on the need for a second repository. However, any site-specific activities cannot be conducted unless Congress has specifically authorized and appropriated funds for such activities.

While Congressional reporting is required under the NWPA in regard to both the proposed Yucca Mountain repository and a second repository, the NWPA is silent on what would transpire thereafter. Ultimately and regardless of whether the fuel cycle is closed as envisioned under the GNEP, Congressional action will be required to address the disposal of additional wastes, above the 63,000 MTHM authorized for disposal at Yucca Mountain, that would be generated by both existing nuclear power plants and any new plants that may be constructed in the future.

- Although National policy exists in regard to disposing wastes in a repository at Yucca Mountain, Congressional action would be necessary to develop policy for disposing nuclear waste should, for any reason, DOE not be successful in securing a license for a repository at Yucca Mountain and for the development of a second repository. What that policy may entail is uncertain.

- Should a license be secured for a repository at Yucca Mountain, it's capacity is still limited under the NWPA. Bills have been submitted to both the House of Representatives [Ref. 76] and the Senate [Refs. 77 and 78] that remove the 70,000 MTHM limit in the NWPA. However, it is unlikely that these bills would become law in the near future since Senator Harry Reid (D-NV) is the Majority Leader and is adamantly opposed to licensing the Yucca Mountain facility. Legislation would ultimately have to be enacted expanding the capacity of a repository at Yucca Mountain or Congressional action would be necessary to develop policy regarding a second repository. Again, what that policy may entail is uncertain.

A revised National nuclear waste policy that would utilize a repository other than Yucca Mountain could potentially take several different forms. However, past precedent indicates that one of two forms would likely be taken. The first form may be a site nomination - selection – characterization – recommendation process as in the 1982 NWPA. The second form may be the designation of a site for characterization as in the 1987 amendment to the NWPA.

The consideration of volunteer sites may also be a part of a revised nuclear waste policy under any scenario. Although a volunteer process is possible and has been successful internationally, past experience in attempting to utilize volunteer sites have not proven successful within the United States. This is evident in failure to find volunteers willing to consider siting a monitored retrievable facility for spent nuclear fuel and high level waste in the late 1980s and early 1990s. While there may be willingness at the local-level to volunteer, there may not be such willingness at a broader regional level and within the state that would host such a facility. Although not a volunteer site, this type of situation exists in regard to the Yucca Mountain facility where the host county (Nye County) is generally supportive of the repository [Ref. 79] whereas the state of Nevada is adamantly opposed.

The development of a multiple site nomination - selection – characterization – recommendation policy may be perceived as the most "fair" way to select a site for another repository. However, past experience has shown that this can be time consuming and costly. This is evident with the 1987 amendment to the NWPA to move from parallel characterization of three sites to characterization of only the Yucca Mountain site.

The designation by Congress of a single site for the development of a repository would require a strong technical basis for the selection of that site. In spite of this, there would likely be a strong negative reaction within the selected host state as is present in the state of Nevada over the Yucca Mountain Site (although there is local support). Besides having technical issues with the site, the State of Nevada strongly believes that the redirection of site characterization to Yucca Mountain site in 1987 is a violation of state rights.

Possible sites that could potentially be considered in either policy scenario are those sites previously considered by the DOE prior to the re-direction of site characterization activities to the Yucca Mountain Site in 1987. Another possibility may be expanding the WIPP site for the disposal of spent nuclear fuel and high level nuclear waste. Congressional action would be required to revise both the NWPA and the WIPP Land Withdrawal Act. In addition, although there may be local support for such Congressional action (e.g., in the Carlsbad NM area), there may be considerable opposition within the State of New Mexico. Opposition of WIPP and any expansion has already been voiced from such groups as the Southwest Research and Information Center [Ref. 80], the Concerned Citizens for Nuclear Safety [Ref. 81], and Nuclear Watch New Mexico [Ref. 82].

Congressional action regarding nuclear waste policy beyond the Yucca Mountain site would also have to establish a regulatory framework. At present, 40 CFR 191 and 10 CFR 60 as discussed above would be applicable to such a repository. However, the establishment of site specific regulations for the Yucca Mountain site as directed by the Energy Policy Act of 1992 could set a precedent for establishing site specific rules for another repository site. In that these regulations, 40 CFR 197 and 10 CFR 63, are risk based and will cover a 1,000,000 year period, precedent is established for such future site specific rules being similar.

5.2 LLRW (Including GTCC)

This section discusses the LLRW policy and regulatory framework both for commercially generated LLRW and for federally generated LLRW. Issues associated with the disposal of GNEP LLRW is then discussed followed by a discussion of potential future strategies.

5.2.1 LLRW Policy

The Low-Level Radioactive Waste Policy Act (LLRWPA) establishes responsibilities for the disposal of low-level radioactive wastes for both the States and the Federal Government [Ref. 84]. Each State, either by itself or in cooperation with other States, are responsible for the disposal of:

- Class A, B, or C radioactive wastes generated within the state;
- Low-level radioactive waste that is generated by the Federal Government except for waste that is owned or generated by the Department of Energy; and
- Class A, B, or C radioactive waste generated outside the State and accepted for disposal

The Federal Government is responsible for the disposal of:

- Low-level radioactive waste owned or generated by the Department of Energy; and
- Any other low-level radioactive waste with concentrations of radionuclides that exceed the limits for class C radioactive waste.

The LLRWPA also established the policy of the Federal Government that the responsibilities of the States for the disposal of low-level radioactive waste can be most safely and effectively managed on a regional basis and that States may enter into compacts to provide for the establishment and operation of regional disposal facilities for low-level radioactive waste [Ref. 84].

The LLWRPA establishes the authority of States to regulate the disposal of low-level radioactive waste as under an agreement with the U.S. Nuclear Regulatory Commission (as “agreement states”) [Ref. 85].

5.2.2 LLW Regulations

Commercial

The regulations in 10 CFR 61 "establish, for land disposal of radioactive waste, the procedures, criteria, and terms and conditions upon which the Commission [NRC] issues licenses for the disposal of radioactive wastes containing byproduct, source and special nuclear material received from other persons" [Ref. 86].

The NRC classifies low-level radioactive waste into one of four categories at 10 CFR 61.55 (Classes A, B, C, and beyond Class C). The bases for the classifications are described at 10 CFR 61.7 and pertain to protection of the general population from releases of radioactivity, stability of the waste, and protection of individuals from inadvertent human intrusion.

Class C wastes are differentiated on the protection of individuals from inadvertent human intrusion. 10 CFR 61.7(b) states:

> (at item 4) "Institutional control of access to the site is required for up to 100 years. This permits the disposal of Class A and Class B waste without special provisions for intrusion protection, since these classes of waste contain types and quantities of radioisotopes that will decay during the 100-year period and will present an acceptable hazard to an intruder."
>
> (at item 5) "Waste that will not decay to levels which present an acceptable hazard to an intruder within 100 years is designated as Class C waste." Further, "Waste with concentrations above these limits is generally unacceptable for near-surface disposal. There may be some instances where waste with concentrations greater than permitted for Class C would be acceptable for near-surface disposal with special processing or design. These will be evaluated on a case-by-case basis."

