Technical Evaluation Report

For the U.S. Department of Energy West Valley Draft Waste Incidental to Reprocessing Evaluation for the Vitrification Melter at the West Valley Demonstration Project

Final Report

U.S. Nuclear Regulatory Commission
Office of Federal and State Materials
and Environmental Management Programs
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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

EXECUTIVE SUMMARY

In March 2011, the U.S. Department of Energy (DOE) submitted the "Draft Waste Incidental to Reprocessing Evaluation for the Vitrification Melter" to the U.S. Nuclear Regulatory Commission (NRC) for consultative review. The DOE's evaluation assesses whether the Vitrification Melter at the West Valley Demonstration Project (WVDP) meets the waste incidental to reprocessing (WIR) criteria of Section II.B (2) (a) DOE-Manual 435.1-1 (DOE-M 435.1-1), Radioactive Waste Management. Demonstration that the criteria in DOE-M 435.1-1 are met allows DOE to dispose of the used Vitrification Melter offsite as low-level waste (LLW). This Technical Evaluation Report (TER) presents information on the DOE's evaluation process, the applicable review criteria, and the NRC's review approach, as well as the NRC's analysis and conclusions with respect to whether there is technical sufficiency to demonstrate that the DOE's proposed approach can meet the criteria of DOE-M 435.1-1 for determining that waste is not high-level waste (HLW). In its statement of work, the DOE also requested that the NRC evaluate its quality assurance program against NRC's guidance found in NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations."

The NRC staff has performed a technical review to assess whether the draft evaluation is technically sufficient to demonstrate that the Melter meets the criteria in DOE-M 435.1-1 accompanying DOE-Order 435.1-1. The NRC has conducted this consultative review at the request of the DOE in accordance with Interagency Agreement DE-EM0000284.

Based on the information provided by DOE and its associated contractor, West Valley Environmental Services, LLC, in the draft evaluation dated March 8, 2011 and letter dated June 27, 2011 (RAI response), the NRC staff has concluded that the DOE's draft evaluation is technically sufficient to demonstrate that the Vitrification Melter meets the NRC-reviewed portions of the criteria in DOE-M 435.1-1 accompanying DOE-Order 435.1-1.

DOE plans to ship the packaged Vitrification Melter to a suitable offsite LLW disposal facility, either the Nevada National Security Site (formerly the Nevada Test Site) in Nevada or the Waste Control Specialists facility in Texas for disposal.

In addition to the NRC, the DOE has solicited review and comment on the draft WIR evaluation and associated documentation from State officials and members of the public. The DOE will make a final determination of whether the Vitrification Melter is or is not HLW after consideration of the NRC's comments and any State and public comments on this draft evaluation.

Section II.B (2)(a) of DOE Manual 435.1-1 states the following:

II. B. Waste Incidental to Reprocessing.

Waste resulting from reprocessing spent nuclear fuel that is determined to be incidental to reprocessing is not high-level waste, and shall be managed under DOE's regulatory authority in accordance with the requirements for transuranic waste or low-level waste, as appropriate. When determining whether spent nuclear fuel reprocessing plant wastes shall be managed as another waste type or as high-level waste, either the citation or evaluation processes described below shall be used:

(1) Citation. Waste incidental to reprocessing by citation includes spent nuclear fuel reprocessing plant wastes that meet the description included in the Notice of Proposed Rulemaking (34 FR 8712) for proposed Appendix D, 10 CFR Part 50, Paragraphs 6 and 7.

These radioactive wastes are the result of reprocessing plant operations, such as, but not limited to: contaminated job wastes including laboratory items such as clothing, tools, and equipment.

(2) Evaluation. Determinations that any waste is incidental to reprocessing by the evaluation process shall be developed under good record-keeping practices, with an adequate quality assurance process, and shall be documented to support the determinations. Such wastes may include, but are not limited to spent nuclear fuel reprocessing plant wastes that:

(a) Will be managed as low-level waste and meet the following criteria:

1. *Have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical; and*

2. *Will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, Performance Objectives; and*

3. *Are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of [*DOE-M 435.1-1*], provided the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in §61.55, Waste Classification; or will meet alternative requirements for waste classification and characterization as DOE may authorize.*

Per the specific requests of the Interagency Agreement, the NRC's review was focused on assessing whether the methodology the DOE employed contained sound technical assumptions, analyses, projections, and conclusions. The NRC employed the relevant review procedures in Chapter 3, 6, 7, and 8 of NUREG-1854 (NRC, 2007), to complete its review. The NRC's review focused on the following general topics, as they relate to the criteria in DOE-M 435.1-1:

- Waste characterization,
- Waste form stability,
- Waste classification,
- Removal of radionuclides to the maximum extent technically and economically practical,
- Operational radiation protection, and
- Applicable quality assurance program elements.

Quality Assurance elements are included in the review because they relate to the inventory characterization which is important for demonstrating removal to the maximum extent practical as well as the waste classification evaluation. The Quality Assurance elements section of this review assesses the dose-to-curie measurements and the sampling performed on the glass from the evacuated canister.

The DOE plans to dispose of the Melter off-site, at a LLW disposal site such as the Nevada National Security Site or Waste Control Specialists (WCS), which are not licensed or otherwise regulated by the NRC. If DOE decides to dispose of the Vitrification Melter at the WCS facility, the Vitrification Melter waste package would be disposed of as Class C LLW in the Federal Facility Waste Disposal Facility. As per the Interagency Agreement, the NRC's review does not address the long-term performance of the site ultimately selected for disposal of the Melter. Therefore, the NRC's review will not address the long-term performance or long-term stability of this disposal site, but will focus only on the general topics listed above.

ABBREVIATIONS/ACRONYMS

AEC	Atomic Energy Commission
BTP	Branch Technical Position on Concentration Averaging
CFMT	Concentrator Feed Makeup Tank
CFR	Code of Federal Regulations
CNWRA	Center for Nuclear Waste Regulatory Analysis
DOE	U.S. Department of Energy
DOE-EM	U.S. Department of Energy – Environmental Management program
GTCC	Greater than Class C
HLW	High-Level Waste
IA	Interagency Agreement
LLW	Low-Level Waste
MFHT	Melter Feed Hold Tank
NNSS	Nevada National Security Site Area 5 Radioactive Waste Management Site
NFS	Nuclear Fuel Services
NRC	U.S. Nuclear Regulatory Commission
NYSERDA	New York State Energy Research And Development Authority
PA	Performance Assessment
PUREX	Plutonium - Uranium Extraction
RAI	Request for Additional Information
SOW	Statement of Work
TER	Technical Evaluation Report
TRU	Transuranic Waste
WCS	Waste Control Specialist
WIR	Waste Incidental to Reprocessing
WMA	Waste Management Area
WNYNSC	Western New York Nuclear Service Center
WVDP	West Valley Demonstration Project
WVNS	West Valley Nuclear Services Company

1.0 INTRODUCTION

1.1 Background

The WVDP is located in western New York State, about 50 km (30 miles) south of Buffalo, New York. The WVDP facilities occupy a security-fenced area of about 0.676 km^2 (167 acres) within the 13.51-km^2 (3,338-acre) Western New York Nuclear Service Center (WNYNSC) located primarily in the town of Ashford in northern Cattaraugus County.

To date, the West Valley site in West Valley, New York is the first and only commercial reprocessing plant to operate in the United States. From 1966 to 1972, Nuclear Fuel Services (NFS) reprocessed 640 metric tons of spent fuel under an Atomic Energy Commission (AEC) license. Approximately four years after shutting down, NFS returned control of the facilities to the site owner, New York State Energy Research and Development Authority (NYSERDA). Operations at the facility resulted in approximately 2.3 million liters (600,000 gallons) of liquid high-level waste (HLW) which was stored below ground in carbon-steel tanks, as well as other radioactive wastes, and residual radioactive contamination.

Figure 1-1 shows the perimeter of the WV site, and the relationship of the underground tanks to the Vitrification Facility, where the Melter was housed. Tank 8D-2 stored approximately 2.12 million liters (560,000 gallons) of waste consisting of neutralized PUREX wastes, which were composed of a bottom sludge layer containing insoluble hydroxides and other salts that precipitated out of solution, covered by a layer of alkaline liquid (supernatant) rich in sodium nitrate. Tank 8D-2 wastes were neutralized by adding sodium hydroxide to the nitric acid-based stream during the reprocessing of uranium fuel using the PUREX process. Neutralizing the initially acidic HLW prior to transfer caused most of the fission product elements (the major exception was cesium) to precipitate out and form sludge at the bottom of Tank 8D-2. In addition, neutralizing the wastes reduced the possibility of corrosion of the carbon steel tank. Therefore, the HLW was not homogeneous but was comprised of supernatant (liquid) and sludge (solids). Tank 8D-4 stored approximately 46,000 L (12,000 gallons) of acidic high-level radioactive liquid waste produced in reprocessing thorium-enriched uranium fuel using the THOREX process; this waste was stored without being neutralized.

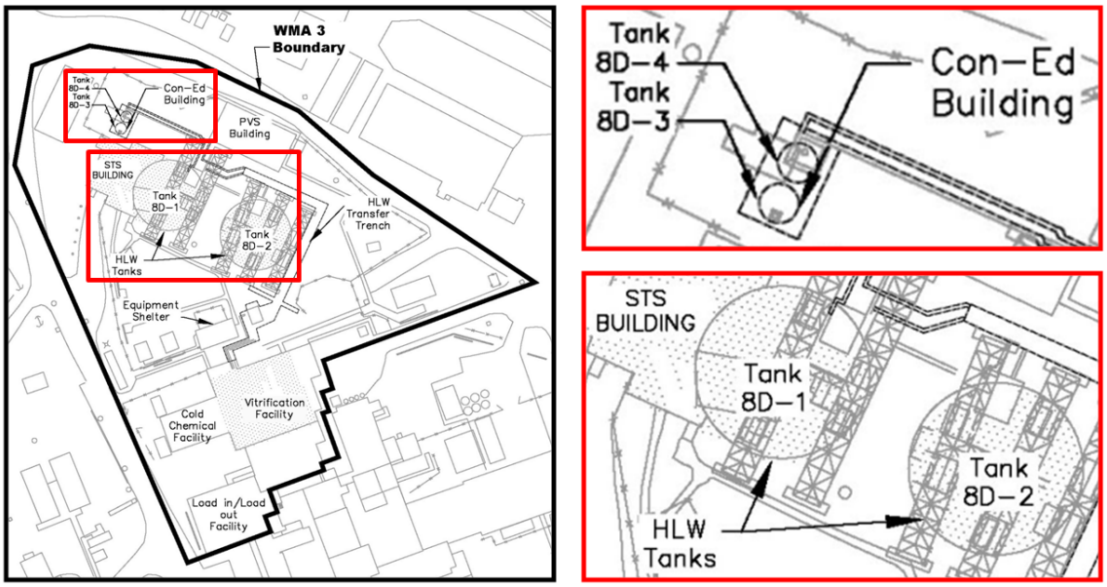

Figure 1-1: HLW Storage and Vitrification Storage Facility

In 1980, the West Valley Demonstration Project Act (WVDP Act) was passed. The WVDP Act made DOE responsible for solidifying the liquid HLW stored in underground tanks, disposing of the waste created by solidification, and decontaminating and decommissioning the facilities used during the process. Under the WVDP Act, the DOE entered into a Cooperative Agreement with NYSERDA that established the framework for cooperative implementation of the WVDP Act. Under the agreement, the DOE was provided exclusive use and possession of a portion of the West Valley site. A supplement to the Cooperative Agreement between the two agencies set forth special provisions for the preparation of a joint Environmental Impact Statement (EIS).

In 1981, the DOE and the NRC entered into a Memorandum of Understanding that established specific agency responsibilities and arrangements for informal review and consultation by the NRC. Because NYSERDA holds the license and title to the West Valley site, the NRC put the technical specifications of the license in abeyance to allow the DOE to carry out the responsibilities of the WVDP Act.

