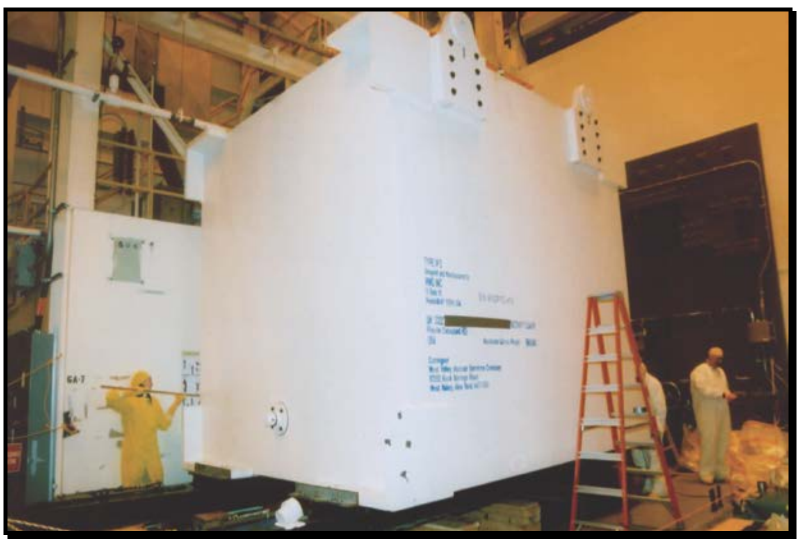
Waste-Incidental-to-Reprocessing Evaluation for the West Valley Demonstration Project

Vitrification Melter



February 2012

Prepared by the
U.S. Department of Energy
West Valley, New York

This page is intentionally blank.

CONTENTS

CONTENTS

CONTENTS

CONTENTS

FIGURES (continued)

TABLES

NOTATION

Acronyms and Abbreviations

ALARA	as low as reasonably achievable
Am	americium
CFR	Code of Federal Regulations
Cm	curium
Co	cobalt
Cs	cesium
DOE	Department of Energy
FR	Federal Register
HLW	high-level waste
I	iodine
IP	industrial package
K	1,000
K	potassium
LLW	low-level waste
Ni	nickel
Mn	manganese
N	north
NDA	NRC-Licensed Disposal Area
Np	neptunium
NRC	Nuclear Regulatory Commission
NYSERDA	New York State Energy Research and Development Authority
PE-g	plutonium equivalent grams
Pu	plutonium
PUREX	plutonium uranium extraction [process]
S	south
SDA	State-Licensed Disposal Area
Sr	strontium
Tc	technetium
THOREX	thorium uranium extraction [process]
U	uranium
WCS	Waste Control Specialists
WIR	waste incidental to reprocessing
WSMS	Washington Safety Management Solutions
WVDP	West Valley Demonstration Project
WVES	West Valley Environmental Services
WVNSCO	West Valley Nuclear Services Company

NOTATION

Units

Bq	Becquerel
Ci	curie
cm	centimeter
cm^2	square centimeter
cm^3	cubic centimeter
g	gram [mass]
h	hour
kg	kilogram, 1,000 grams
L	liter
m	meter
m^2	square meter
m^3	cubic meter
mg	milligram, 0.001 gram
mrem	millirem, 0.001 Roentgen equivalent man (rem)
millirem	0.001 Roentgen equivalent man
mSv	millisievert, 0.001 sievert
mL	milliliter, 0.001 liter
µCi	microcurie, 0.000001 curie
µL	microliter, 0.000001 liter
nCi	nanocurie, 10^{-9} curie
pCi	picocurie, 10^{-12} curie
R	Roentgen
rem	Roentgen equivalent man
s	second
Sv	sievert
y	year

1.0 INTRODUCTION

Section Purpose

The purpose of this section is to provide introductory information that lays the foundation for detailed discussions in later sections.