Class A and B wastes are differentiated primarily on stability. As discussed at 10 CFR 61.7(b)(2), stability is a cornerstone of the disposal system, stating "stability of the waste and the disposal site so that once emplaced and covered, the access of water to the waste can be minimized. Migration of radionuclides is thus minimized, long-term active maintenance can be avoided, and potential exposures to intruders reduced." Class A wastes do not have sufficient amounts of radionuclides to be of great concern from a stability aspect and are not required to meet stability criteria. Class B and C waste forms should be designed to be stable (i.e., maintain gross physical properties and identity) over 300 years. Class A waste forms must be segregated from Class B and C wastes unless they meet the stability criteria at 10 CFR 61.56(b).

10 CFR 61.55(a)(2)(iv) states that waste that is not generally acceptable for near-surface disposal, greater than Class C (GTCC) must utilize waste forms and disposal methods that are different, and in more general more stringent, than for Class A, B, and C low-level wastes. The section further states that such wastes must be disposed of in a geologic repository as defined in 10 CFR 60 or 10 CFR 63 unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the NRC.

Performance objectives for a low-level radioactive disposal facility are established in Subpart C of 10 CFR 61 as discussed below:

Protection of the general population from releases of radioactivity, established at 10 CFR 6.41 requires that "concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public." No specific time frame for demonstrating compliance is given.

Protection of individuals from inadvertent intrusion, established at 10 CFR 6.41 requires that "design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed." No specific time frame for demonstrating compliance is given.

The relationships between the overall 10 CFR Part 61 data and design requirements, and detailed low-level radioactive waste (LLW) performance assessment needs, are not directly apparent from the existing U.S. Nuclear Regulatory Commission (NRC) guidance documents. To address this concern, NRC's Performance Assessment Working Group (PAWG) has prepared NUREG-1573 [Ref. 87] as a means of providing information and recommendations on performance assessment methodology as it relates to the objective concerned with the radiological protection of the general public established at 10 CFR 61.41.

NUREG-1573 describes a multi-step process for conducting iterative performance assessments of a low-level radioactive waste disposal facility with the ultimate goal of demonstrating compliance with 10 CFR 61.41. In conducting such a performance assessment, NUREG-1573 recommends:

- (in Executive Summary, Section 3.3) that the timeframe of a performance assessment cover a period of 10,000 years "to capture the peak dose from more mobile long-lived radionuclides and to demonstrate the relationship of site suitability to the performance objective." It is further stated that "shorter periods, on the order of 1,000 years are generally considered inappropriate for assessment of LLW facilities."

- (in Executive Summary, Section 3.2) that any period of time claimed for the performance of engineered barriers should be supported by suitable information and technical justification. However, it is further stated that "materials typically used in engineered barriers can alternatively be assumed to have physically degraded after 500 years following site closure" to limit unnecessary speculation. It is argued that for timeframes longer than 500 years, it is unreasonable to assume that any physical engineered barriers, such as a cover or a reinforced concrete vault can be designed to function long enough to influence the eventual release of long-lived radionuclides (i.e., ^{14}C, ^{99}Tc, ^{129}I).

- (in Executive Summary, Section 3.4) that either bounding or probabilistic performance analyses be conducted. In the case of probabilistic performance assessment, it is recommended that the peak of the mean dose as a function of time be less than the performance objective and that the upper 95^{th} percentile of the dose at each discrete point in time be less than 100 mrem. Formal sensitivity and uncertainty analyses are recommended for either approach chosen.

States, as "agreement states" must establish regulations that are compatible with the NRC's program for the regulation of such materials [Ref. 88]. In general, individual states have established regulations that are consistent with 10 CFR 61. However, some states have additional regulations beyond those in 10 CFR 61 relating to the manner in which low-level radioactive wastes are disposed. For example:

- Illinois [Ref. 89] forbids disposal via shallow land burial and requires "the use of above- ground modules or other designs to provide greater confinement".
- Nebraska [Ref. 90] forbids disposal via "traditional shallow land burial as used prior to 1979" and requires that "disposal design shall include above-ground disposal or other technology which contains one or more engineered, artificially constructed barriers to isolate the waste from the surrounding environment" and that "the disposal cells of the facility shall be built above grade levels and designed to meet the zero-release objective."
- North Carolina [Ref. 91] requires that the bottom of a low-level radioactive waste disposal facility shall be at least seven feet above the seasonal high water table.
- Illinois [Ref. 92] forbids disposal via injection wells whereas Texas [Ref. 93] allows for the disposal of wastes in Class I injection wells (defined as a disposal well which inject fluids below the lower-most formation containing an underground source of drinking water within 1/4 mile of the wellbore).

Federal

The objective of DOE order 435.1, stated in Section 1 is to "ensure that all Department of Energy (DOE) radioactive waste is managed in a manner that is protective of worker and public health and safety, and the environment [Ref. 94]. Section 4.b.(1) states that "Radioactive waste shall be managed to protect the publich from exposure to radiation from radioactive materials. Requirements for public radiation protection are in DOE 5400.5, *Radiation Protection of the Public and the Environment*." Section 4.c states "All Radioactive waste shall be managed in accordance with the requirements in DOE M 535.1-1, *Radioactive Waste Management Manual*.

Low-Level Waste requirements are provided in Chapter IV of DOE M 535.1-1 [Ref. 95]. The definition of low-level waste is given in Section A as "Low-level radioactive waste is radioactive waste that is not high-level radioactive waste, spent nuclear fuel, transuranic waste, byproduct material (as defined in section 11e.(2) of the *Atomic Energy Act of 1954*, as amended), or naturally occurring radioactive material." Note that the DOE does not utilize waste classifications as used by the NRC in 10 CFR 61. The requirements for disposal are established at Chapter IV, Section P and are summarized below.

- Dose to representative members of the public shall not exceed 25 mrem in a year total effective dose equivalent form all exposure pathways, excluding the dose from radon and its progeny in air (Section P(1)(a));
- Dose to representative members of the public via the air pathway shall not exceed 10 mrem in a year total effective dose equivalent form all exposure pathways, excluding the dose from radon and its progeny in air (Section P(1)(b));
- Release of radon shall be less than an average flux of 20 $pCi/m^2/s$ at the surface of the disposal facility or alternatively, a limit of 0.5pCi/l of air may be applied at the boundary of the facility (Section P(1)(c));

A site-specific radiological performance assessment shall be prepared and shall included calculations for a 1,000 year period after closure to demonstrate compliance with the performance objectives listed immediately above (Section P(2)). Performance assessment is also used to establish limits on concentrations of radionuclides for disposal based on the performance measures for inadvertent human intrusion. These performance measures are given at Chapter IV, Section P(2)(h) as 100 mrem in a year for chronic exposure scenarios and 500 mrem in a year for acute exposure scenarios (total dose equivalent excluding radon in air). Institutional control shall be assumed for at least 100 years following closure.