In 1982, West Valley Nuclear Services (WVNS), a Westinghouse subsidiary, was chosen by the DOE to be the management and operating contractor. WVNS commenced operations at the WVDP on February 28, 1982. Shortly thereafter, the DOE selected the vitrification process as the preferred method for solidifying the HLW into glass.

In 1988, to prepare for vitrification, DOE constructed the integrated radioactive waste treatment system to process the liquid supernatant from the underground HLW tanks by removing most of the radioactivity in the supernatant, concentrating the liquid, and blending it with cement. This process separated the low activity waste stream from the high activity waste stream that was vitrified into borosilicate glass. The process was completed in 1995.

Waste pretreatment and vitrification took place from 1996 to 2002. The salt/sludge separation process used to treat the waste prior to vitrification involved the following: separating the supernatant from the sludge in Tank 8D-2; removing the radioactive Cs-137 from liquids in both tanks by adsorbing it onto zeolite; combining the cesium-loaded zeolite with the sludge from

Tank 8D-2. Once the waste had been pretreated, the zeolite and sludge mixture was ready to be vitrified into the borosilicate glass waste form.

Vitrification of the HLW took place between 1996 and 2002, producing a total of 275 stainless-steel canisters of hardened radioactive glass. The canisters were each 305 cm (10 ft) tall and contained more than 451 million cesium/strontium gigabecquerels (12.2 million curies) in total. The glass melter was shut down in September 2002.

Figure 1-2 shows the vitrification process flow. Waste was pumped from Tank 8D-2 to the Concentrator Feed Makeup Tank (CFMT), where glass formers were added to the system. Then, waste was transferred to the Melter Feed Hold Tank (MFHT) before it was fed into the Melter through the Feed Delivery System. These other components are relevant to this review in that the some of the activities to clean the Melter involved processing increasingly dilute material through these systems and through the Melter.

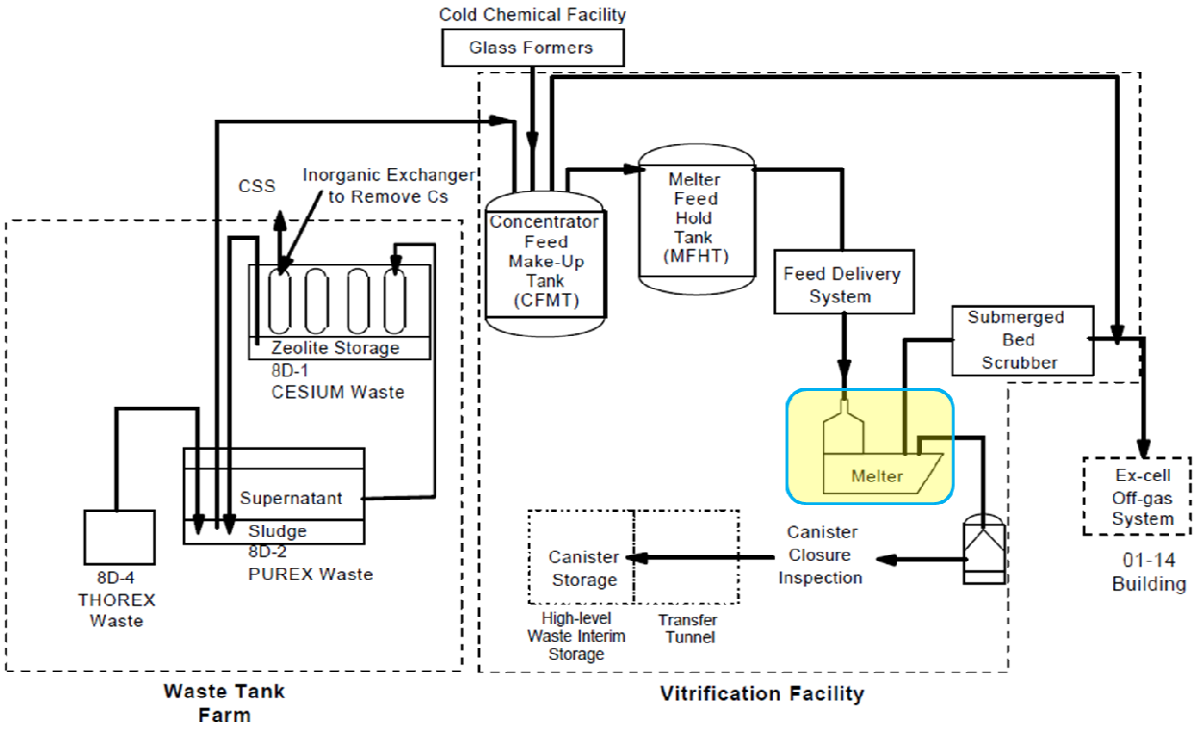

Figure 1-2: Vitrification Process Flow

The Melter cavity consisted of an upper rectangular tank above an inverted rectangular pyramid. Inside the Melter, the waste was initially heated by radiant heat transfer from electric heaters. When the temperature of the glass reaches 800°C (1,470°F) the electrodes positioned at 40.6 cm (16 inches) from the cavity floor were energized to heat the glass to the operating temperature of 1,050°C to 1,150°C (1,900°F to 2,100°F). This evaporated water in the slurry and the remaining solids calcified. The calcined waste and glass formers melted into a pool of homogenous glass that was mixed with natural convection. At regular intervals, glass was poured from the Melter discharge port into HLW canisters held beneath the Melter, which were sealed after the glass cooled. Figure 1-3 illustrates the Melter and canister turntable.

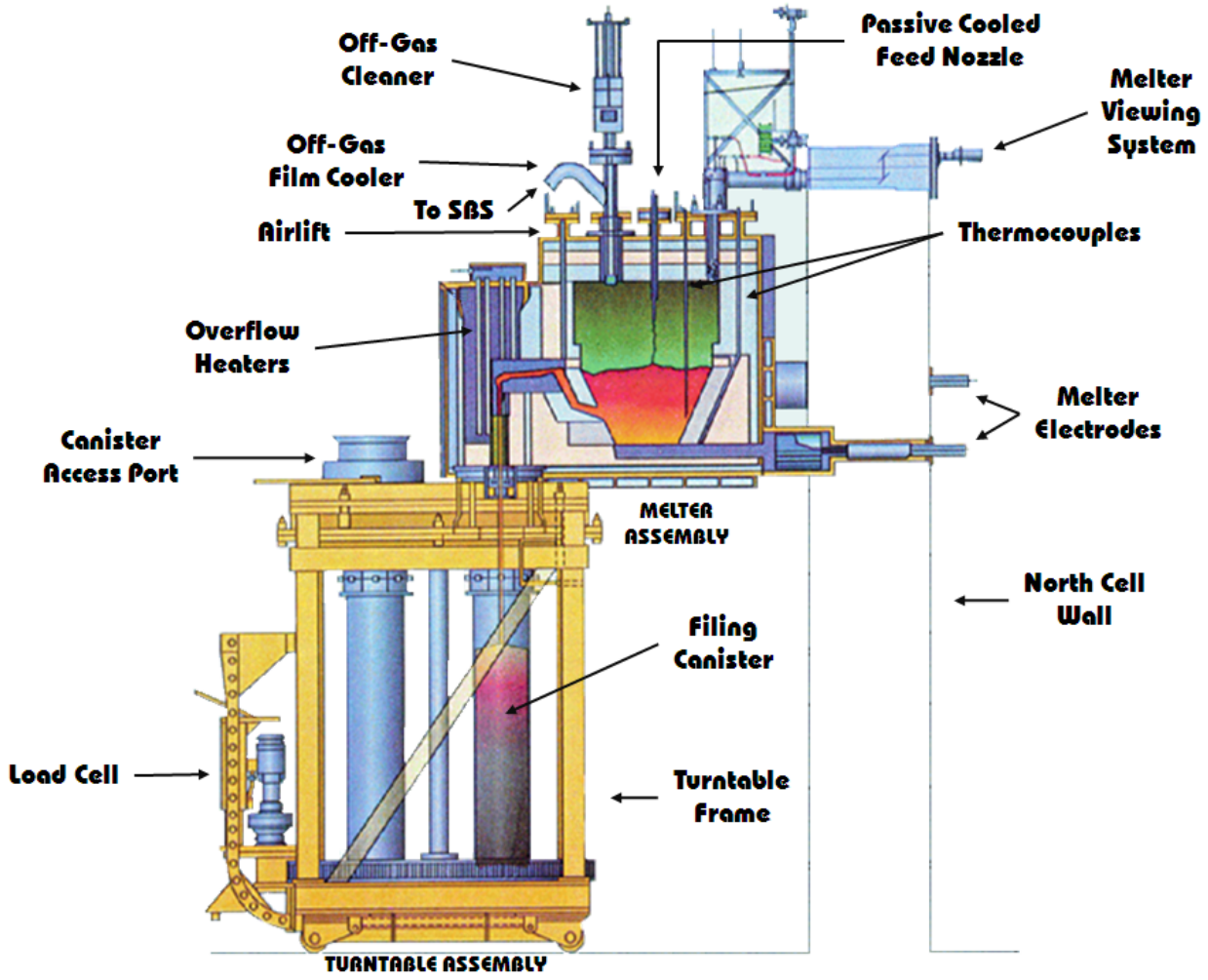

Figure 1-3: Melter and Canister Turntable

An important operating problem occurred near the end of operations when the primary glass exit port plugged. After this event, the secondary port and corresponding pour chamber were utilized. After vitrification was completed in 2002, decontamination fluids were added to the CFMT and MFHT which was then processed in the Vitrification Melter; the Melter cavity emptied an estimated 20.3 cm (8 in) into evacuated canisters, and deactivated. Also, the outside of the Melter was decontaminated.

Since DOE completed vitrification of the treated HLW in 2002, efforts have focused on decontaminating and deactivating facilities and shipping LLW offsite, including characterization, and disposal of the Vitrification Melter. The NRC staff routinely visits the WVDP to facilitate review and consultation as required under the WVDP Act. Two such visits in 2004 focused on vitrification equipment. During these visits, NRC staff members reviewed information on the characterization, classification, and packaging for the Vitrification Melter and concluded that all applicable regulatory requirements had been met (NRC, 2004).

1.2 Waste Incidental to Reprocessing Evaluation

The DOE is evaluating whether the Vitrification Melter at the West Valley Demonstration Project meets the waste incidental to reprocessing criteria of DOE-M 435.1-1, Radioactive Waste Management in order to dispose of the used Vitrification Melter offsite as LLW. DOE plans to ship the packaged Vitrification Melter to a suitable offsite LLW disposal facility, either the Nevada National Security Site (NNSS, formerly the Nevada Test Site) in Nevada or the Waste Control Specialists (WCS) facility in Texas for disposal.

Consistent with the Department's policy, the DOE has reached out to the NRC, as well as state officials and members of the public, to solicit review and comment on a draft WIR evaluation. The DOE will make a final determination of whether the Vitrification Melter is or is not high level waste after considering comments provided by the NRC, the public and any state.

Section II.B (2)(a) of DOE-M 435.1-1 states the following:

II. B. Waste Incidental to Reprocessing.

Waste resulting from reprocessing spent nuclear fuel that is determined to be incidental to reprocessing is not high-level waste, and shall be managed under DOE's regulatory authority in accordance with the requirements for transuranic waste or low-level waste, as appropriate. When determining whether spent nuclear fuel reprocessing plant wastes shall be managed as another waste type or as high-level waste, either the citation or evaluation processes described below shall be used:

(1) Citation. Waste incidental to reprocessing by citation includes spent nuclear fuel reprocessing plant wastes that meet the description included in the Notice of Proposed Rulemaking (34 FR 8712) for proposed Appendix D, 10 CFR Part 50, Paragraphs 6 and 7. These radioactive wastes are the result of reprocessing plant operations, such as, but not limited to: contaminated job wastes including laboratory items such as clothing, tools, and equipment.