Section Contents

This section describes the purpose and scope of this evaluation, identifies the technical requirements on which it is based, summarizes the background, and outlines the contents of the rest of the evaluation.

Key Points

- The Department of Energy has evaluated whether the Vitrification Melter at the West Valley Demonstration Project in New York meets the waste-incidental-to reprocessing criteria of Department of Energy Manual 435.1-1, *Radioactive Waste Management Manual*.
- Reprocessing of irradiated nuclear fuel at the West Valley site produced approximately 600,000 gallons of high-level radioactive waste.
- The Department of Energy's West Valley Demonstration Project pretreated this waste to partition it into a high activity waste stream that was vitrified into borosilicate glass, and low activity waste streams that were solidified for disposal as low-level waste.
- The Vitrification Melter was used in the vitrification process to turn high-level waste slurry and glass formers into homogenized molten glass.
- The Department is responsible for disposal of low-level radioactive waste produced by the solidification of high-level waste under the West Valley Demonstration Project, as part of the Department's obligations under the West Valley Demonstration Project Act.
- The Vitrification Melter has been characterized for radioactivity, determined to have radionuclide concentrations that do not exceed limits for Class C low-level radioactive waste, and will be grouted for stability purposes and packaged for shipment to an offsite low-level waste disposal facility.
- To dispose of the used Vitrification Melter offsite as low-level radioactive waste, the Department must first demonstrate it meets criteria of Department of Energy Manual 435.1-1, *Radioactive Waste Management Manual*, which it has accomplished by the evaluations described in this evaluation.
- This evaluation was issued initially in draft form to facilitate consultation with the U.S. Nuclear Regulatory Commission as well as state and public review and comment, consistent with the Department's policy.
- The Department is making its final determination that the Vitrification Melter is not high-level waste after consideration of this evaluation, the U. S Nuclear Regulatory Commission comments, and public comments on the draft evaluation.

1.1 Purpose

The purpose of this waste-incidental-to-reprocessing evaluation is to serve as the basis for the Department of Energy's (DOE's) determination that the Vitrification Melter, which contained pre-treated high-level waste (HLW) at the West Valley Demonstration Project (WVDP) site in Western New York, meets the waste-incidental-to-reprocessing criteria, is not HLW, and may be managed as low-level waste (LLW) pursuant to DOE Manual 435.1-1, *Radioactive Waste Management Manual*.

This equipment was used in DOE's process to vitrify liquid HLW that was generated by commercial reprocessing of spent nuclear fuel by Nuclear Fuel Services, Inc. from 1966 to 1972 and stored in underground waste tanks at the West Valley site. HLW is the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations, and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation (DOE Manual 435.1-1).

DOE plans to dispose of this equipment offsite to meet its obligations under the *West Valley Demonstration Project Act*, Public Law 96-368, 42 U.S.C 2010a, which is described below.

Waste-Incidental-to-Reprocessing Requirements

The term *waste incidental to reprocessing* refers not to a type of waste but rather to a "process", whereby "[c]ertain waste streams produced during the generation of high-level waste may be determined to be non-high-level waste through the waste-incidental-to-reprocessing determination process." (DOE Guide 435.1-1). DOE Manual 435.1-1 provides two methods for determining whether waste associated with spent nuclear fuel reprocessing can be determined to be incidental to reprocessing and managed as LLW: the citation method and the evaluation method.

The citation method applies to radioactive wastes such as, but not limited to, contaminated job wastes including laboratory items such as clothing, tools, and equipment which have been found to routinely meet the criteria for management as LLW.

Consistent with DOE Guide 435.1-1, the equipment addressed in this determination does not fall within a category of materials to which DOE considers the citation process can be applied. Therefore, the evaluation method was used for the subject equipment as described in this waste-incidental-to-reprocessing evaluation.

1.2 Scope and Technical Basis

This evaluation applies only to the WVDP Vitrification Melter and to no other equipment. The Vitrification Melter was previously decontaminated and characterized. Such considerations are discussed further in Section 4.2 below.