NRC Activities

In October, 2007 the NRC staff released SECY-07-1080 entitled *Strategic Assessment of Low-Level Radioactive Waste Regulatory Program* [Ref. 96]. In this paper, the NRC staff provides the results of a strategic assessment of the agency's LLW regulatory program, along with a description of the process used to perform the assessment. The NRC staff identified twenty potential activities that the staff could undertake to improve the LLW regulatory framework were evaluated and prioritized. The staff ranked 7 of these as high priority, as shown in Table XX with a summary of each task provided below

Of interest are tasks that that were given low or medium priority. While the staff considers all the tasks identified to be worthy endeavors, it has no plans at present to schedule work on those that were ranked as having a medium or low priority. Specific low and medium tasks identified by the NRC that could potentially have relevance to the disposal of potential GNEP wastes as LLRW include [Ref. 96[d]]:

- Evaluate potential changes to LLW regulatory program as a result of severe curtailment of disposal capacity (low priority);
- Promulgate rule of disposal of low-activity waste (low priority);
- Develop licensing criteria for greater-than-Class-C disposal facility (medium priority);
- Identify and evaluate potential legislative changes (low priority);and
- Implement major revisions to 10 CFR 61 (low priority).

Table 9. High Priority NRC LLRW Program Tasks [Ref. 96[e]]

Task No.	Task Description	Resources (FTE)	Schedule
1	Review and Update Guidance on Extended Storage of LLW for Materials and Fuel Cycle Licensees, and Review Industry Guidance for Reactors	1.2	Complete 2nd Quarter FY08 (first task) Complete 4th Quarter FY08 (second task)
2	Develop and Implement Guidance on 10 CFR 20.2002 Alternate Disposal Requests	1.3	Initiated 3rd Quarter FY07 Complete 4th Quarter FY08
3	Determine if disposal of large quantities of depleted uranium from enrichment plants warrant change in uranium waste classification	1.4	Initiated 3rd Quarter FY07 Complete 4th Quarter FY08
4	Update Branch Technical Position on Concentration Averaging and Encapsulation	2.0	Begin 2nd Quarter FY08
5	Develop Procedures for Import/Export Reviews	1.0	Initiate in FY09
6	Develop Guidance Document on Alternate Waste Classification (10 CFR 61.58)	3.6 - 4.3	Revisit in FY09
7	Perform Scoping Study on Byproduct Material Financial Assurance	0.2 - 0.4	Revisit in FY09

5.2.3 Issues Relating to Disposing Potential GNEP Wastes as LLRW

This section discusses three key issues regarding the disposal of potential GNEP wastes as LLRW. The first issue discusses responsibility for disposing LLRW that would be generated from a recycling facility while the second and third issues discuss the availability of disposal capacity.

Responsibility

[d] SECY-07-1080, Table C-I.

[e] SECY-07-1080, Table I.

Individual states and compacts would be responsible for the disposal of LLRW generated by commercial nuclear reactors as established in the LLRWPA. However, the responsibility (commercial or Federal) for disposing LLRW that would be generated from recycling facilities is unclear.

The responsibility for disposing LLRW generated by uranium enrichment facilities established in the USEC Privatization Act [Ref. 97] may establish precedent for the responsibility of disposing LLRW generated from recycling facilities. The USEC Privatization Act states that:

- the DOE, at the request of the generator, shall accept for disposal LLRW generated by either United States Enrichment Corporation or any person licensed by the NRC to operate a uranium enrichment facility under applicable sections of the Atomic Energy Act;
- no State or interstate compact shall be liable for the treatment, storage, or disposal of any LLRW attributable to the operation, decontamination, and decommissioning of any uranium enrichment facility; and
- a generator may enter into agreements for the disposal of LLRW with any other person other than the DOE that is authorized by applicable laws and regulations to dispose of such wastes.

LLRW generated by uranium enrichment facilities can be disposed either in Federal or commercial disposal facilities. However, LLRW disposal is expected to be in commercial disposal facilities. Revision 2 of the Low-Level Waste Disposal Capacity Report [Ref. 59] indicates that the LLRW generated at the Portsmouth and Paducah gaseous diffusion plants will be disposed in both Federal and commercial disposal facilities. The environmental impact statement for the National Enrichment Facility states that all LLRW (Class A only) would be disposed in a commercial facility [Ref. 98].

Disposal Capacity

The availability of sufficient capacity for the disposal of commercially generated LLRW is an issue with respect to nuclear power in the United States. The closure of Barnwell in mid-2008 to non-compact states would require Class B and C wastes to be stored on-site at generators in all states except for those in the Northwest Compact (36 total), and possibly the Texas Compact if the Andrews County site is ultimately licensed. The volume of Class B and C wastes generated may not be sufficient to create a commercial interest in developing a new facility. The disposal of Class A wastes is not a near-term issue because the Energy Solutions facility in Clive, Utah has sufficient capacity for disposing wastes from current generators through approximately 2020.

The deployment of new nuclear reactor plants would generate additional LLRW that would have to be disposed. Disposal pathways may exist for Class B and C LLRW depending on where a plant is built (e.g., in a Northwest or Texas Compact state), otherwise on-site storage would be required if new disposal capacity is not developed. The generation of additional Class A wastes could lead to the Clive, Utah site filling faster. This would also require additional on-site storage of larger volume Class A wastes, except possibly for those generators not located in compacts with disposal facilities, if new disposal capacity is not developed.

Deployment of spent fuel recycling facilities under the GNEP would result in the generation of additional LLRW, potentially further challenging disposal capacity if these wastes needed be disposed in a commercial facility. Approximately five-million cubic meters could potentially be available in existing DOE LLRW disposal facilities (the Hanford Site, the Nevada Test Site, and Los Alamos National Laboratory) if these wastes could be disposed in a Federal facility.

Greater Than Class C LLRW

There currently is no facility available for disposing GTCC LLRW. The DOE has issued a notice of intent to develop an EIS for a Federal GTCC disposal facility and the development of that EIS is underway. As discussed above, the GTCC EIS will evaluate the disposal of 5,600 m^3 of GTCC LLRW and DOE GTCC-like waste.

The volume that will be evaluated in the EIS is significantly lower than the volume that would be generated from a spent fuel recycling plant. Current estimates indicate that approximately 900 m^3/yr of GTCC job control, process, and maintenance LLRW would be generated each year from an 800 MTHM/year aqueous processing facility and approximately 700 m^3/yr from a 300 MTHM/year electro-chemical processing facility. The disposal of additional wastes, such as spent fuel hardware, as GTCC LLRW would further increase these volumes.

Regulatory requirements for the disposal of GTCC LLRW are not expected to be established in the near-future. However, it is expected that such regulations would be established to support the development of DOE's GTCC disposal facility.

5.2.4 Potential Strategies Related to Disposing GNEP Wastes as LLRW

This section summarizes issues associated with disposing GNEP LLRW wastes. There are three fundamental issues: responsibility for the LLRW, available disposal capacity for Class A, B, and C LLRW, and available disposal capacity for GTCC LLRW.

Responsibility

It is likely that legislation would have to be enacted to establish the responsibility for disposing LLRW from recycling facilities. It would be advantageous that such legislation be flexible to allow for the disposal of LLRW generated by recycling facilities either in Federal or commercial facilities. Legislation similar to that enacted for the privatization of the USEC enrichment facilities would have this flexibility.