(2) Evaluation. Determinations that any waste is incidental to reprocessing by the evaluation process shall be developed under good record-keeping practices, with an adequate quality assurance process, and shall be documented to support the determinations. Such wastes may include, but are not limited to, spent nuclear fuel reprocessing plant wastes that:

(a) Will be managed as low-level waste and meet the following criteria:

1. *Have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical; and*
2. *Will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, Performance Objectives; and*
3. *Are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of [*DOE-M 435.1-1*], provided the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in §61.55,*

Waste Classification; or will meet alternative requirements for waste classification and characterization as DOE may authorize.

1.3 The NRC Review Approach

NRC's review of the draft WIR evaluation was performed in accordance with the specific requests as described in the Interagency Agreement (IA), DE-EM0000284. To avoid confusion, this IA was very specific in its requests in an attempt to avoid any overlap with analogous programs either between NRC and DOE-EM, or other NRC/DOE interactions at the West Valley site.

The NRC has reviewed the draft evaluation for the Melter to provide a technical opinion on whether the draft evaluation is technically sufficient to demonstrate that the WVDP Vitrification Melter meets the criteria in DOE-M 435.1-1, which accompanies DOE-O 435.1-1. DOE's request explicitly stated that the NRC not review matters covered by other sections of DOE-M 435.1-1. Potentially applicable sections that are not accounted for in the NRC's review include: (i) the applicable LLW requirements in Chapter IV of the DOE-M 435.1-1, (ii) the sufficiency of the waste acceptance criteria for the DOE LLW disposal facility, and (iii) the sufficiency of the PA for whichever disposal facility might be receiving the waste.

The NRC's review of the draft WIR evaluation focused on assessing whether the methodology the DOE employed was based on sound technical assumptions, analyses, and projections, and resulted in acceptable conclusions. The NRC employed the relevant review procedures in Chapter 3, 6, 7, and 8 of NUREG-1854 (NRC, 2007), to complete its review of the general topics identified in Table 1-1 below. The NRC reviewed the general topics identified in Table 1-1 below; as they relate to the criteria in DOE-M 435.1-1 (followed by applicable chapter of NUREG-1854, and applicable DOE-M 435.1-1 criteria):

Table 1-1: Applicable NRC Review Guidance and DOE-M 435.1-1 Criteria

General Review Topic	NUREG-1854 Chapter	435.1 Criterion
Waste characterization	Chapter 3: Radionuclide Removal and Concentration Limits	1
Waste form stability	Chapter 7: Site Stability, Waste Stability, and Facility Stability	2 and 3
Waste classification	Chapter 3: Radionuclide Removal and Concentration Limits	3
Removal of radionuclides to the maximum extent technically and economically practical	Chapter 3: Radionuclide Removal and Concentration Limits	3
Operational radiation protection	Chapter 6: Protection of Individuals During Operations	2
Applicable quality assurance program elements	Chapter 8: Quality Assurance Program	-

The DOE plans to dispose of the Melter off-site, at a LLW disposal site such as the NNSS or WCS, neither of which are licensed or otherwise regulated by the NRC (but are instead licensed and regulated by DOE — in the case of NNSS — or by an NRC Agreement State, the State of Texas in the case of WCS). As per the IA, the NRC's review does not address the long-term

performance of the disposal site or the sufficiency of the waste acceptance criteria for the potential disposal facilities being considered for disposal of the Melter. Therefore, the NRC's review will not address the long-term performance or long-term stability of this disposal site, but will focus only on the general topics listed above.

2.0 THE WASTE HAS BEEN PROCESSED TO REMOVE KEY RADIONUCLIDES TO THE MAXIMUM EXTENT THAT IS TECHNICALLY AND ECONOMICALLY PRACTICAL

This section of the NRC staff's technical review covers Section II.B(2)(a)(1) – that the waste has been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical.

2.1 The DOE's Characterization of Waste Inventory

The DOE estimates the amount of residual material in the Melter cavity after decontamination to be 425 kg, consisting of 99 kg of glass in the plugged discharge port, approximately 20.3 cm (8 in) or 300 kg of glass in the bottom of the Melter cavity, and approximately 26 kg of glass as a thin layer 0.32 cm (0.125 in) on the Melter cavity surfaces or to a depth of 1.3 cm (0.5 in) in refractory fissures. The inventory of the residual glass, as estimated in 2004, is described in two reports Lachapelle (2003) and WMG (2004a).

The DOE's characterization is based on measured gamma dose rates using a Ludlum Model 1337 Geiger-Mueller detector probe (WMG, 2004a) and analytical results from two samples of the residual glass. The measured gamma dose rates were attributed solely to Cs-137 in order to estimate Cs-137 activity. The QAD-GCCP-A(2) code was used for point kernel modeling of the complex geometry in the Melter cavity, while Microshield® was used to model the simpler geometry of the plugged port.

The DOE used results from the laboratory samples to determine scaling factors that relate other radionuclides to Cs-137 activity. RADman™ software was applied to calculate the activities of other radionuclides using the scaling factors and calculated Cs-137 activity. The glass samples used to characterize the residual material were taken from the evacuation canisters filled with the material that remained in the Melter after flushing with decontamination solutions. DOE considers these samples to adequately represent the heterogeneity of the residual waste within the Melter due to the mixing of the HLW in the Melter Feed Hold Tank, and subsequent homogenization of the waste as it reached a molten state and was mixed by natural convection in the Melter.

The results of the glass sample analyses are documented in WMG (2004a). The three results for each of the two samples are averaged to find the scaling factor. Section 2.5.3 of the draft WIR Evaluation does not include a discussion of this variability, and whether it was attributable to waste heterogeneity, or analytical uncertainty in measurements. The DOE states that applying the arithmetic average of the samples to estimate activity is conservative, despite this variability not being considered.

From application of the gamma dose rates and the scaling factors, the DOE estimated that the total activity remaining in the Melter as of October 1, 2004 was 175.8 TBq (4,750 Ci), 90% of which was Cs-137. This estimate is referred to as the dose-to-curie estimate, because it is based on measured dose rates that were directly converted to activity for Cs-137; the activities of the other radionuclides were inferred from Cs-137 using scaling factors. Approximately 192.4 GBq (5.2 Ci) [as of October 2004] is assumed to be on the external surface of the Melter. The inventory (also listed in Table 2-2 of the draft WIR Evaluation) is replicated below in Table 2-1.

Table 2-1: Vitrification Melter Total Activity Estimates (October 2004)

Nuclide	Activity (Ci)**	Nuclide	Activity (Ci)**
C-14	2.12E-02	U-238	2.25E-03
K-40	8.19E-02	Np-237	6.20E-03
Mn-54	8.57E-02	Pu-238	6.84E-01
Co-60	8.33E-02	Pu-239	1.59E-01
Sr-90	2.47E+02	Pu-240 *	5.93E-02
Zr-95	1.65E+00	Pu-241	3.12E+00
Tc-99	1.11E-02	Pu-242 *	1.12E-05
Cs-137	4.31E+03	Am-241	3.00E+00
Eu-154	1.21E+00	Am-242m *	9.16E-05
Th-228	4.09E-02	Am-243	3.50E-02
Th-230	3.65E-04	Cm-242	7.33E-02
Th-232	4.01E-04	Cm-243	1.68E-02
U-232	5.01E-04	Cm-244 *	1.55E-02
U-234	9.81E-03	Cm-245 *	1.77E-03
U-235	3.76E-04		

*Radionuclides are not expected to be present in significant quantities, but have been listed to be consistent with the list of Key Radionuclides. Concentrations for these radionuclides are estimated using an alternative method to the dose-to-curie method that employs glass sample inventory concentration measurements and approximate mass in melter.
**Conversion to Bq from curies, multiply by 0.37 TBq/Ci.

2.2 NRC Evaluation of Waste Inventory

In this part of the review, the NRC staff evaluated the physical and chemical form of the waste, previous inventory estimates, data quality objectives, homogeneity, volume and mass estimates, and uncertainties.

The NRC staff has reviewed information about waste generation and treatment activities and finds that the predicted physical and chemical forms of radionuclides, as described by DOE, are consistent with the properties of contributing waste streams and treatment processes.

The NRC staff has evaluated previous inventory estimates for the glass canisters in an earlier report (Eisenstatt,1986), and compared those with the estimates in the 2004 characterization report (WMG, 2004a) to identify any differences between historical and more recent estimates. During its review, the NRC staff noted the differences in the radionuclides included in these two inventories. In the response to NRC RAIs, the DOE explained that these differences were due to different characterization objectives, and has revised the inventory table and waste classification table to include Pu-242, Am-242m, Cm-245, and Cm-246 for consistency purposes (DOE, 2011b).

The NRC staff evaluated the sampling analysis and data quality objectives to verify that relevant data quality objectives were met. While explicit data quality objectives were not stated in WMG (2004a), approaches were taken to verify calculations and accuracy of measurements. The DOE states that data were validated in accordance with the requirements of the Characterization Management Plan for the Facility Characterization Project (Michalczak, 2004), which includes data quality objectives. Furthermore, the 2004 characterization was reviewed by the site contractor, West Valley Nuclear Services Company, and DOE incorporated recommendations resulting from this review. The NRC staff concluded that the approaches

taken for data verification as outlined in the Characterization Management Plan for the Facility Characterization Project were appropriate.

In regards to the homogeneity of the waste inventory, the NRC staff notes that one of the references provided suggests the potential for hardened glass in the Melter to build up over time in various small spaces (e.g., joints around nozzles, pipes used for airlifting, discharge port) (Brooks, 1993). Depending on the variability in the waste streams being fed into the Melter throughout its operation, there could be potential for pockets of hardened glass to contribute variability to the glass content remaining in the Melter. For example, residual, hardened glass deposited in the Melter might vary over space and time (e.g., glass that hardens in refractory material joints over time). Similarly, residual contamination that accumulated on the ceiling of the Melter due to vaporization of radioactivity would be expected to have a different radionuclide mix compared to the contamination in the Melter and the contamination deposited in piping and components that feed into the Melter.

The DOE provided additional information about the Melter processes, including data from samples taken from the concentrator feed makeup tank and gamma dose rates on the outside of the HLW canisters over the lifetime of the Melter, to support the assumption of homogeneity and representativeness of the samples. Specifically, DOE showed that the ratio of Cs-137 to Sr-90 in the CFMT is fairly consistent in Batches 10 through 67, as are the gamma dose rates from the filled canisters. NRC staff notes that DOE did not provide a basis for why the ratio of Cs-137 to Sr-90 increases for Batches after 67, nor does DOE discuss the consistency of the ratio of Cs-137 to other radionuclides besides Sr-90 in the CFMT samples from Batches 10 through 67. In reviewing the CFMT data for Batches 70 through 77 (the only Batches for which other nuclides were provided), NRC staff found that the ratio of Am-241 to Cs-137 ranged an order of magnitude from 2.8E-03 to 2.1E-04. Also, one would expect the gamma dose rates to be fairly consistent regardless of whether or not the waste was homogeneous since Cs-137 is the primary gamma emitter, contributing over 90% of the dose rate. However, the staff agrees that the processes within the tank (e.g., convective mixing in the Melter, mixing paddles, and agitation in the MFHT) would contribute to a homogeneous mixture (DOE, 2011b).

The DOE also supported the representativeness of the glass samples for the plugged discharge port by pointing to the similarity in the scaling factors for Sr-90, Np-237, Pu-238, and Am-241 of the glass samples to the CFMT batch samples that were being processed at the time the port was plugged (Batches 74 and 75). This data is reproduced in Table 2-2.

Table 2-2: Radionuclide Scaling Factors (Ratios to Cs-137) (DOE, 2011b)

Radionuclide	Glass Sample Data	Batch 74 Data	Batch 75 Data
Sr-90	5.73E-02	4.00E-02	7.47E-02
Np-237	1.41E-06	3.25E-07	6.13E-07
Pu-238	1.57E-04	5.48E-05	1.09E-04
Am-241	6.84E-04	2.19E-04	3.28E-04

Furthermore, the DOE assumes that most of the glass that accumulated in the fissures in the refractory of the Melter is from the testing performed with nonradioactive slurry prior to the start of HLW vitrification. Finally, the DOE acknowledged that while the radionuclide contamination on the upper portions of the walls and roof may be somewhat different in composition than that on the floor and sides up to 66 cm (26 in), the amounts on these surfaces are negligible in comparison to the rest of the material. Through the supporting additional information provided,

the NRC has concluded that the samples adequately represent the potential variability of the composition of the waste remaining in the Melter in various areas.