This evaluation has been prepared in accordance with DOE Manual 435.1-1, *Radioactive Waste Management Manual*, following guidance in DOE Guide 435.1-1, *Implementation Guide For Use With DOE M 435.1-1*. The method used involved evaluating whether the used Vitrification Melter is incidental to reprocessing and can be managed under DOE's authority in accordance with requirements for LLW waste. Criteria in Section II.B(2)(a) of DOE Manual 435.1-1 for determining that waste is incidental to reprocessing, is not HLW, and can be managed as LLW are that the wastes:

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

(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 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] 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize.

This evaluation focused on the criteria of DOE Manual 435.1-1, Section II.B(2)(a) summarized above, which are discussed in Section 3 of this evaluation and addressed in detail in Sections 4, 5, and 6, respectively.

Although criteria in DOE Manual 435.1-1 for managing evaluated waste or equipment as LLW are generally similar to the provisions in Section 3116(a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, Public Law 108-375, that Act does not apply to the West Valley site. Nevertheless, as a matter of policy, DOE considered the Section 3116(a) criteria for perspective and information. This matter is addressed in detail in Appendix E.

Waste-incidental-to-reprocessing criteria were also established by NRC as part of the decommissioning criteria for the WVDP in accordance with the WVDP Act (NRC 2002 and NRC 2003). However, these criteria were issued "for the classification of reprocessing wastes that will likely remain in tanks at the site after the HLW is vitrified" and "to clarify the status of and classify any residual waste present after cleaning of the HLW tanks." These statements, which appear in Section 6.4 of the NRC Implementation Plan (NRC 2003), indicate that these criteria apply to the HLW tanks themselves.

NUREG-1854, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations* (NRC 2007), reiterates that these NRC criteria apply to "any residual materials remaining at the [West Valley] site, including any incidental [to reprocessing] waste." NRC also acknowledges in NUREG-1854 that "For [West Valley] offsite waste disposal, it is DOE's responsibility to determine which [waste-incidental-to-reprocessing] criteria are applicable; for example, DOE may decide to apply DOE Order 435.1."

1.3 Consultation and Opportunity for Public Review

DOE consulted with the U.S. Nuclear Regulatory Commission (NRC) and made the evaluation available for state and public review in draft form (DOE 2011a) before making the final determination. This consultation was consistent with guidance in DOE Guide 435.1-1, *Implementation Guide For Use With DOE Manual 435.1-1*, which states that while formal involvement by NRC in making incidental waste determinations is not required, NRC involvement as a consultant to Field Offices and Programs on technical issues is recommended. It was also consistent with NRC's role in providing consultation on DOE activities related to the WVDP as expressed in the DOE-NRC Memorandum of Understanding on this project (DOE and NRC 1981).

Making the draft evaluation available to the public at the time it was provided to NRC for review provided stakeholders an opportunity to review it and submit comments to the Department before the final determination was made.

The NRC submitted a request for additional information in connection with its review (NRC 2011a). DOE provided written responses to the request for additional information (DOE 2011c). On September 30 2011, NRC issued its Technical Evaluation Report to document its review of the draft evaluation (NRC 2011b). The executive summary of this report states that:

> "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."

As indicated in the Technical Evaluation Report, the NRC review was focused on assessing whether the methodology that 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 Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations* (NRC 2007), to accomplish its review. The NRC's review focused on the following general topics, as they relate to the criteria in DOE Manual 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 were 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 assessed the dose-to-curie measurements and the sampling performed on the glass from the evacuated canisters." (NRC 2011b)

DOE also considered public comments on the draft evaluation before making its final determination.[1] The DOE responses to these comments can be found on the West Valley Demonstration Project website (http://www.wv.doe.gov/) and the website of the DOE Office of Environmental Management (http://www.em.doe.gov/Pages/EMHome.aspx).