Commercial LLRW Disposal Capacity

The issue of adequate commercial LLRW disposal capacity has been recognized by several organizations. In 2004 the GAO evaluated LLRW disposal capacity in the United States and concluded that [Ref. 99]:

> "Although no shortfall in disposal availability appears imminent, uncertainties remain about future access to disposal facilities. Even with the prospect of new disposal options, there is no guarantee that they will be developed or be available to meet national needs for class B and C wastes disposal. While LLRW generators have options available to mitigate any future disposal

> shortfall, including storing waste, storage is costly and it can lead to increased safety and security risks. Therefore, continued federal oversight of disposal availability and the conditions of stored waste is warranted."

The GAO recommended that:

> "The Congress may wish to consider directing NRC to report to it if LLRW disposal and storage conditions should change enough to warrant congressional evaluation of alternatives to ensure safe, reliable and cost effectiveness of disposal availability."

In commenting on the GAO report, the NRC stated that the GAO provided an accurate summary of the current LLRW disposal activities at that time, of which there has been no significant change, and potential issues that may arise in the future. The NRC stated that given the failure to develop any new sites under the LLRWPA, the GAO should explore alternatives that "would potentially provide a better legal and policy framework for new disposal options for commercial generators of LLRW." In response, the GAO stated that such an evaluation by them was not required "as long as the NRC places no time limits on storage and provides assurance that it is safe and secure, and any shortfalls in disposal capacity would be managed in the short-term." The GAO believes it is the NRC's responsibility to report to Congress on when such an evaluation is needed.

Thus, there has been no effort to evaluate alternatives for increasing the commercial capacity for commercially generated LLRW. This is further evident by the NRC staff determination in SECY-07-1080 that such activities are of low priority. The issue continues as is evident in an article published in the May/June 2007 issue of Radwaste Solutions [Ref. 100]. That article concludes that there is a crisis in regard to commercial disposal capacity.

The issue of LLRW disposal capacity will ultimately need to be solved independently of GNEP. The existing 104 nuclear power plants, other generators of LLRW, and any new plants that are constructed will ultimately need capacity to dispose of their LLRW. However, the GNEP is predicated on the deployment of new reactors and recycling facilities. Capacity for disposing LLRW will be needed to support a growing nuclear enterprise as envisioned by the GNEP. It is possible for the commercial LLRW capacity issue to be solved prior to the deployment of any GNEP facilities, forced by the need to dispose of LLWR from the existing reactor fleet and other generators.
However, it may be prudent for the program to be supportive of and involved in efforts to evaluate alternatives for assuring LLRW disposal capacity in order to assure that a disposal pathway exists for LLRW:

- generated by new reactors deployed as part of the GNEP program; and
- generated at recycling facilities should there be a desire to dispose of the waste in a commercial facility.

GTCC LLRW Disposal Capacity

As discussed above, the DOE is developing an EIS for a Federal facility for disposing 5,600 m^3 of GTCC LLRW and DOE GTCC-like waste. Following completion of this EIS, the DOE will decide a site, or sites, for constructing GTCC disposal capacity. The volume of waste being evaluated in this EIS is significantly less than that which would require disposal under the GNEP. Completion of the EIS followed by a decision to site and construct a Federal GTCC facility would require that the NRC establish a regulatory framework for the disposal of GTCC wastes. Two alternatives exist regarding inclusion of GNEP wastes in a Federal GTCC facility as described below.

- Immediately interact with DOE's Office of Environmental Management to increase the volume of GTCC estimated to require disposal for the inclusion of GNEP wastes and evaluate this volume in the EIS. This would ensure that all sites being evaluated in the EIS would consider the potential volume of GTCC that would be generated by the GNEP along with the projections currently being evaluated in the EIS.
- Follow the DOE Office of Environmental Management's development of the EIS to ensure that it does not preclude the future use, including expansion, of any of the sites for the disposal of GTCC generated by the GNEP. The EIS would likely have to be revised or supplemented in the future to include the volume of GNEP GTCC LLRW.

The second option is likely preferred. Under this option an EIS evaluating the disposal of GTCC in a Federal facility at multiple sites would be completed, the regulatory framework would be established, and a site (or sites) would be designated, and a facility (or facilities) would be constructed. The disposition of GNEP GTCC could be accommodated in a future revision or supplement to the EIS and either a disposal facility at one of the sites could be expanded or a new facility at a different site could be constructed. This approach would also allow the GNEP program to better establish the volumes and characteristics of the GTCC wastes that would require disposal. Estimating the volume and characteristics of the GTCC wastes for inclusion in the initial EIS would require speculation and would be very uncertain. In addition, some of the GNEP wastes that may ultimately be designated for GTCC may not be possible under current law and therefore could not be included in the initial EIS.

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Appendix B

A Survey of Potentially Useful Packages for Storage, Transportation, and Disposition of Un-irradiated, Irradiated, and Source Materials for the Global Nuclear Energy Partnership (GNEP)

**A Survey of Potentially Useful Packages
for Storage, Transportation, and Disposition
of Un-irradiated, Irradiated, and Source Materials
for the Global Nuclear Energy Partnership (GNEP)**

1 Introduction

The Global Nuclear Energy Partnership (GNEP)[iii] will require the use of Type A and Type B packages for the transportation and storage and disposition of nuclear materials and nuclear wastes.[iv] Many packages are currently certified by the Nuclear Regulatory Commission (NRC) under 10 CFR 71, Packaging and Transportation of Nuclear Material, for the transportation of un-irradiated and irradiated nuclear fuel in addition to specific separated fission products such as ^{90}Sr and ^{137}Cs. A number of these packages have been identified using the RAMPAC (Radioactive Material Packaging) Website [http://www.rampac.com/], maintained for EM-60's (Office of Safety Management and Operations) Packaging Certification Program by the Eagle Research Group for the Department of Energy (DOE). The Certificates of Compliance (CofC) and the Safety Evaluation Reports are both available on this Website. This status of NRC's CofC is effective as of August 3, 2007. The NRC Website is also available at http://www.nrc.gov/. A brief description of the GNEP fuel cycle is found in the following paragraph.

Spent Light Water Reactor (LWR) fuel will have to be shipped to the LWR Spent Nuclear Fuel Separation facility for separation using UREX-1A as a baseline technology. At present, this fuel is stored on site at many different utilities, producing commercial nuclear power, until it can be moved to a central long term storage location such as the proposed Yucca Mountain facility in Nevada or processed as described in the GNEP fuel cycle. Isotopes such as ^{90}Sr and ^{137}Cs and uranium from the separation process will be sent to storage. Transuranics, including neptunium, plutonium, americium, curium, californium, etc., will be processed at the Transmutation Fuel Fabrication or Advanced Fuel Cycle Facility (AFCF) into feeder fuel for the Advanced Burner Reactor (ABR) (fast reactor). Initially, this will be an Advanced Burner Test Reactor (ABTR). Fresh fuel for the ABR or ABTR will necessitate remote handling, shielded transport, and heat removal due radiation levels and thermal output somewhat less than found for LWR spent nuclear fuel. Spent feeder fuel from the Advanced Burner Reactor will be shipped to the Transmutation Fuel Separation facility. Processing at the Transmutation Fuel Separation facility will generate fission products such as ^{90}Sr and ^{137}Cs, other fission products, and uranium. Other fission products include ^{85}Kr, ^{3}H, and ^{14}C. Solids such as cladding hulls and assembly hardware will also require disposition. Transuranics and uranium will go to the Transmutation Fuel Fabrication facility for production of more feeder fuel for the Advanced Burner Reactor completing the cyclic process. Inputs to the cyclic process include transuranics from the LWR Spent Nuclear Fuel Separation facility while the outputs are uranium, ^{90}Sr, ^{137}Cs, and other fission products. Co-location of the LWR Spent Nuclear Fuel Separation facility, Advanced Fuel Cycle Facility, Advanced Burner Test Reactor, and Transmutation Fuel Separation facility would minimize near term transportation requirements.