The NRC staff has reviewed the references cited as providing the technical basis for estimating the residual volume of waste. To summarize, the DOE estimates that a 0.318 cm (0.125 in) thick layer exists on the cavity walls (Brooks, 1993), and 20.3 cm (8 in) or 300 kg of solidified glass resides in the bottom of the cavity, and that glass exists in fissures to a depth of 1.3 cm (0.5 in) [Petkus, 2002b]. The DOE includes an alternative calculation of the depth of the remaining glass in the cavity of the Melter and additional information from the January 2003 visual inspection to support its assumption. By assuming a 66 cm (26 in) starting level and subtracting the volume removed by the evacuated canisters, the DOE calculates a remaining heel of 16.5 cm (6.5 in). The NRC staff finds DOE's independent assessment to be technically acceptable and therefore agrees that a 20.3 cm (8 in) assumption is conservative.

To provide additional confidence in its inventory estimate, the DOE derives the total residual activity from the mass and compares that to the activity developed using the dose-to-curie method. Deriving the activity from the assumed mass, density, and concentrations from the glass samples yields a total activity of 82.88 TBq (2,240 Ci) in comparison to 175.8 TBq (4,750 Ci), which DOE claims is evidence of the conservatism in the dose-to-curie method of estimation. Given that the DOE's alternative calculation provided additional support that the volume was not underestimated, the NRC agrees that the DOE's assumptions regarding the activity remaining in the Melter are conservative.

The impact of uncertainties due to heterogeneity, sample variability, analytical methods, and volume estimates should be evaluated as it could affect waste classification. The DOE provided data supporting its assumptions regarding sample homogeneity, but did not quantify uncertainty caused by the variability of the concentration in the waste remaining in the tank (i.e., a single concentration for each radionuclide was assumed for all waste remaining in the Melter). DOE did not put an uncertainty bound on the volume estimate, but did provide an alternative calculation that supported their assumption that the original estimate of 20.3 cm (8 in) was conservative. As far as sample uncertainty, the NRC staff reviewed the uncertainties reported in the lab analysis results for the glass samples (WMG, 2004a) and other historical references. The 1987 report (Eisenstatt, 1986) states the following about uncertainty, "It is expected that the activities for the fission products, uranium, and plutonium may vary about five percent from these values. Thorium may vary about 20%. The remaining actinides may vary about 50%." The uncertainty in the glass samples was smaller than the uncertainty predicted in the Eisenstatt report; the uncertainties for Pu-238 and Am-241, which are important radionuclides for the waste classification, were 6% and 5% respectively. However, the uncertainties in lab samples are not included in the final scaling factors or inventories. The DOE states that the sample uncertainty is bounded by the conservatism in the dose-to-curie approach. Also, a basis for using the average of the measurements is not provided, but the DOE states that averaging of the scaling factors is common practice as evidenced by its incorporation into RADman™ software, which is endorsed by the NRC. The NRC staff concludes that if, after all the uncertainties are taken into account waste classification could potentially be impacted (the sum of fractions could exceed one as discussed in Section 4.2). However, considering the conservatism inherent in the dose-to-curie method in comparison to the alternative method for calculating inventory that was provided in the response to RAIs and is described in Section 2.1, the inventory and waste classification are most likely appropriately bounded.

2.3 DOE Selection of Key Radionuclides

Section II.B of DOE-Guide 435.1-1, states the following regarding key radionuclide selection:

> *"... it is generally understood that [the term] key radionuclides applies to those radionuclides that are controlled by concentration limits in §61.55. Specifically these are: long-lived radionuclides, C-14, Ni-59, Nb-94, Tc-99, I-129, Pu-241, Cm-242, and alpha emitting transuranic nuclides with half-lives greater than five years and; short-lived radionuclides, H-3, Co-60, Ni-63, Sr-90, and Cs-137. In addition, key radionuclides are those that are important to satisfying the performance objectives of 10 CFR Part 61, Subpart C [for near-surface radioactive waste disposal facilities]."*

In following this guidance, DOE identifies key radionuclides for the draft WIR evaluation based on the radionuclides listed in §61.55 and the performance assessments (PA) for the disposal facilities.

The key radionuclides, (also listed in Table 4-3 of the draft WIR Evaluation) are replicated below in Table 2-3.

Table 2-3: Key Radionuclides

Radionuclide	§61.55 Long-lived Radionuclides	§61.55 Short-lived Radionuclides	Radionuclides Important to PA
H-3		X	
C-14	X		X
Co-60		X	
Ni-59	X		
Ni-63		X	
Sr-90		X	
Nb-94	X		
Tc-99	X		X
I-129	X		X
Cs-137		X	
Th-229			X
U-233			X
U-234			X
U-238			X
Np-237(3)	X		
Pu-238(3)	X		
Pu-239(3)	X		
Pu-240(3)	X		
Pu-241	X		
Pu-242(3)	X		
Am-241(3)	X		
Am-243	X		
Cm-242	X		
Cm-243(3)	X		
Cm-244(3)	X		

2.4 NRC Evaluation of Key Radionuclides

While the DOE followed the guidance put forth in DOE-M 435.1-1 in selecting key radionuclides, the guidance in NUREG-1854 (which is slightly different than DOE-M 435.1-1) encourages key radionuclides to be defined in manner which is risk-informed by a particular waste inventory. NUREG-1854, Section 3.2, states that the list of key radionuclides is expected to be specific to a particular waste determination, and that the DOE may start with the radionuclide inventories and eliminate radionuclides from the list of potentially key radionuclides based on screening criteria. It is also the NRC's position that the list of key radionuclides should be those that contribute most to risk, including risk to the inadvertent intruder.

One way to identify key radionuclides would be to use the actual Melter inventory as input to the performance assessments for each disposal site, and identify those that contribute most to dose. Following this approach one would expect that the list of key radionuclides would include the most risk-significant radionuclides in the inventory.

As part of its review, the NRC compared the differences between three separate lists of radionuclides used by DOE in the WIR evaluation:

(1) Key Radionuclide List, or the radionuclides that are important to risk for each disposal site when assessing the disposal site performance,
(2) Inventory List, or the radionuclides that are expected to remain in the Melter, and
(3) Waste Classification List, or the radionuclides that were considered in calculating the sum of fractions for purposes of waste classification.

This comparison is shown in Table 2-4 and was completed to identify any inconsistencies in the radionuclides included within each list since they should all three be identical unless certain radionuclides in the inventory list are deemed unimportant for either waste classification purposes or performance assessment purposes (e.g., short-live radionuclides).

In response to a request for additional information the DOE explained the differences between the Melter Inventory List and the Key Radionuclide List. Specifically, Am-242m, Cm-245, and Cm-246 were omitted from the Key Radionuclide List even though they are expected to remain in the Melter because of their low activity levels (denoted with "Omitted (Low Activity)" in Table 2-4). Also, to improve consistency, the DOE included additional radionuclides to the Inventory List (denoted with "Added to Rev 1" in Table 2-4), even though these radionuclides are expected to be present in insignificant quantities.

DOE also explained the differences between the Melter Inventory List and the Waste Classification List. DOE stated that several non-transuranic radionuclides that were in the inventory would not have an impact on the waste classification so these were not included (indicated by the "No Effect" in Table 2-4). The NRC staff reviewed those radionuclides that were listed in the inventory but not listed for the purposes of waste classification (H-3, Ni-59, Nb-94, Co-60, Th-229, and uranium) and concluded that omission of these radionuclides from Table 6-1 in the draft WIR evaluation is in accordance with §61.55. Specifically, although Co-60 is listed in Table 2 of §61.55, the concentration in the Melter is less than the concentration in Table 2, so DOE did not include its contribution for the purposes of waste classification. Also, Th-229 and uranium are not listed in the §61.55 Tables. To improve consistency, the DOE included Pu-242 and Am-242m in the Waste Classification List (denoted with "Added to Rev 1").

The NRC did not receive a request to review the PA for either disposal site, so was unable to independently evaluate the importance of the radionuclides that were listed as part of the inventory but were not included as key radionuclides. However, because the technologies that the DOE applied or the alternative technologies considered for removal of radionuclides did not target specific key radionuclides and instead removed bulk volume from the Melter, the DOE's cleaning approach effectively removed all radionuclides, as opposed to only the key radionuclides. Therefore, the NRC staff does not have to make a determination on whether or not the key radionuclide list is appropriate because the staff does not rely on this list for coming to a conclusion about removal to the maximum extent practical.

In response to RAIs (i.e., RAI KR-2) and based on DOE's review of the radionuclides that are known to be important to the PAs for the disposal sites, DOE expressed confidence that no risk-significant radionuclides present in the Melter inventory are omitted from the list of key radionuclides. Given that the specific inventory of the West Valley Melter was considered in a special assessment for the Nevada disposal site that included long-term public exposure and intruder dose, the radionuclides that are present in the inventory that are risk-significant for the Nevada site most likely were appropriately captured in this special assessment (DOE, 2010b).

For the WCS disposal site, DOE relied on the PA that was submitted along with the WCS license application, which evaluated a wide range of waste streams from multiple generators, but did not specifically consider the Melter inventory (WCS, 2007). Among the radionuclides that are listed in the Melter inventory, but are not key radionuclides, the NRC staff has concluded that these are either omitted due to very short half-lives or have been evaluated as part of the PA for the WCS facility. The NRC staff notes that information on the radionuclides that were most important to the intruder scenario for the Waste Control Specialist LLW disposal facility was not available. This impacts the staff's ability to compare the radionuclides in the inventory to those that were important to the intruder scenario for WCS, and contributes to why the staff is relying on DOE's expressed confidence regarding the appropriateness of the Key Radionuclide List.

Table 2-4 compares the radionuclides that were included in the Key Radionuclide List, the Inventory List, and the Waste Classification List. The NRC's evaluation of waste classification is further discussed in Section 4.2 of this TER; the Waste Classification List is reproduced here for comparison to the other two lists.

Table 2-4: Comparison of Consideration of Radionuclides in Draft WIR Evaluation

Radionuclide	Key Radionuclide List	Melter Inventory List	Waste Classification List
H-3	X		
C-14	X	X	X
K-40		X	No Effect [(1)]
Mn-54		X	No Effect
Co-60	X	X	Conc. < §61.55, Tbl 2 Col. 1[(2)]
Ni-59	X		
Ni-63	X		X
Sr-90	X	X	X
Nb-94	X		
Zr-95		X	No Effect
Tc-99	X	X	X
I-129	X	Added to Rev 1[(4)]	X

Cs-137	X	X	X
Eu-154		X	No Effect
Th-228		X	No Effect
Th-229	X	Added to Rev 1	Not Listed [3]
Th-230		X	Not Listed
Th-232		X	Not Listed
U-232		X	Not Listed
U-233	X	X	Not Listed
U-234	X	X	Not Listed
U-235		X	Not Listed
U-238	X	X	Not Listed
Np-237	X	X	X
Pu-238	X	X	X
Pu-239	X	X	X
Pu-240	X	Added to Rev 1	X
Pu-241	X	X	X
Pu-242	X	Added to Rev 1	Added to Rev 1
Am-241	X	X	X
Am-242m	Omitted (Low Activity)[5]	Added to Rev 1	Added to Rev 1
Am-243	X	X	X
Cm-242	X	X	X
Cm-243	X	X	X
Cm-244	X	X	X
Cm-245	Omitted (Low Activity)	X	X
Cm-246	Omitted (Low Activity)		

(1) No Effect indicates that DOE performed an analysis confirming that these transuranic radionuclides would not have an impact on classifying waste.
(2) Not Listed indicates that the radionuclides are not listed in the §61.55 Tables.
(3) Conc. < §61.55, Tbl 2 Col. 1 indicates that Co-60 is listed in Table 2 of §61.55 but the concentration of this radionuclide in the Melter is less than the concentration in Table 2, so DOE did not include its contribution for the purposes of waste classification
(4) Added to Rev 1 indicates that DOE added these radionuclides the lists in response to RAIs.
(5) Omitted (Low Activity) indicates that even though these radionuclides may be in the Melter inventory, they were omitted from the key radionuclide list because they are present in low activity quantities.