1.4 Background

The following general information is provided to put the evaluation into context. Section 2 provides more detailed background information on reprocessing of spent nuclear fuel, HLW waste pretreatment, HLW vitrification, the Vitrification Melter design, residual radioactivity in the Melter, and WVDP waste management plans.

1.4.1 The Western New York Nuclear Service Center

The WVDP is located at the Western New York Nuclear Service Center. The center is a 3,340-acre site located approximately 30 miles south of Buffalo, New York. It is owned by the New York

[1] DOE did not receive comments from any State.

State Energy Research and Development Authority (NYSERDA) on behalf of the State of New York, the original owner. Figure 1-1 shows the location of the Center and the WVDP.

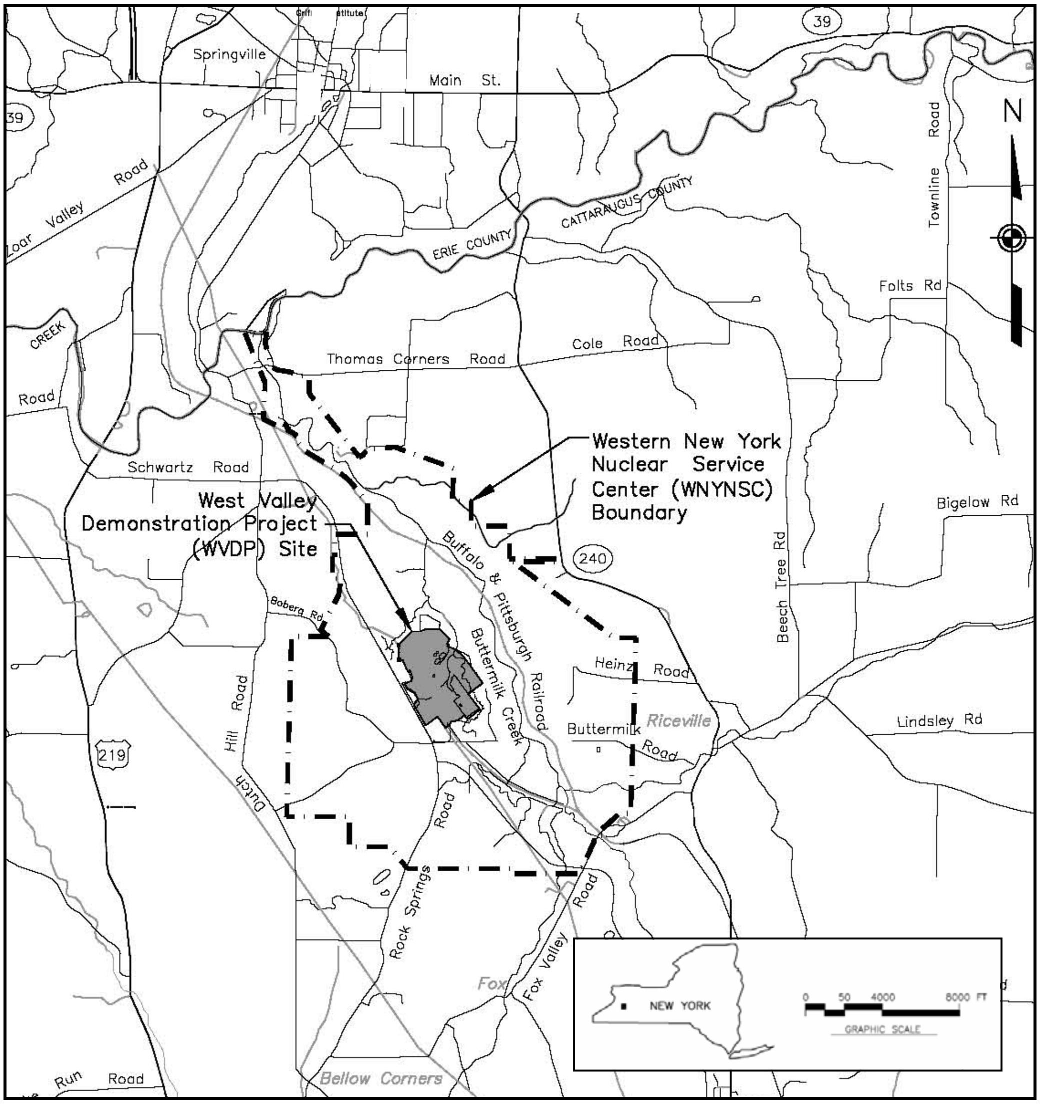


Figure 1-1. The Western New York Nuclear Service Center and the WVDP

The Center was established in the early 1960s as a nuclear industrial complex that would include spent nuclear fuel reprocessing and waste disposal facilities. The reprocessing facilities were constructed and operated by a private company, Nuclear Fuel Services, Inc. Nuclear Fuel Services also operated the two onsite radioactive waste disposal facilities – the NRC-Licensed Disposal Area (NDA) and the State-Licensed Disposal Area (SDA).

The major facilities at the Center are located within a central area of approximately 200 acres. Among these facilities are the reprocessing plant itself, also referred to as the Process Building, and four underground liquid waste storage tanks. Figure 1-2 shows the WVDP area of the Center.