The GNEP fuel cycle is shown pictorially in Figure 1.

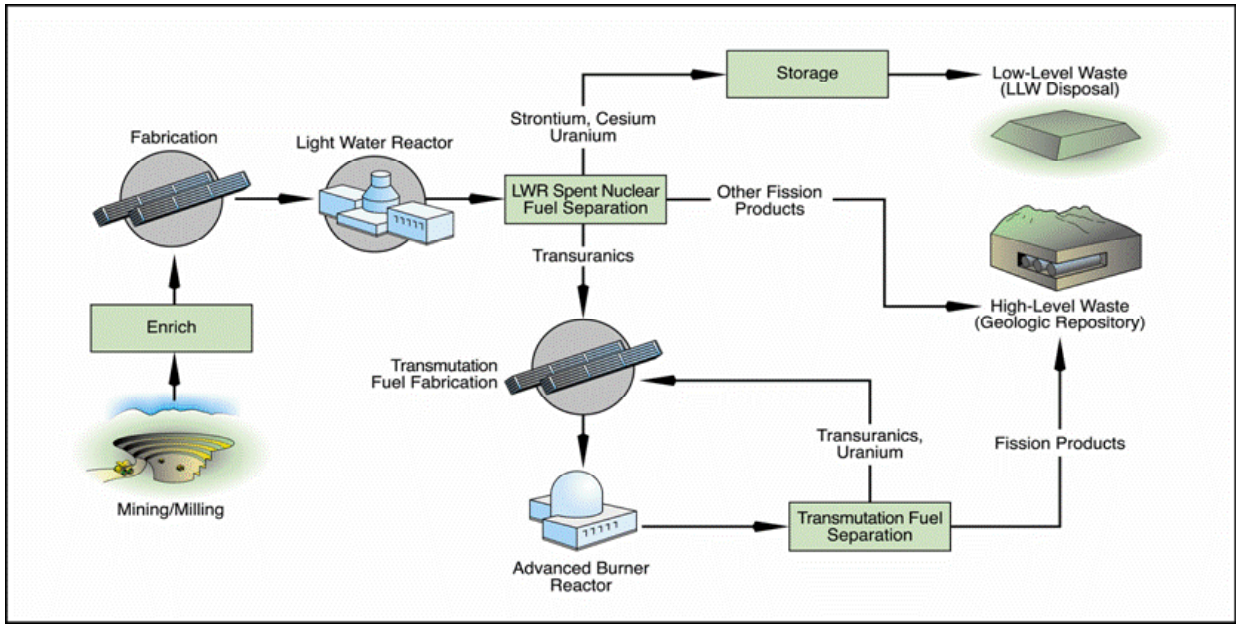

Figure 1. GNEP Fuel Cycle

2 Packages for Transportation of Unirradiated, Irradiated, and Isotopic Materials.

Part 2A. Packages based on Nuclear Regulatory Commission Certificates of Compliance.
Sixty-eight (68) packages with Nuclear Regulatory Commission-approved Certificates of Compliance currently listed on the RAMPAC Website were examined for their characteristics to transport irradiated and un-irradiated nuclear materials and isotopes, for example, ^{90}Sr, ^{137}Cs, and ^{60}Co, where the first two are fission products arising from the separation process. For adding new or revised contents, the NRC typically issues a revision to its CofC. A total of 128 citations for NRC Certificates are currently listed on the RAMPAC Website. A number of the citations (14) are for letters of termination for packages. Others were deemed not relevant to GNEP. It is important to emphasize that the contents of the RAMPAC Website are fluid for the Certificates of Compliance.

A summary of packages for transportation of un-irradiated materials is given in Table I below. Information in the Table includes Certificate Number, Package Identification Number, Package Model Number, company or governmental organization to whom the Certificate is issued, and the expiration date for the Certificate. For the Packaging Certification Number, an example such as USA/9212/B(M)F-85, implies the Certificate is granted by the Nuclear Regulatory Commission for Certificate Number 9212, as a Type B Package for fissile material and develops a Maximum Normal Operating Pressure greater than 100 psig under the conditions of 10 CFR 71.71(c)(1) and that the package meets the regulatory requirements of the International Atomic Energy Agency (IAEA) for their 1985 standard.[v] Thirty-eight packages are described in the Table. An asterisk indicates the certificate is not renewable by NRC. The following packages have an expiration date on or before October 1, 2008: RA-3, 814A, 927A1 and 927C1, Model B, FSV-3, NNFD-10, 6400, 51032-1, CNS 1-13C, BW-2901, and ABB-2901 per 10 CFR 71.19(a)(3):

"A Type B package previously approved by the NRC, but not designated as B(U), B(M), B(U)F, or B(M)F in the identification number of the NRC CofC, or Type AF packages approved by the NRC prior to September 6, 1983, may be used under the general license of 71.17 with the following additional conditions: Paragraph (a) of this section expires October 1, 2008."[vi]

Most of the packages in the table are for transportation of uranium-based fuel although some pertain to mixed oxide fuels containing uranium and plutonium. An example of the latter is the MFFP Package for a Pressurized Water Reactor, based on the MK-BW/MOX 17x17 design. Packages are available for transport of fuel assemblies for Pressurized Water (PWR) and Boiling Water Reactors (BWR). For example, for the former, the Traveller STD and the Traveller XL Packages are available. The CE-B1 is available for the latter reactor type. The 6400 Package can be used for ^{233}U oxide and thorium oxide fuel rods for a Light Water Breeder Reactor (LWBR). The Model 1500 Package can be used for the transportation of $^{90}SrF_2$ and $^{137}CsCl$ capsules.

An EXCEL data base is additionally available with more detailed information providing a summary of each Certificate. The EXCEL data base includes, in addition to the information presented here, a listing of the fuel components, isotopes, uranium enrichment, mass limits for uranium, decay heat, cooling time for spent fuel, and activity where appropriate. Reference to the specific Certificate and respective Safety Analysis Report (SAR) for the package will allow for access to the details. The NRC prefers the term Safety Analysis Report while the Department of Energy uses instead Safety Analysis Report for Packaging (SARP).