2.5 Removal to the Maximum Extent Technical and Economically Practical

2.5.1 The DOE Selection of Waste Removal Technologies

Along with the design of the Vitrification Facility, the DOE developed a tailored method for removing the molten glass that was expected to be in the Melter at the time it was shutdown. This method involved the use of decontamination fluids and the use of evacuated canisters that would retrieve the waste using vacuum pressure. The DOE also considered a range of additional potential removal technologies documented in the *Decommissioning Handbook* (DOE, 1994). The DOE concluded that, while none of the technologies listed were *directly* applicable to the Melter, the category which was most relevant to the Melter was “Embedded Material and Some Oxide Surfaces.” Approaches listed under this category included mechanical means of removing radionuclides.

The DOE fully evaluated the following options for technical and economical practicality: (1) processing decontamination fluids consisting of nitric acid and demineralized water mixed with glass formers; (2) use of evacuated canisters; (3) mechanical removal through ball milling; and (4) dismantling. These options included those methods that reduced the volume of waste in the contaminated Melter (i.e., mechanical means). Decontamination fluids were not applied in a traditional way that would chemically remove key radionuclides because the nitric acid and demineralized water would immediately lose their identity in the molten glass pool. Therefore, the chemicals had no independent impact on the Vitrification Melter surfaces and would not selectively remove particular elements or radionuclides. Consequently, the processing of decontamination fluids that had been fed through the CFMT and MFHT along with glass formers was a method of diluting the waste which was in the Melter. This diluted material was removed mechanically using HLW canisters and two specially designed evacuation canisters.

The selection criteria for technology solution included technology maturity, usefulness, cost per unit activity removed, limitations, and net social costs. The DOE determined that processing decontamination fluids with a mixture of glass formers, and use of evacuated canisters were technical and economically practical, but that ball grinding and dismantlement were not practical. Specifically, the improvements necessary to the ball grinding process to make it effective for removing waste from the Melter would have been so extensive that the resulting technology would no longer be considered proven; this would make its use technically impractical. Dismantlement was considered economically impractical because the costs of dismantlement far outweighed the cost savings gained by shipping smaller containers.

The flushing process involved flushing the tanks and related equipment that were a part of the vitrification system, and slowly feeding the flushing material along with glass formers to the Melter while periodically airlifting molten glass into a HLW canister. Batch 75 was the last batch of “pure” HLW (i.e., before any decontamination solution had been added to the system), which is estimated to have filled canisters 262 through 268. Canisters 269 to 272 were filled by the first flush of decontamination fluids, whereas the second flush filled canisters 273 to 275, along with the two evacuated canisters. These removal technologies were not expected to remove 100% of the material from the Melter; the DOE expected that 7.6 to 15.2 cm (three to six inches) of glass would be left in the bottom of the Melter cavity after use of the evacuated canisters.

2.5.2 The NRC Evaluation of Waste Removal Technologies

The NRC staff has reviewed DOE’s basis for radionuclide removal technology and has concluded that DOE evaluated a reasonable range of technologies that included methods to remove volumes of waste. The selection process was based on relevant sources of information and the selection criteria were reasonable.

2.5.3 The DOE’s Estimation of Waste Removal Efficiency

This section describes the DOE estimates of removal efficiency through processing of decontamination solutions and use of evacuated canisters. The removal efficiency for the decontamination fluids are reported based on Cs-137 activity. The DOE claims that other key radionuclides are removed in equal proportions due to the homogeneity of the glass waste, and therefore the removal efficiencies for other radionuclides would closely follow that of Cs-137.

The Cs-137 activity assumed before flushing (1,110 TBq [30,000 Ci]) was estimated assuming a typical Cs-137 concentration of 11 GBq/cm^3 (30,000 µCi/cm^3) and a volume of 1,000 L, which is an approximate value (rounded up from 860 L) for the typical working level of 66 cm (26 in) of

glass in the Melter. In support of the concentration assumption, DOE alternatively calculated this value to be 1,104.80 TBq (29,860 Ci) using the measured melter level after processing the last canister associated with Batch 75 (Canister 268) and the concentration of that canister.

In the revised approach following RAIs, the DOE estimates the effectiveness of processing the decontamination solution by comparing the dose rates of the canisters filled before the first batch of decontamination fluids (Batch 75) was fed into the Melter to those filled after all decontamination fluids were processed. The DOE calculates a reduction factor of 5.5 by comparing dose rates on canisters 266-270 (which averaged 272,000 Gy/h [2,716 R/h]) to the dose rates on the two evaluated canisters (which averaged 49,000 Gy/h [490 R/h]). As stated, the processing of decontamination fluids involved sending less contaminated fluids through the Melter in lieu of sending sludge from the tanks through the Melter. Accordingly, the dose rates from the canisters during the processing of these Batches should decrease. The NRC staff does not believe the dose rate measurements are the most appropriate way to present the effectiveness of this removal activity.

In Attachment 3 of its response to RAIs, the DOE also provided a revised estimate of Cs-137 present after each batch of flushing fluids was processed (but before use of the evacuated canisters utilizing concentration measurements from the last canisters of each flushing batch), measurements from the evacuated canisters, and the measured levels of the Melter. This analysis calculates 264.9 TBq (7,160 Ci) remaining in the Melter cavity after flushing but before use of evacuated canisters. The DOE alternatively estimates the value remaining just prior to the evacuated canisters to be approximately 196.4 TBq (5,310 Ci) assuming the measured Cs-137 concentration in the glass shard samples taken from the canisters was 88.1 TBq/g (2,380 Ci/g), multiplied by the 2,230 kg of molten glass present before use of the evacuated canisters (DOE, 2011a). Another separate analysis provided to support the economic impracticality of additional removal estimated that 159.2 TBq (4,303 Ci) remained in the Melter cavity prior to use of the evacuated canisters (Kurasch, 2011). The DOE states that the evacuation canisters removed approximately 2,200 kg of the approximate 2,500 kg that was in the Melter before the evacuated canisters. This is estimated by assuming a starting volume of 1000 L, which is an approximate value (rounded up from 860 L) for the typical working level of 66 cm (26 in) of glass in the Melter, and a density of 2.5 g/cm^3, along with the estimated 300 kg of material that remained in the Melter based on visual inspection (Petkus, 2002b). The DOE also provided a separate estimate of the mass remaining in the Melter cavity in response to RAIs, based on a level of 61.93 cm (24.38 inches) [WVNSCO, 2002b], a curie density of 144.7 GBq/kg (3.91 Ci/kg) of glass, and a glass density of 2.4 g/cm^3. Under these assumptions, the DOE estimates that just prior to the deployment of the evacuation canisters the Melter held 1,830 kg of glass containing about 265 TBq (7,160 Ci). Approximately 91 TBq (2,460 Ci) was estimated to remain after the use of the evacuated canisters (DOE, 2011a).

2.5.4 NRC Evaluation of Waste Removal Efficiency

In its review, the NRC staff requested additional support for the assertion that flushing removed radionuclides in equal proportions. In response, the DOE compared the fractional abundance of selected radionuclides in the waste being fed into the Melter to that of the final flush solutions removed with the evacuated canisters. The DOE noted that the lack of any major difference as evidence for removal in equal proportions.

The NRC staff notes that DOE's estimate of 1,110 TBq (30,000 Ci) present prior to removal activities did not directly account for the possibility that some contamination was likely pushed through the system from the other tanks and equipment into the Melter during the flushing

activities (the intent of flushing the system was to remove contamination from the Melter Holder Feed Tank, transfer the material back to the Concentrator Feed Make-up Tank, and ultimately use this as feedstock to the Melter). Ignoring this possibility could lead to an underestimation of a "starting position" for amount of radionuclides present in the system prior to removal activities. However, due to the other assumptions (e.g, volume of 1000 L) embedded in the estimation of 1,110 TBq (30,000 Ci) that would bias the value high, this omission is not considered to be important.

Because DOE provided three estimates for the amount of cesium activity remaining in the Melter after the decontamination fluids were processed [264.9 TBq (7,160 Ci), 196.4 TBq (5,310 Ci), and 159.2 TBq (4,303 Ci)] there is uncertainty as to the actual effectiveness of the decontamination fluids. The potential for additional removal using this cleaning method is discussed in Section 2.5.5 and 2.5.6. However, since the decontamination fluids were not ever intended to be processed without also using the evacuated canisters, the overall combined efficiency of both these technologies is most important. Depending on whether the dose-to-curie method, which estimated 159.5 TBq (4,310 Ci) of Cs-137 remaining, is applied or DOE's alternative approach which estimated 91.8 TBq (2,480 Ci) remaining, the overall removal efficiency ranges from 85% through 92% of the activity assuming a 1,110 TBq (30,000 Ci) starting point. The NRC staff concludes that while the original estimation of removal efficiency was not technically sufficient, the alternative estimate for removal efficiency provided by the DOE in Attachment 3 of the RAI response is reasonable and sufficient (DOE, 2011b).

Since volume was being added and removed from the system during the processing of the decontamination fluids, it is not appropriate to try to estimate removal efficiency in terms of volume from the flushing stage of cleaning activities. Instead, a volume removal estimate could be given for just the use of the evacuated canisters. The NRC staff notes that if the alternative estimate of the glass remaining in the Melter prior to use of the evacuated canisters is assumed (1,830 kg instead of 2,500 kg), maintaining that 300 kg remained after use of the evacuated canisters, then the removal efficiency for volume is similar (83% instead of 88%).

The NRC staff asked the DOE whether it considered modifications to the flushing method to improve efficiency prior to using the evacuated canisters. The DOE states that another batch of decontamination solution was not used because of concerns over the acid hindering the integrity of the system (Kocialski, 2003). The DOE also stated that a decision was made in 2002 that the flushing had been satisfactorily completed before the evacuated canisters were employed (DOE, 2011a). In reviewing the *HLW Processing Systems Flushing Operations Run Plan* (WVNSCO, 2002b), as well as the noted concerns over the limited life of the Melter, and the risk of its malfunction, the NRC staff concluded that the deployment of the evacuated canisters after flushing with two batches of decontamination solution was adequate and reasonable.

The NRC staff inquired about whether the evacuated canisters performed as well as the DOE expected, and whether the DOE considered using additional evacuated canisters. The DOE stated that two evacuated canisters were expected to be more than sufficient to remove all of the molten glass in the Melter cavity within the design limitations of the canister system, so therefore only two were deployed. Although no specific removal goals were stated, the DOE expected that 7.6 to 15.2 cm (three to six inches) of glass would be left in the bottom of the Melter cavity after use of the evacuated canisters (DOE, 2011b). Because the DOE assumed 20.32 cm (8 in) were remaining, and alternatively calculated 16.5 cm (6.5 in) remaining, the removal technologies can be said to have performed roughly as expected. The DOE provided additional information as to the potential use of a third canister. After the second canister was

deployed, the level in the Melter was below the height of the electrodes, eliminating the ability to heat the glass and maintain a high temperature. Therefore, since a third evacuation was not immediately available, and there was no method for maintaining heat while a third canister could have been produced, the use of a third canister was impractical. Furthermore, if a third canister had hypothetically been available, it would have only been filled to 8% of its capacity, even assuming 100% removal efficiency. Taking these factors into account, the NRC staff concludes that use of two evacuated canisters is reasonable.

2.5.5 The DOE Estimation of Costs and Benefits of Additional Waste Removal

The DOE's decision not to continue with additional removal operations in 2004 was based on a net social cost study (Perdue, 2004). In the draft WIR evaluation, the DOE summarizes a more streamlined cost benefit analysis than what was documented in 2004, relying partially on the Perdue report, as well as other supporting information, including the alternative calculation of the effectiveness of removal through processing additional flushing (Kurasch, 2011).