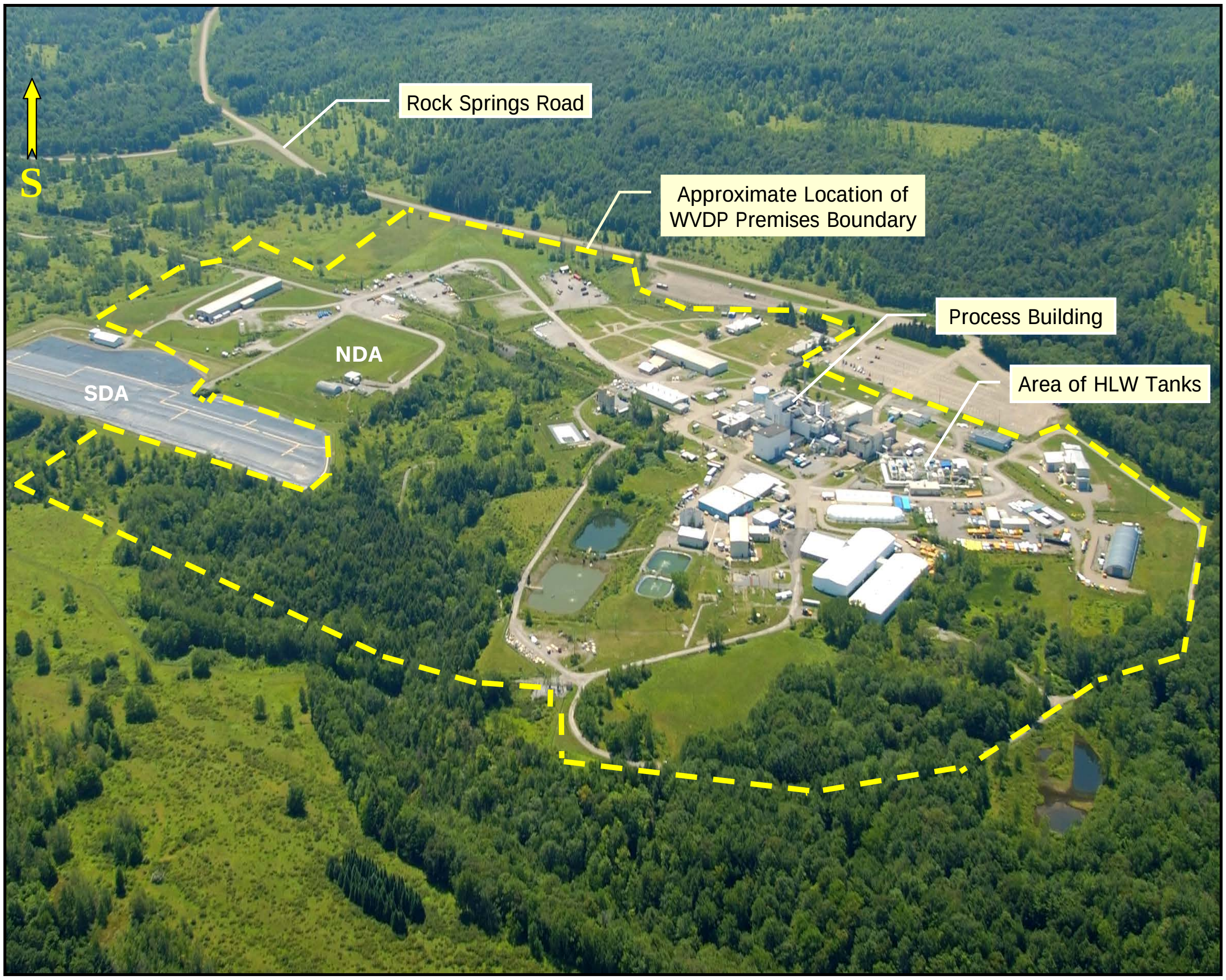


Figure 1-2. The WVDP Area of the Center in 2006 (WVDP photo)

1.4.2 Spent Nuclear Fuel Reprocessing

Reprocessing operations at the West Valley site began in 1966 and were performed under license from the U. S. Atomic Energy Commission; licensing and related regulatory functions of the Atomic Energy Commission were transferred to the NRC in 1974. During six years of operation, the plant reprocessed spent nuclear fuel recovering approximately 620 metric tons of uranium and approximately 1,926 kilograms of plutonium (DOE 1999). Nuclear Fuel Services used the PUREX (plutonium uranium extraction) chemical separations process for most of the irradiated fuel and the similar THOREX (thorium uranium extraction) process for a single fuel lot enriched in uranium and thorium.

Approximately 600,000 gallons of liquid HLW were produced during reprocessing and stored in Tanks 8D-2 and 8D-4. The approximately 560,000 gallons of neutralized PUREX waste inside Tank 8D-2 consisted of a bottom sludge layer containing insoluble hydroxides and other salts that precipitated out of solution, covered by liquid (supernatant) rich in sodium nitrate and sodium nitrite. Additionally, approximately 12,000 gallons of acidic THOREX waste commingled with recovered thorium was stored in Tank 8D-4. Tanks 8D-1 and 8D-3 served as standby spares and were not used by Nuclear Fuel Services for HLW storage, as indicated previously. (Rykken 1986)

In 1972, Nuclear Fuel Services shut the reprocessing plant down for expansion, modifications, and additions. However, reprocessing never resumed.

The HLW produced during plant operation and stored in the underground waste storage tanks contained an estimated 31 million curies of radioactivity. This estimate, adjusted for radioactive decay and in-growth to July 1987, included approximately 15 million curies of cesium 137 and its short-lived progeny barium 137m, 14.8 million curies of strontium 90 and its short-lived progeny yttrium 90, and approximately 196,000 curies of transuranic radionuclides, as well as lesser amounts of other radionuclides including but not limited to carbon 14, iron 55, cobalt 60, and nickel 63 (Rykken 1986).

1.4.3 The West Valley Demonstration Project

Federal legislation was enacted in 1980 in the form of the WVDP Act to provide for solidification of the high-level liquid radioactive waste generated by reprocessing, followed by clean-up of related areas and wastes.

In 1982, DOE assumed control, but not ownership, of a 156-acre portion of the central area of the Center in order to carry out its responsibilities under the WVDP Act. The NRC license technical specifications were effectively suspended for the duration of the DOE project.