Table I. Selected Nuclear Regulatory Commission Certificates of Compliance
for
Radioactive Material Packages

Packages for **Un-irradiated** Materials

Certificate Number	Package Identification Number	Model Number	Issued to	Expiration Date
4986	USA/4986/AF	RA-3	Global Nuclear Fuel-Americas	March 31, 2008
5149	USA/5149/B( )F	814A	BWX Technologies, Inc.	October 1, 2008
5086	USA/5086/B(U)F	UNC-2600	BWXT, Nuclear Products Division	February 28, 2009
5797	USA/5797/B(U)F	Inner HFIR Outer HFIR	DOE	September 30, 2007

Table I. Packages for **Un-irradiated** Materials (Continued)

Certificate Number	Package Identification Number	Model Number	Issued to	Expiration Date
6078*	USA/6078/AF	927A1 927C1	Westinghouse Electric Company	October 1, 2008
6206*	USA/6206/AF	Model B	Framatome ANP, Inc.	October 1, 2008
6347	USA/6347/AF	FSV-3	General Atomics	September 30, 2007
6357*	USA/6357/AF	NNFD-10	BMX Technologies, Inc	October 1, 2008
6400	USA/6400/B( )F	6400	Westinghouse Electric Company	November 30, 2007

6581*	USA/6581/AF	51032-1	Framatome ANP, Inc.	October 1, 2008
9034	USA/9034/AF	TRIGA-I	General Atomics	December 31, 2010
9037	USA/9037/AF	TRIGA-II	General Atomics	December 31, 2010
9081	USA/9081/B( )	CNS 1-13C	Duratek	January 31, 2008
9099	USA/9099/B( )F-85	ATR	DOE	January 31, 2009
9168	USA/9168/B(U)	CNS 8-120B	Duratek	June 30, 2010
9203	USA/9203/AF	DHTF	Framatome ANP, Inc.	February 28, 2011
9204	USA/9204/B(U)-85	CNS 10-160B	Duratek	October 31, 2010
9212	USA/9212/B(M)-85	RH-TRU-72-B	DOE	February 28, 2010
9217	USA/9217/AF	ANF-250	Framatome ANP Richland, Inc.	June 30, 2010
9239	USA/9239/AF	MCC-3, MCC-4, MCC-5	Westinghouse Electric Company	March 31, 2012
9248	USA/9248/AF	SP-1, SP-2, SP-3	Framatome ANP, Inc.	February 28, 2009
9250	USA/9250/B(U)-85	5X22	BWX Technologies	March 31, 2008
9251	USA/9251/AF	BW-2901	Framatome ANP, Inc.	October 31, 2007
9252	USA/9252/AF	51032-2	Framatome ANP, Inc.	October 31, 2008

Table I. Packages for **Un-irradiated** Materials (Continued)

Certificate Number	Package Identification Number	Model Number	Issued to	Expiration Date
9272	USA/9272/AF-85	CE-B1	Westinghouse Electric Company, LLC	January 31, 2007
9274	USA/9274/AF	ABB-2901	Westinghouse Electric Company, LLC	September 30, 2007
9285	USA/9285/AF-85	SRP-1	Global Nuclear Fuel-Americas, LLC	October 31, 2007
9288	USA/9288/B(U)F-96	CHT-OP-TU	Columbiana Hi Tech, LLC	March 31, 2010
9289	USA/9289/B(U)F-85	WE-1	Framatome ANP, Inc.	February 28, 2009
9291	USA/9291/B(U)F-96	Liqui-Rad (LR) Transport Unit Package	Columbiana Hi Tech, LLC	October 31, 2011
9292	USA/9292/AF-85	PATRIOT	Westinghouse Electric Company, LLC	August 31, 2010
9294	USA/9294/AF-85	NPC	Global Nuclear Fuel-Americas,	November 30, 2010

			LLC	
9295	USA/9295/B(U)F-96	MFFP	Packaging Technology, Inc.	June 30, 2010
9297	USA/9297/AF-96	Traveller STD and Traveller XL	Westinghouse Electric Company	March 15, 2010
9301	USA/9301/AF-85	TNF-XI	Packaging Technology, Inc.	August 30, 2008
9309	USA/9309/B(U)F-96	RAJ-II	Global Nuclear Fuel-Americas, LLC	November 30, 2009
9315	USA/9315/B(U)F-96	ES-3100	DOE	April 30, 2011
9328	USA/9328/AF-96	TN-55	Packaging Technology, Inc.	April 30, 2012

Table II below gives a summary of packages for transportation of irradiated materials. An asterisk indicates the Certificate is not renewable by NRC. Twenty-four packages are described in the Table. The following packages have an expiration date on or before October 1, 2008: T-2, GE-100, 1500, BMI-1, IF-300, , NLI-1/2, TN-8 and TN-8L, TN-9, NLI-10/24, CNS 1-13C, and FSV-1 Unit 3 per 10 CFR 71.19(a)(3) as described above in the section under packages for un-irradiated materials.[vi] The T-2 Package can transport irradiated clad fuel as solid metal, oxides, nitrides, and carbides of uranium, plutonium, or mixed uranium and plutonium. The following packages have the capability to transport PWR and BWR spent fuel: IF-300, NLI-1/2, TN-8/TN-8L, NLI-10/24, NAC-LWT, HI-STAR 100 System, UMS Universal Transport Cask, and the Fuel Solutions TS 125 Transportation Package.

Table II. Selected Nuclear Regulatory Commission Certificates of Compliance
for
Radioactive Material Packages

Packages for **Irradiated** Materials

Certificate Number	Package Identification Number	Model Number	Issued to	Expiration Date
5607	USA/5607/B( )F	T-2	DOE	October 1, 2008
5926	USA/5926/B( )F	GE-100	General Electric Company	May 31, 2008
5939*	USA/5939/B( )F	1500	General Electric Company	October 1, 2008
5957	USA/5957/B( )F	BMI-1	DOE	October 1, 2008
9001*	USA/9001/B( )F	IF-300	Duratek	October 1, 2008
9010*	USA/9010/B( )F	NLI-1/2	NAC International, Inc.	October 1, 2008
9015	USA/9015/B( )F	TN-8 TN-8L	Transnuclear, Inc.	October 1, 2008
9016	USA/9016/B( )F	TN-9	Transnuclear, Inc.	October 1, 2008
9023	USA/9023/B( )F	NLI-10/24	NAC International, Inc.	July 31, 2008
9200	USA/9200/B(M)F	125-B	DOE	June 30, 2011
9216	USA/9216/B( )F	CNS 1-13G	Duratek	January 31, 2008
9225	USA/9225/B(U)F-96	NAC-LWT	NAC International, Inc.	February 28, 2010
9226	USA/9226/B(U)F-85	GA-4	General Atomics	October 31, 2008
9228	USA/9228/B(U)F-96	2000	General Electric	May 31, 2011

			Company	
9233	USA/9233/B(U)	TN-RAM	Transnuclear, Inc.	April 30, 2010
9235	USA/9235/B(U)F-96	NAC-STC	NAC International, Inc.	March 31, 2009
9253	USA/9253/B(U)-85	TN-FSV	DOE	May 31, 2009
9255	USA/9255/B(U)F-85	NUHOMS MP187 Multi-Purpose Cask	Transnuclear, Inc.	October 31, 2008
9261	USA/9261/B(U)-85	HI-STAR 100 System	Holtec International	March 31, 2009

Table II. Packages for **Irradiated** Materials (Continued)				
Certificate Number	Package Identification Number	Model Number	Issue to	Expiration Date
9270	USA/9270/B(U)-96	UMS Universal Transport Cask Package	NAC International, Inc.	October 31, 2007
9276	USA/9276/B(U)-85	TS125 Transportation Package	BNFL Fuel Solutions	September 30, 2007
9277*	USA/9277/B( )F	FSV-1 Unit 3	General Atomics	October 1, 2008
9293	USA/9293/B(U)F-85	TN-68 Transport Package	Transnuclear, Inc.	February 28, 2011
9302	USA/9302/B(U)-85	NUHOMS-MP197	Transnuclear, Inc.	July 31, 2007