In 2004, two alternatives were compared to the baseline of no additional removal. Alternative 1 assumed additional flushing while Alternative 2 assumed additional flushing followed by dismantlement of the Melter. At that point in time, DOE determined that the incremental costs far outweighed the benefits of either alternative. In 2008, dismantlement was reconsidered, but was still found to be economically impractical due to its high cost. The DOE states that other disadvantages of dismantlement include increased worker exposure from having to handle additional packages, and uncertainties regarding orphaned waste given that some of the pieces would have concentrations consistent with greater than Class C waste, which currently has no disposal path.

2.5.5.1 Benefits Considered by the DOE

The benefit quantified by DOE in the draft WIR evaluation was the reduction in shielding required for shipment which could potentially lead to reduction in number of LLW shipping and disposal containers, as well as savings in disposal cost due to reducing the size of the package in the case of dismantlement. This benefit is valued to be $100,000 in 2002 dollars.

The DOE also considers a number of benefits related to potential reduction in worker or public dose, but concludes that these benefits would not be of significance. In response to an NRC RAI (NRC, 2011), the DOE claims that the potential social benefits from avoided public exposure due to additional waste removal would be negligible given the low impact of the Melter in its current form. Uncertainty surrounding the potential social benefits is not discussed in the DOE's analysis.

2.5.5.2 Costs Considered by the DOE

The costs included by the DOE in the draft WIR evaluation were primarily related to the cost of extending the shutdown date and continued operation of the facility. DOE conservatively did not quantify other costs such as capital costs specific to the cleaning technology, canister/container costs, the additional processing costs, shipping and disposal costs, as well as worker and public exposure costs. The Vitrification Facility operating cost per month is determined to be $1.89 M per month. An additional flush, assumed to require two weeks, would have cost approximately $1 M.

The Perdue report included a sensitivity analysis to estimate changes in some of the cost parameters that were assumed, such as the time to complete a flush, and the cost to operate the vitrification facility. Specifically, the time to complete a flush was reduced by 50%, and the operating cost per month was reduced by 50%.

2.5.6 The NRC Evaluation of Costs and Benefits of Additional Waste Removal

The NRC staff evaluated the following key assumptions regarding benefits:

- Estimated curies removed from additional flushing
- No benefit from reduced worker exposure
- No benefit from reduced public exposure, including an intruder

In its review, the NRC staff focused on the radioactivity removal that would have resulted from additional flushing, which could have potentially translated into benefits. In response to an NRC RAI (NRC, 2011), the DOE revised its approach for determining the effectiveness of hypothetical additional flushes from the method presented in the Perdue study. The revised method employs actual samples taken from the Melter system, considers total activity as opposed to only Cs-137, and conservatively assumes that the fluids entering the Melter from the additional hypothetical flush would not contain any radioactivity from other parts of the system. The DOE estimates that an additional hypothetical flush would have removed another 50-60% of the remaining radioactivity after canister 275 was processed, or an approximate 92.5 TBq (2,500 Ci). Assuming that the evacuated canisters would have removed the same volume of material after an additional flush (88% of the volume), DOE calculates that the residual activity in the cavity would have been about 8.1 TBq (218 Ci). The residual radioactivity in the entire Melter, including the discharge port, would have been 18.5 TBq (500 Ci). This can be compared to either 175.8 TBq (4,750 Ci) [dose-to-curie estimate], or the alternative estimate using concentration and mass of 82.88 TBq (2,240 Ci). Therefore, an additional flush could have reduced the final activity in a range of 77%-89%. While the NRC staff does not find the approximation method presented in the Perdue study that estimates the curie content removed from the i^{th} flush to be technically robust, the alternative method which produced a much higher potential effectiveness provided in response to the RAI is determined to be adequate and reasonable.

Even though additional removal activities are estimated to have potentially removed much of the residual activity, the additional removal does not translate into quantifiable benefits in terms of worker dose. Due to the already low doses resulting from the Melter without further removal, as well as shielding inherent in the Melter refractory design, the NRC staff agrees that further radionuclide removal would not significantly reduce worker doses at the disposal facility.

In terms of public dose, as stated in NUREG-1854, page 3-11, "*The primary benefit of radionuclide removal is expected to be a reduction in the risk the waste will pose to the general public, including inadvertent intruders…. an analysis of the costs and benefits of additional radionuclide removal will depend in part on the performance assessment and inadvertent intruder analysis predictions*" (NRC, 2007). Since the NRC staff was not requested to review the PAs for the disposal facilities, the staff was unable to independently verify the doses provided and any corresponding benefit of avoided public or intruder exposure by additional radionuclide removal. However, assuming that the dose estimates are accurate, even a large percentage of additional radionuclide removal would have negligible impact on the long-term

dose in comparison to the social costs. If the Melter were to be dismantled, it is likely that some components such as residual glass would need to be disposed of as HLW, which would presumably not be a risk to an intruder but would have high disposal costs since there is currently no disposal option for this type of waste. However, because the costs of additional removal would be so high, the NRC staff concludes that the impact on dose from additional flushing or dismantlement is minimal in comparison to the large costs of additional radionuclide removal.

The NRC staff also evaluated the following key assumptions regarding costs:

- The vitrification operating costs and time to complete a flush
- Storage transport and disposal costs for HLW canisters

Annual operating costs for the Vitrification Facility were estimated to be between $25M and $30M per year ($0.5M per week). The DOE assumed that creation of a single new canister would have taken about two weeks from start to end, thus costing $1.0 M (DOE, 2011a). The sensitivity analysis discussed in the Perdue report assumed a 50% reduction in the operating costs and the time to complete a flush. The NRC staff finds these assumptions regarding cost and timing of additional operation to be reasonable considering canisters 267 through 275 (containing the flush solutions of Batches 76 and 77) were processed over roughly 3 months from May 7, 2002 to August 14, 2002 (Kurasch, 2011). An additional flush could have required longer than 2 weeks, including downtime should the Melter have failed.

While the DOE does not attempt to quantify cost of HLW disposal in its summary cost benefit analysis, the NRC staff notes that the cost of storing, transporting and disposing of a HLW canister to the repository could be substantial given that there is currently no disposal path for HLW canisters.

Based on the discussion presented above, the NRC staff concludes that the costs and benefits considered are conservative and reasonable, and that the costs outweigh the benefits of additional removal activities.

3.0 THE WASTE WILL BE MANAGED TO MEET SAFETY REQUIREMENTS COMPARABLE TO THE PERFORMANCE OBJECTIVES OF 10 CFR PART 61, SUBPART C

This section of the NRC staff's technical review partially covers Section II.B(2)(a)(2) – that waste will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, *Performance Objectives*.

The Performance Objectives listed in 10 CFR Part 61 are as follows:

- §61.41 Protection of the general population from releases of radioactivity.
- §61.42 Protection of individuals from inadvertent intrusion.
- §61.43 Protection of individuals during operations.
- §61.44 Stability of the disposal site after closure.

According to the June 30, 2010 IA, this assessment does not include reviewing the sufficiency of the waste acceptance criteria or the sufficiency of the performance assessment for the potential disposal facilities being considered to receive the waste (§61.41 or §61.42). Instead, NRC is focusing on the requirements for operational radiation protection (§61.43), and the waste form stability (partial review of §61.44).

3.1 Protection of Individuals during Operations

10 CFR §61.43 requires:

> "*Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in 10 CFR Part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by Section 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.*"

The NRC review for this section focused on the DOE commitments to adhere to the appropriate regulations for protection of individuals during operations, and descriptions of how the regulations are implemented with respect to the draft WIR evaluation.

The DOE requirements for protection of individuals during operations are outlined in DOE-M 435.1-1, Section I.E(13) and are as follows:

> "*Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR Part 835, Occupational Radiation Protection, and DOE 5400.5, Radiation Protection of the Public and the Environment.*"

10 CFR §61.43 references 10 CFR Part 20, Standards for Protection Against Radiation, which contains similar radiological protection standards to those of the DOE's for workers and the

public. The State of Texas regulations (Rule §336.726) mirror the NRC regulations in §61.43, and the State of Texas also has comparable dose standards to those of the NRC.[1]

Table 3-1: Comparison of Regulations for Dose Standards

Crosswalk Topics	DOE	NRC	TX (Texas Rule)
Annual Air Emission Limit for Individual Member	DOE Order 5400.5 10 mrem (0.1 Sv)	§20.1101(d) 10 mrem (0.1 Sv)	§336.304 10 mrem (0.1 Sv)
Annual TEDE for Adult Workers	§835.202(a)(1) 5 rem (0.05 Sv)	§20.1201(a) 5 rem (0.05 Sv)	§336.305 5 rem (0.05 Sv)
Any Individual Organ or Tissue Annual Dose Limit for Adult Workers	§835.202(a)(2) 50 rem (0.5 Sv)	§20.1201(a) 50 rem (0.5 Sv)	§336.305 50 rem (0.5 Sv)
Annual Dose Limit to the Lens of the Eye for Adult Workers	§835.202(a)(3) 15 rem (0.15 Sv)	§20.1201(a) 15 rem (0.15 Sv)	§336.305 15 rem (0.15 Sv)
Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers	§835.202(a)(4) 50 rem (0.5 Sv)	§20.1201(a) 50 rem (0.5 Sv)	§336.305 50 rem (0.5 Sv)
Limit on Soluble Uranium Intake	DOE Order 440.1A 2.4 mg/week	§20.1201(e) 10 mg/week	§336.305 10 mg/week
Dose Equivalent to Embryo/Fetus	§835.206(a) 0.5 rem (5 mSv)	§20.1208(a) 0.5 rem (5 mSv)	§336.312 0.5 rem (5 mSv)
Dose Limit for Individual Members of the Public (Total Annual Dose)	DOE Order 5400.5 100 mrem (1 mSv)	§20.1301(a) 100 mrem (1 mSv)	§336.313 100 mrem (1 mSv)
Dose Limit for Individual Members of the Public (Dose Rates in Unrestricted Areas)	§835.602 0.05 mrem/hr (0.005 mSv)	§20.1301(a) 2 mrem/hr (0.02 mSv)	§336.313 2 mrem/hr (0.02 mSv)
Dose Limits for Members of the Public with Access to Controlled Areas	§835.208 0.1 rem (0.001 Sv)	§20.1301(b) 0.1 rem (0.001 Sv)	§336.313 0.1 rem (0.001 Sv)
As Low As Reasonably Achievable	§835.2	§20.1003	§336.2

[1] NRC is not including a discussion about Nevada State Regulations because the Nevada National Security Site is a Federal facility and doesn't come under the purview of state regulations.

3.2 Waste Form Stability

10 CFR §61.44 states:

> "*The disposal facility must be sited, designed, used, operated, and closed to achieve long term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.*"

As specified in the IA, dated June 30, 2010, this review focuses on waste form stability and does not include review of the stability of the disposal site. Review of waste form stability involves verification of the structural stability of the waste after site closure as well as verification that void spaces in the waste will not cause differential settling of the waste. Waste form instability could compromise the disposal facility's ability to demonstrate compliance with §61.41 and §61.42.

The following three characteristics of the waste form support the argument that the waste is in a stable form: the structural integrity of the Melter itself; the type IP-2 package which contains the Melter; and the grout that will fill the void spaces in the Melter and the package.

The outer shell of the Melter is made of stainless steel and the interior is lined with a composite of various refractory materials to withstand high temperatures, specifically Monofrax™ K-3, a chrome alumina fusion cast refractory with high corrosion resistance used in applications such as fiberglass furnaces and nuclear encapsulation melters (RHI). This Industrial Package (IP)-2 type shipping container is formed of steel that is 15.2 cm (six inches) thick on the sides and 10.2 cm (four inches) thick on the top and bottom.

As part of the Criterion 3 demonstration, DOE shows that the Vitrification Melter is in a solid physical form. DOE describes in the draft WIR that "the void spaces in the Vitrification Melter and its waste container will be filled with grout consisting of low-density cellular concrete, to help stabilize the component within the shipping container during transport and to encapsulate surface contamination" (DOE, 2011a).

As part of the Criterion 2 demonstration, DOE shows that the Melter meets the Waste Acceptance Criteria for the Nevada National Security Site, and that DOE would confirm that the waste package meets the WCS waste acceptance criteria should the Melter be disposed of at WCS. The draft WIR states that "(t)he low-density cellular concrete will be compatible with the waste acceptance criteria and will help reduce the possibility of disposal cell subsidence in the area of the buried waste container by eliminating void spaces" (DOE, 2011a).

The NRC staff concludes that the waste is in a stable form due to the structural integrity afforded by the Melter, the IP-2 package, and the grout that will fill void spaces. Furthermore, mechanisms or processes that would cause significant degradation to the waste form and packaging which would impact structural stability are not expected to occur within the period of performance. As previously noted, this review did not evaluate the durability of the waste form or packaging with respect to meeting the §61.41 or §61.42 performance objectives (e.g., release rates).

4.0 THE WASTE WILL NOT EXCEED CLASS C CONCENTRATION LIMITS AND WILL BE MANAGED IN ACCORDANCE WITH DOE REQUIREMENTS AS LLW

This section of the TER pertains to the third criterion in Section II.B(2)(a) of DOE-M 435.1-1: (i) that the Vitrification Melter waste package will be in a solid physical form, (ii) will not exceed Class C concentration limits, and (iii) will be managed in accordance with DOE requirements as LLW as applicable. The DOE indicated that the focus of NRC's review under this criterion is waste form stability and classification.

4.1 The DOE's Waste Classification Approach

Criterion 3 of DOE-M 435.1-1 Section II.B (2)(a) is reproduced below for ease of reference:

> "*[The wastes] are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of this Manual, provided the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in [Code of Federal Regulations] §61.55, Waste Classification, or will meet alternative requirements for waste classification and characterization as DOE may authorize.*"

Table 4-1 presents the DOE's waste classification results for the Vitrification Melter and the results support a conclusion that the Vitrification Melter does not exceed the Class C concentration limits.

To evaluate the DOE's approach for determining the class of the Melter, the staff reviewed the waste classification requirements found in 10 CFR §61.55. Table 1 of §61.55 contains concentration limits for long-lived radionuclides including a specific class of radionuclides, alpha-emitting transuranic radionuclides with half-lives greater than 5 years. DOE listed several radionuclides that were members of this class (e.g., Np-237, Pu-238, Pu-239, Pu-240, Am-241, Am-243, Cm-243, and Cm-244) along with their corresponding contributions to the sum of fractions. Table 2 of §61.55 contains a list of concentration limits for relatively short-lived radionuclides. DOE also evaluated the contributions of short-lived radionuclides found in Table 2 to the sum of fractions. The sum of fractions approach used to determine waste classification for mixtures of radionuclides is described in §61.55(a)(7). Because the Vitrification Melter contains a mixture of long- and short-lived radionuclides, DOE applied §61.55(a)(5) to determine waste classification. 10 CFR §§61.55(a)(5) and (a)(7) are reproduced below for additional background information on the waste classification calculations.

> *§61.55(a)(5), "Classification determined by both long- and short-lived radionuclides. If radioactive waste contains a mixture of radionuclides, some of which are listed in Table 1, and some of which are listed in Table 2, classification shall be determined as follows: (i) If the concentration of a nuclide listed in Table 1 does not exceed 0.1 times the value listed in Table 1, the class shall be determined by the concentration of the radionuclides listed in Table 2. (ii) If the concentration of a nuclide listed in Table 1 exceeds 0.1 times the value listed in Table 1 but does not exceed the value in Table 1, the waste shall be Class C, provided the concentration of nuclides listed in Table 2 does not exceed the value shown in Column 3 of Table 2."*

§61.55(a)(7), "The sum of the fractions rule for mixtures of radionuclides. For determining classification for waste that contains a mixture of radionuclides, it is necessary to determine the sum of fractions by dividing each nuclide's concentration by the appropriate limit and adding the resulting values. The appropriate limits must all be taken from the same column of the same table. The sum of the fractions for the column must be less than 1.0 if the waste class is to be determined by that column. Example: A waste contains Sr-90 in a concentration of 50 Ci/m^3 and Cs-137 in a concentration of 22 Ci/m^3. Because the concentrations both exceed the values in Column 1, Table 2, they must be compared to Column 2 values. For Sr-90 fraction 50/150 = 0.33; for Cs-137 fraction, 22/44 = 0.5; the sum of the fractions = 0.83. Because the sum is less than 1.0, the waste is Class B."

In its statement of work, the DOE requested that the NRC evaluate the waste classification calculations against NRC's guidance found in NUREG-1854, Chapter 3, Section 3.5.1.1, "Concentration Averaging" (NRC, 2007). The NRC's guidance on concentration averaging considers three categories of averaging based on (1) physical homogeneity, (2) stabilization, and (3) site-specific (intruder) analysis considerations.

Because residual radioactivity associated with the vitrification melter is not physically homogeneous (not evenly distributed across the entire melter package), NRC's guidance on concentration averaging would also allow DOE to consider site-specific factors in evaluating the risk to an inadvertent intruder under Option (3). However, DOE did not elect to use this option although it may have led to a smaller Class C sum of fractions, thereby providing additional confidence that the waste is not greater than Class C.

DOE elected to use a method that is consistent with Option (2) (stabilization), although DOE did not explicitly take full advantage of this option. For example, DOE did not attempt to average residual radioactivity over the weight of stabilizing grout DOE is planning to use to fill void spaces in the Vitrification Melter and its waste container. However, DOE may have made reasonable arguments on how at least a fraction of the stabilizing grout may have been necessary to encapsulate or assist with immobilization of contamination within the Melter or on melter surfaces. However, it is NRC's position that DOE is, in essence, averaging the residual radioactivity over the *volume* of grout that DOE plans to use to fill void spaces within the Vitrification Melter itself, as the total volume of the Vitrification Melter is used to calculate concentrations for the purposes of comparison against Class C concentration limits found in §61.55. This point, and NRC's conclusions regarding the acceptability of this approach, is discussed further below.

Radionuclide concentrations for purposes of waste classification for the West Valley Vitrification Melter are based on averaging the total activity of each radionuclide to the total weight or volume of the Melter, not considering the weight of the shipping container (i.e., shielded IP-2) or the weight of stabilizing grout used to fill voids in the Melter and voids between the Melter and the shipping container. The total weight of the Melter used in the calculation was 48,000 kg (106,000 lbs) and the volume used was 21 m^3 (750 ft^3). The resulting sum of fractions was 0.95 for Table 1 radionuclides and 0.05 for Table 2 radionuclides, as shown in Table 4-1.

4.2 NRC Evaluation of the DOE's Waste Classification Approach

The NRC staff has evaluated DOE's methodology for classifying waste and finds the approach an acceptable application of Category 2 of NRC staff's guidance on concentration averaging found in NUREG-1854 (NRC, 2007) and consistent with Section 3.4 of NRC's BTP (NRC, 1995). It is important to note that NRC's guidance on concentration averaging in NUREG-1854 does not replace the BTP. Rather NUREG-1854 attempts to apply the concentration averaging guidance principles specifically to WIR applications, providing additional flexibility, where appropriate, that are consistent with the general principles of the BTP. NRC considers DOE's waste classification calculations consistent with Section 3.4 of the NRC's BTP entitled "Contaminated Materials." Section 3.4 of NRC's BTP provides for averaging over the total weight of displaced volume of the contaminated item with major void volumes subtracted from the envelope volume.

As mentioned above, while the NRC would note that large void volumes should usually not be part of the averaging process as indicated in Section 3.4 of NRC's BTP, the NRC finds DOE's use of averaging appropriate in this case. In this case, longer-lived transuranics (§61.55, Table 1 radionuclides: Pu-238 and Am-241) dominant the sum of fractions, and the concentration limits in §61.55 Table 1 for these transuranics are based on averaging over the *weight*, not the volume, of the component. Therefore, the impact of averaging over larger void volumes within the Melter is not expected to significantly impact the waste classification results. In fact, the DOE performed a sensitivity analysis on averaging over the reduced volume and reached the same conclusion. Using a reduced volume of 50% or 10.6 m^3 (375 ft^3), the resulting sum of fractions was 0.95 for Table 1 and 0.09 for Table 2 radionuclides, as expected. The sensitivity analysis appears appropriate and sufficiently bounding to support DOE's conclusions regarding waste classification.

Consistent with Section 2.2 of this TER, the NRC staff notes that the Melter could be greater than Class C if uncertainty in the inventory of key radionuclides found in §61.55 is considered simply due to the fact that the sum of fractions without including uncertainty is so near to unity as shown in Table 2-1. This possibility depends on how conservative the dose-to-curie inventory approach is compared to other sources of uncertainty. However, the DOE did not take credit for the weight of stabilizing grout used to encapsulate the residual radioactivity. As stated above, Pu-238 and Am-241 dominate the sum of fractions for waste classification and are based on averaging the inventory over the weight of the Melter. Because the uncertainties both in the scaling factors for transuranics (see Section 2.2 of this TER) and in the overall range in transuranic inventory is expected to be modest (within a factor of two), these uncertainties are expected to be compensated by the conservatism of the waste classification calculations. In other words, the weight of the waste package could increase by over a factor of two if the DOE attempted to take credit for encapsulation of the waste and the sum of fractions could decrease by a corresponding factor of two and still meet Class C concentration limits considering uncertainty in the inventory. In addition, as DOE showed with the alternative method of estimating inventory activity, the original dose-to-curie method may have overestimated the remaining activity by almost 50%. Therefore, the NRC staff concludes that the DOE's assessment that the Melter is Class C considering uncertainty in the volume of the Melter and uncertainty in the inventory estimates is technically sufficient.

As noted in Section 2.2, the NRC staff reviewed historical inventories and compared those to the radionuclides considered for waste classification purposes. In response to an RAI, the DOE clarified that certain transuranics with half-lives greater than 5 yrs that were listed in the sludge/supernatant inventory (Pu-242, Am-242m, Cm-245, Cm-246), were not originally

considered in the waste classification because they were present in small quantities. For consistency, the DOE added these radionuclides to the waste classification table. Since they are present in small quantities, inclusion of these transuranics does not significantly impact the sum of fractions.

The Melter may be transported to the Nevada National Security Site Area 5 Radioactive Waste Management Site for disposal. The Vitrification Melter waste package would be disposed of as LLW and managed in accordance with DOE requirements for LLW disposal found in Chapter IV of DOE-M 435.1-1. The waste form meets the site's acceptance criteria and the waste profile has been formally approved by the potential disposal facility. The DOE may alternatively decide to ship the waste to the commercially operated WCS federal facility waste disposal facility in Texas. The State of Texas Class C concentration limits are consistent with §61.55 and would meet the Class C LLW concentration limits established in the Texas Administrative Code.

Based on the NRC's review of DOE's draft WIR evaluation and supporting references, the NRC considers the DOE's conclusions to be adequate and reasonable such that the DOE can meet Criterion 3 of DOE-M 435.1-1 related to management of WIR as LLW.