Table III below summarizes packages for transportation of isotopes such as ^{90}Sr and ^{137}Cs as well as ^{60}Co. The isotopes are either present as special form material or doubly encapsulated. For example The Model Sentinel-25 series can accommodate up to 125,000 curies (Ci) of ^{90}Sr as its titanate or fluoride compounds as special form radioactive material. The 1500 Package can accommodate up to 458,000 Ci $^{90}SrF_2$ or $^{137}CsCl$ in capsule form. Certificate Numbers 4888, 5862, 5939, 5984, 6703, 6786, and 9030 expire on or before October 1, 2008 per 10 CFR 71.19(a)(3) as described in the section pertaining to packages for un-irradiated materials.[vi] Model Numbers RG-1 and 1500 are for doubly encapsulated isotopes while the other Models are special form. One of the contents for the Model 1500 is in special form. The Model 1500 also appears in Table II. The Model 2000, listed under packages for irradiated materials, can accommodate 70,000 Ci ^{60}Co.

Table III. Selected Nuclear Regulatory Commission Certificates of Compliance for Radioactive Material Packages				
Packages for **Isotopes as Special Form** or **Doubly Encapsulated** Materials				
Certificate Number	Package Identification Number	Model Number	Issued to	Expiration Date
4888*	USA/4888/B( )	Sentinel-25A, LCG-25A;Sentinel-25B,LCG-25B;Sentinel-25C,LCG-25C;Sentinel-25C3,-25D,-	Department of the Air Force	October 1, 2008

		25E,-25F		
5862*	USA/5862/B( )	Sentinel-100F	Department of the Air Force	October 1, 2008

Table III. Packages for **Isotopes as Special Form** or **Doubly Encapsulated** Materials (Continued)				
Certificate Number	Package Identification Number	Model Number	Issued to	Expiration Date
5939*	USA/5939/B( )F	1500	General Electric Company	October 1, 2008
5984	USA/5984/B( )	5984	J.L. Shepard & Associates	August 31, 2007
6703*	USA/6703/B( )	RG-1	General Atomics	September 30, 2008
6786*	USA/6786/B( )	URIPS-8A URIPS-8B	Department of the Navy	October 1, 2008
9030*	USA/9030/B( )	MW-3000 Sentinel-8	Department of the Navy	October 1, 2008
*Indicates Certificate not renewable by NRC				

Part 2B. Packages based on Department of Transportation Competent Authority Certificates. Another area, covered by the RAMPAC Website, is issuance of Competent Authority Certification for packages by the United States Department of Transportation (DOT) Pipeline and Hazardous Materials Safety Administration or the DOT Research and Special Programs Administration such that IAEA Regulations for the Safe Transport of Radioactive Material[v] and DOT 49 CFR 100-199[vii] are adhered to by the applicant and user. The package design is "approved for use within the United States for import and export shipments only." Some three hundred and one (301) certificates are available on the RAMPAC Website as DOT-IAEA. Many of the entries pertain to transportation of isotopes particularly for medical or laboratory applications. In a number of instances, NRC Certificates of Compliance are appended to the DOT Competent Authority Certification. About seventeen of the Competent Authority Certifications examined here overlap with NRC CofC, discussed previously, and have relevance to transportation of un-irradiated, irradiated, and isotopic materials as part of GNEP. The Packages include RAJ-II (revalidation of Japanese Competent Authority), RAJ-II (with NRC CofC appended), RA-3, 51032-1 (DOT Research and Special Programs Administration), TRIGA-I, TRIGA-II, NAC-LWT, 2000, MCC-3, MCC-4, and MCC-5, SP-1, SP-2, and SP-3 (DOT Research and Special Programs Administration), 5X22, ABB-2901, SRP-1 (DOT Research and Special Programs Administration), PATRIOT, NPC, Traveller STD and Traveller XL, TNF-XI, and ES-3100. These packages do not have the restriction of "approved for use within the United States for import and export shipments only." since they were previously reviewed by the NRC for domestic use.

Thirty (30) Competent Authority Certifications, evaluated here, are revalidation of competent authority for other nations including the United Kingdom, Canada, Japan, France, and Germany. Documentation such as a Competent Authority Certification is necessary for each nation the package must pass through from point of origin to destination. The Competent Authority Certifications examined here are those potentially relevant to GNEP. They are enumerated in Table IV. Ten of the packages are for irradiated materials while thirteen are for transportation of isotopes, in some instances as special form materials. One package for irradiated materials (JMS-87Y-18.5T) appears three times as the approved contents vary among the Certificate Numbers. The 7N-2 package appears twice but for different irradiated contents in

each case. The AECL-CRL Irradiated Material Transportation Package can be used for shipment of spent CANDU fuel. The Croft Associates Model 2773A (SAFSHIELD) can transport up to 1,180 g ^{137}Cs in special form. The MDS Nordion F-168 and F-168-X is allowed 1×10^5 Ci of ^{137}Cs in special form (similar to the SAFSHIELD). Even though the latter application is for use as a source, it gives an idea of the magnitude of ^{137}Cs allowed for shipments.

Table IV. Selected United States Department of Transportation Competent Authority Certification for Radioactive Material Packages

Certificate Number (USA)	Package Identification Number	Model Number	Issued to	Expiration Date
0208**	USA/0208/B(U)F-96 J/61/B(U)F-96	JRC-80Y-20T	Japan	September 8, 2008
0337***	USA/0337/B(U)-96 GB/2773A/B(U)-96	Croft Associates 2773A (SAFSHIELD)	United Kingdom	December 31, 2009
0371**	USA/0371/B(U)F-85 D/4160/B(U)-85	TN 7-2 Transport Package	Germany	December 31, 2008
0382***	USA/0382/B(U)-96 GB/2835A/B(U)-96	Croft Associates 2835A	United Kingdom	July 31, 2012
0401**	USA/0401/B(U)F-96 J/111/B(U)F-96	JMS-87Y-18.5T	Japan	October 12, 2009
0452**	USA/0452/B(U)F-96 J/119/B(U)F-96	JRF-90Y-950K	Japan	October 12, 2009
0453*&***	USA/0453/S	6810/143-512	IAEA Certificate of Competent Authority for J.L. Shepherd & Associates	September 30, 2009
0460	USA/0460/AF-96 D/4306/AF-96	RA-3D	Germany	July 31, 2008
0464*&***	USA/0464/S	6810-190	IAEA Certificate of Competent Authority for J.L. Shepherd & Associates	September 30, 2009
0485	USA/0485/B(U)F CDN/4212/B(U)F	4H (Serial Numbers 1 to 8)	Canada	April 30, 2009

Table IV (Continued)

Certificate Number (USA)	Package Identification Number	Model Number	Issued to	Expiration Date
0490	USA/0490/AF-96 J/37/AF-96	NT-IV	Japan	May 25, 2009
0542	USA/0542/AF-96 J/134/AF-96	NFI-V	Japan	January 16, 2009
0545***	USA/0545/B(U)-96 GB/3605C/B(U)-96	3605C (multiple isotopes)	United Kingdom	September 30, 2007
0551**	USA/0551/B(U)F-85	GNS-16	Germany	November 23,