Table 4-1: Vitrification Melter Waste Concentration Results

Isotope	Estimated Activity (Ci) *	Class C Limit**		Melter Concentration**		Fraction	
		Ci/m^3	nCi/g	Ci/m^3	nCi/g	**Table 1** (2)	**Table 2** (2)
C-14	2.12E-02	8.00E+00		1.00E-03		1.25E-04	
Ni-63	1.01E+00	7.00E+02		4.76E-02			6.81E-05
Sr-90	2.47E+02	7.00E+03		1.17E+01			1.66E-03
Tc-99	1.11E-02	3.00E+00		5.24E-04		1.75E-04	
I-129	5.64E-03	8.00E-02		2.66E-04		3.33E-03	
Cs-137	4.31E+03	4.60E+03		2.03E+02			4.42E-02
Np-237	6.20E-03		1.00E+02		1.29E-01	1.29E-03	
Pu-238	**6.84E-01**		**1.00E+02**		**1.42E+01**	**1.42E-01**	
Pu-239	1.59E-01		1.00E+02		3.31E+00	3.31E-02	
Pu-240	1.21E-01		1.00E+02		2.52E+00	2.52E-02	
Pu-241	3.12E+00		3.50E+03		6.49E+01	1.85E-02	
Pu-242			1.00E+02		4.67E-04	4.67E-06	
Am-241	**3.00E+00**		**1.00E+02**		**6.24E+01**	**6.24E-01**	
Am-242m			1.00E+02		1.90E-03	1.90E-05	
Am-243	3.50E-02		1.00E+02		7.28E-01	7.28E-03	
Cm-242	7.33E-02		2.00E+04		1.52E+00	7.62E-05	
Cm-243	1.68E-02		1.00E+02		3.49E-01	3.49E-03	
Cm-244	4.35E-01		1.00E+02		9.04E+00	9.04E-02	
Cm-245			1.00E+02		2.25E-01	2.25E-03	
Cm-246			1.00E+02		3.66E-02	3.66E-04	
Sums of fractions(3)						**9.52E-01**	**4.59E-02**

NOTES: (1) From WSMS (2008), the calculations are based on the Vitrification Melter weight and size. The activity estimates used were as of October 1, 2004; the activities are now somewhat lower due to radioactive decay. The weight used in the calculation was 48,094 kilograms (106,000 pounds) and the volume used was 21.2 cubic meters (750 cubic feet).

(2) Table numbers refer to 10 CFR §61.55, Tables 1 and 2. (Table I and Table II to Appendix E to Rule §336.362 of the Texas Administrative Code are identical to NRC's Table 1 and Table 2.)

(3) As a sensitivity analysis related to the Vitrification Melter volume, another calculation was performed with the volume reduced by 50 percent to 10.6 cubic meters (375 cubic feet). The resulting sums for fractions were 9.52E-01 for Table 1 and 9.19E-02 for Table 2 (WSMS, 2008).

*Conversion to GBq from Ci, multiply by 37 GBq/Ci.

**Conversion to Bq from nCi, multiply by 37 Bq/nCi

5.0 THE WASTE HAS BEEN PROCESSED AND DISPOSED OF WITH ADEQUATE QUALITY ASSURANCE

5.1 The DOE Quality Assurance Program

The DOE describes the management controls to assure quality in Appendix B of the draft WIR evaluation. It covers the following: (1) requirements and procedures, (2) quality assurance, (3) data evaluation and engineering calculations, (4) report preparation and review, (5) personnel qualification, (6) oversight and independent review, and (7) document and record control.

5.2 The NRC Evaluation of DOE's Quality Assurance Program

In its statement of work, the DOE requested that the NRC evaluate its quality assurance program against NRC's guidance found in NUREG-1854. Chapter 8 of NUREG 1854 suggests that the staff choose a sample of analyses associated with the evaluation (based on risk-significance) of which to review quality assurance. Because the inventory characterization is important for demonstrating removal to the maximum extent practical, as well as for properly classifying the waste, this section discusses the NRC's review of the dose-to-curie measurements and the sampling performed on the glass from the evacuated canister.

The NRC staff has found that the data are traceable to their sources through all calculations and data reductions, and the data have been obtained or qualified using an acceptable quality assurance program. As described in Section 2.2 of this report, data were validated in accordance with the requirements of the Characterization Management Plan for the Facility Characterization Project, which includes specific data quality objectives (Michalczak, 2004).

The Characterization Management Plan for the Facility Characterization Project outlines procedures for field custody, sampling documentation, transfer of custody and shipment, laboratory custody, requirements for instrument testing, maintenance and calibration and data management. The plan also defines the procedures for dose rate measurements and the dose to curie modeling using Microshield®. The conceptual model used in Microshield®, as well as the dose rates used as input to the software are documented in WMG (2004a) and Lachapelle (2003). As described in Section 2.1, Microshield® was used to model the relatively simple geometry of the discharge port, and QAD-GCCP-A(2) code was applied for point kernel modeling of the complex geometry in the Melter cavity. This code has been used to support decommissioning activities and was evaluated for use by the NRC (CNWRA, 1995).

In conclusion, the NRC staff finds that DOE followed adequate quality assurance and quality control procedures, and that the use of software was appropriate in calculating the inventory estimates.

6.0 CONCLUSIONS

6.1 NRC Review: Waste has been Processed to Remove Key Radionuclides to the Maximum Extent that is Technically and Economically Practical

Overall Conclusion: Based on the NRC's review of DOE's draft WIR evaluation and supporting references, the NRC considers the DOE's conclusions to be adequate and reasonable such that the DOE can meet the NRC-reviewed portions of Criterion 1 of DOE-M 435.1-1 related to the removal of key radionuclides. This is based on the following specific topical conclusions.

Waste Inventory:

- With regard to information about waste generation and treatment activities, the NRC staff finds that the predicted physical and chemical forms of radionuclides, as described by DOE, are consistent with the properties of contributing waste streams and treatment processes.

- By assuming a 66 cm (26 in) starting level and subtracting the volume removed by the evacuated canisters, the DOE calculates a remaining heal of 16.5 cm (6.5 in). While the NRC staff does not believe that the memo reference provides sufficient technical basis supporting a 20.3 cm (8 in) level (Petkus, 2002b), the NRC staff finds DOE's independent assessment to be technically acceptable and therefore agrees that a 20.3 cm (8 in) assumption is conservative.

- Given that the DOE's alternative calculation provided additional support that the volume was not underestimated, the NRC agrees that the DOE's assumptions, regarding the activity remaining in the Melter, are conservative.

- NRC believes that the impact of uncertainties due to sample variability, analytical methods, heterogeneity, and volume estimates should be evaluated.

NRC Evaluation of Key Radionuclides:

- The NRC staff was not requested to review the PAs for either disposal site, so did not independently evaluate the importance of the radionuclides that were listed as part of the inventory but were not included as key radionuclides. However, because the technologies that the DOE applied or the alternative technologies considered for removal of radionuclides did not target specific key radionuclides and instead removed bulk volume from the Melter, the DOE's cleaning approach effectively removed all radionuclides, as opposed to only the key radionuclides. Therefore, the NRC staff does not have to make a determination on whether or not the key radionuclide list is appropriate because the staff does not rely on this list for coming to a conclusion about removal to the maximum extent practical. This is not to imply that use of bulk removal technologies alone eliminates the importance of the key radionuclide list for waste evaluations, but instead, the key radionuclide list is not important for this specific review due to both the type of removal technologies and the results of the cost benefit analysis.

Waste Removal Technologies:

The NRC staff has reviewed DOE's basis for the radionuclide removal technology selected and has concluded that DOE evaluated a reasonable range of technologies that included methods to remove volumes of waste. NRC staff concludes that the selection process is based on relevant sources of information and the selection criteria are reasonable.

Waste Removal Efficiency:

- The NRC staff concludes that the estimation of removal efficiency is reasonable and sufficient (DOE, 2011b).

- The NRC staff concludes that the deployment of the evacuated canisters after flushing with two batches of decontamination solution was adequate and reasonable. NRC staff reviewed the *HLW Processing Systems Flushing Operations Run Plan* (WVNSCO, 2002b), as well as the noted concerns over the limited life of the Melter, and the risk of its malfunction.

- With regard to the DOE's use of a third canister, the NRC staff concludes that use of two evacuated canisters is reasonable. Following deployment of the second canister, the level in the Melter was below the height of the electrodes, eliminating the ability to heat the glass and maintain a high temperature. Since a third evacuation was not immediately available, it was an impractical option.

Costs and Benefits of Additional Waste Removal:

- In conclusion, because the costs are so high, the NRC staff concludes that DOE's assumption that the impact on dose would be minimal in comparison to the large costs of additional radionuclide removal is reasonable. While the DOE does not attempt to quantify cost of disposal in its summary cost benefit analysis, the NRC staff notes that the cost of storing, transporting, and disposing of a HLW canister to the repository could be substantial given that there is currently no disposal path for HLW canisters.

- The NRC staff agrees that further radionuclide removal would not significantly reduce worker doses at the disposal facility. Even though additional removal activities may have potentially removed much of the residual activity, the additional removal did not translate into quantifiable benefits in terms of worker dose due to the already low doses resulting from the Melter without further removal, as well as shielding inherent in the Melter refractory design.

- The NRC staff agrees that the costs of dismantlement exceed the benefits. If the Melter were to be dismantled, it is likely that some components such as residual glass would need to be disposed of as HLW, which would presumably not be at risk to an intruder but would have high disposal costs since there is currently no disposal option for this type of waste.

6.2 NRC Review: Waste will be Managed to Meet Safety Requirements Comparable to the Performance Objectives of 10 CFR 61, Subpart C

Conclusion: Based on the NRC's review of DOE's draft WIR evaluation and supporting references, the NRC considers the DOE's conclusions to be adequate and reasonable such that the DOE can meet the NRC-reviewed portions of Criterion 2 of DOE-M 435.1-1 related to Safety Requirements associated with the Performance Objectives of 10 CFR 61, Subpart C. This is based on the following specific topical conclusions.

The scope of this review included verification of the structural stability of the waste after site closure as well as verification that void spaces in the waste will not cause differential settling of the waste.

This review does not include the sufficiency of the waste acceptance criteria or the sufficiency of the PA for the potential disposal facilities being considered to receive the waste (§61.41 or §61.42). This review also did not include an assessment of the stability of the disposal site, an evaluation of the durability of the waste form or packaging with respect to meeting the §61.41 or §61.42 performance objectives (e.g., release rates).

Waste Form Stability:

- The NRC staff concludes that the waste is in a stable form due to the structural integrity afforded by the Melter, the IP-2 package, and the grout that will fill void spaces. Furthermore, mechanisms or processes that would cause significant degradation to the waste form and packaging which would impact structural stability are not expected to occur within the period of performance.

6.3 NRC Review: Waste will not Exceed Class C Concentration Limits and will be Managed in Accordance with DOE Requirements as LLW

Conclusion: Based on the NRC's review of DOE's draft WIR evaluation and supporting references, the NRC considers the DOE's conclusions to be adequate and reasonable such that the DOE can meet the NRC-reviewed portions of Criterion 3 of DOE-M 435.1-1 related to management of WIR as LLW. This is based on the following specific topical conclusions.

- The NRC staff has evaluated DOE's methodology for classifying waste and finds the approach an acceptable application of Category 2 of NRC staff's guidance on concentration averaging found in NUREG-1854 (NRC, 2007) and consistent with Section 3.4 of NRC's BTP (NRC, 1995).

- The NRC staff reviewed those radionuclides that were part of the Inventory List but were not part of the Waste Classification List and concludes that omission of these radionuclides in the Waste Classification List is in accordance with 10 CFR §61.55.

- While the NRC would note that large void volumes should usually not be part of the averaging process as indicated in Section 3.4 of NRC's BTP, the NRC finds DOE's use of averaging appropriate in this case, as the voids will be filled before disposal.

- If DOE had taken credit for the weight of stabilizing grout used to encapsulate the residual radioactivity, the concentrations of Pu-238 and Am-241, which dominate the sum of fractions, could have been reduced by about half.

- The NRC staff recognizes that the Melter could be greater than Class C if uncertainty in the inventory is considered and assuming the dose-to-curie inventory estimate is accurate since the sum of fractions without including uncertainty is close to unity (as shown in Table 2-1). However, as DOE demonstrated with the alternative inventory estimate, inventory uncertainties are compensated by the fact that DOE did not take credit for the weight of the stabilizing grout in calculating the sum of fractions. Therefore, the NRC staff has reasonable assurance that the Melter could be not greater than Class C considering uncertainty in the volume of the Melter and uncertainty in the inventory estimates.

6.4 NRC Review: Waste has been Processed and Disposed of with Adequate Quality Assurance

Conclusion: The NRC staff concludes that DOE followed adequate quality assurance and quality control procedures, and that the use of software was appropriate in calculating the inventory estimates.

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