	D/4326/B(U)F-85			2008
0553**	USA/0553/B(U)F-85 CDN/2061/B(U)F-85	Irradiated Material Transportation Package	Canada	May 31, 2010
0558**	USA/0558/B(U)F-96 J/150/B(U)F-96	JMS-87Y-18.5T (Kyoto University)	Japan	October 31, 2009
0573**	USA/0573/B(U)F-85 D/4342/B(U)F-85	TN 7-2 Irradiated Fuel Assembly Cask	Germany	December 31, 2008
0587*&***	USA/0587/B(U)-85 CDN/2067/B(U)-85	MDS Nordion Gammacell 40MK3 Irradiator (Serial Numbers 11 and Subsequent)	Canada	February 29, 2008
0595	USA/0595/AF-96 J/156/AF-96	RAJ-III (1996)	Japan	May 31, 2008
0617***	USA/0617/B(U)-96 CDN/2081/B(U)-96	MDS Nordion F-168 (Serial Numbers 53-76 & 83-up) F-168-X [1996] (Serial Numbers 77-X, & up)	Canada	November 30, 2007
0629*&***	USA/0629/S	X.14 & X14/1 (^{241}Am)	IAEA Certificate of Competent Authority for AEA Technology QSA, Inc.	July 31, 2008
0653	USA/0653/AF-96 F/381/AF-96	TNF-XI	France	December 31, 2011

Table IV (Continued)

Certificate Number (USA)	Package Identification Number	Model Number	Issued to	Expiration Date
0665***	USA/0665/B(U)-96 CDN/2083/B(U)-96	MDS Nordion F431/GC-1000 F431/GC-3000	Canada	November 30, 2007
0674***	USA/0674/B(U)-96 CDN/2076/B(U)-96	MDS Nordion Model No. F-430/GC-40, F-430/GC-1000 & GC-3000, F-430/CIS Model IBL 437C, F430/CIS Model IBL 637, F-430/Molsgaard Model GC-2000	Canada	February 28, 2011

0696***	USA/0696/S-96	QSA Global Inc. Model II Source Capsule (multiple isotopes)	IAEA Certificate of Competent Authority for QSA Global, Inc.	February 28, 2011
0713**	USA/0713/B(U)F-96 J/166/B(U)F-96	JMS-87Y-18.5T (Musashi Institute)	Japan	March 16, 2008
0742**	USA/0742/B(U)F-96 J/167/B(U)-96	JRF-90Y-950K	Japan	July 20, 2008
0745	USA/0745/AF-96 D/4365/AF-96	ANF-50	Germany	January 1, 2008
6217*&***	USA/6217/B(U) CDN/2003/B(U)	MDS Nordion F-143 Transfer Case, Serial Numbers 20, 50, 53, 54, 59, 62 & 64 F-158 Transfer Case, Serial Numbers 3-6, 8-10 & 14	Canada	March 31, 2008

Table IV (Continued)

Certificate Number (USA)	Package Identification Number	Model Number	Issued to	Expiration Date
6355***	USA/6355/B(U) CDN/2009/B(U)	MDS Nordion F-147 Transfer Case, Serial Numbers 18, 24, 26, 27, 34-36, 39-48, 50, 52, 54, 56-60	Canada	November 30, 2010

*Approved by DOT Research and Special Programs Administration
**Packages for Irradiated Materials
***Packages for Isotopes

An EXCEL data base, separate from the one discussed above in Part 2A, is additionally available with more detailed information providing a summary of each Competent Authority Certification. The EXCEL data base includes, in addition to the information presented here, a listing of the fuel components, isotopes, uranium enrichment, mass limits for uranium, decay heat, cooling time for spent fuel, and activity where appropriate. Reference to the specific Certificate and respective Safety Analysis Report or Safety Analysis Report for Packaging for the package will allow for access to the details.

3 Conclusions

Some of the packages, approved by the NRC with Certificates of Compliance, certainly have viability for short term demonstration of GNEP where facilities are co-located. However, for long-term demonstration

of GNEP where other than domestic shipments are required, packages are needed for transportation of large activities of transuranic elements in addition to plutonium. The presence of transuranic elements will create new contents and shielding issues that, at the very minimum, will lead to a review and revision of the SAR or SARP if not outright design of new packages to accommodate higher photon and neutron radiation fluxes from feeder fuel going to the ABTR or ABR.

A SAR for a proposed, new ATR fresh fuel cask (Certificate Number 9330) has been submitted to the NRC for review. AREVA has submitted the TN-40 cask for certification. The TN-40 can accommodate up to 40 irradiated PWR assemblies. AREVA has the TN-68 cask for up to 68 irradiated BWR assemblies. The MP-187 (AREVA) can accommodate 24 irradiated PWR assemblies.

Storage of nuclear materials in packages, designed for transportation, is currently a contentious issue for packages certified by EM-60's Packaging Certification Program for the Department of Energy. Over time, flammable gas concentrations will increase and radiation damage to primary containment boundary components such as O-rings will accumulate. Also, the issue of yearly maintenance to the DOE-certified packages currently in use for storage of nuclear materials requires resolution. The Model 9975 Package in use at Savannah River National Laboratory is one example of a transportation package accommodating storage of nuclear materials.

References

[a] GNEP—Basis Document, Integrated Strategy for Nuclear Material Transportation, Storage, and Disposal Strategy Under the Global Nuclear Energy Partnership, Review Draft 2—October 4, 2006.

[a] Transportation and Storage Regulations Applicable to the GNEP, Richard H. Yoshimura, Paul McConnell, Ken B. Sorenson, Sandia National Laboratory, January 31, 2007.

[a] Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[a] Packaging and Transportation of Radioactive Material, Code of Federal Regulations, Title 10, Part 71, Washington, DC (December 2006).

[a] Title 49, Code of Federal Regulations, Parts 100–199, United States of America, Washington, DC (October 1, 2006).

[i] GNEP—Basis Document, *Integrated Strategy for Nuclear Material Transportation, Storage, and Disposal Strategy Under the Global Nuclear Energy Partnership*, Review Draft 2—October 4, 2006. See Appendix A of this report.

[ii] *Transportation and Storage Regulations Applicable to the GNEP*, Richard H. Yoshimura, Paul McConnell, Ken B. Sorenson, Sandia National Laboratory, January 31, 2007.

[iii] GNEP—Basis Document, *Integrated Strategy for Nuclear Material Transportation, Storage, and Disposal Strategy Under the Global Nuclear Energy Partnership*, Review Draft 2—October 4, 2006.

[iv] *Transportation and Storage Regulations Applicable to the GNEP*, Richard H. Yoshimura, Paul McConnell, Ken B. Sorenson, Sandia National Laboratory, January 31, 2007.

[v] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[vi] *Packaging and Transportation of Radioactive Material*, Code of Federal Regulations, Title 10, Part 71, Washington, DC (December 2006).

[vii] Title 49, Code of Federal Regulations, Parts 100–199, United States of America, Washington, DC (October 1, 2006).