To meet the objective of solidifying HLW at the site, the WVDP developed and built the Integrated Radwaste Treatment System and built the Vitrification Facility. The Integrated Radwaste Treatment System was used to separate the waste into high activity and low activity radioactive constituents and to solidify and store the low activity portion. Its primary component was the Supernatant Treatment System, which decontaminated solutions from the underground storage tanks through an ion exchange and removal process.

The Vitrification Facility was designed for the solidification of high-activity sludge and spent ion removal media (zeolite) generated from Supernatant Treatment System operations. The former reprocessing facilities were modified to accommodate the vitrification system and ancillary waste treatment and storage systems, and some new facilities were constructed by the WVDP for this purpose. For example, the new Supernatant Treatment System was installed by DOE adjacent to and inside of Tank 8D-1.

DOE completed vitrification of the treated HLW in 2002. Since then, the WVDP has focused on decontaminating and deactivating facilities and shipping LLW offsite. Alternatives for decommissioning of the WVDP and the rest of the Center were evaluated in an Environmental Impact Statement (DOE and NYSERDA 2010).[2]

1.4.4 Characterization of the Vitrification Melter

The Vitrification Melter has been characterized for radioactivity based on measured gamma radiation levels and sample analytical data (WVES 2010b) and found to not exceed NRC limits for Class C LLW under 10 CFR 61.55 as described in Section 6 of this evaluation. Section 2 of this evaluation provides more detail on the characterization process.

[2] The *Final Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center* (DOE and NYSERDA 2010) and the associated Record of Decision, 75 FR 20582 (April 20, 2010), were issued in 2010.

1.4.5 Incorporation into a Solid Physical Form

The Vitrification Melter is in a solid physical form. However, void spaces in the Vitrification Melter and its waste container will be filled with grout consisting of low-density cellular concrete, mainly to help stabilize the component within the shipping container during transport and to encapsulate surface contamination. This grout will not increase the waste disposal volume and was not considered in the classification of the waste.

1.4.6 Potential Waste Disposal Facilities

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 (WCS) facility in Texas for disposal.

The DOE's Nevada National Security Site maintains two separate LLW disposal facilities known as the Area 3 and the Area 5 Radioactive Waste Management Sites.

The Area 3 site, which is presently inactive, is located in the Yucca Flat area on the east side of the 1,350-square-mile Nevada National Security Site. It has three disposal units located in shallow depressions in the ground created by underground nuclear weapons detonations hundred of meters below the surface. Waste is placed in these unlined craters and covered with soil. The Area 5 site is located in the Frenchman Flat area about 12 miles south of the Area 3 site. Waste in the Area 5 site is generally disposed of in trenches approximately 22 feet deep and covered with eight feet of soil. If DOE decides to dispose of the Vitrification Melter at the Nevada National Security Site, it would be disposed of in the Area 5 site.

The commercial WCS radioactive waste disposal facility is located near Andrews, Texas on a semi-arid, isolated 1,338-acre site. It is licensed by the State of Texas[3] for near-surface disposal of Class A, B, and C LLW from Texas Compact[4] waste generators and limited quantities of nonparty compact waste imported from other states, as well as Class A, B, and C and mixed low-level Federal facility waste[5]. Federal facility waste, which includes LLW owned or generated by DOE, would be disposed of in a separate landfill disposal unit called the Federal Facility Waste Disposal Facility that is under construction and expected to open in 2012. 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.[6]

---

[3] Texas became an NRC Agreement State in 1963, and as an NRC Agreement State, regulates and licenses certain radioactive materials within its borders, including the disposal of certain LLW. The Texas program is periodically reviewed by the NRC; under the NRC Agreement State Program, NRC evaluates technical licensing and inspection issues from Agreement States, and periodically evaluates State rules for health and safety and compatibility with NRC requirements. Pursuant to applicable law, including Title 30 of the Texas Administrative Code, WCS was initially issued a license, with conditions, by the Texas Commission on Environmental Quality in 2009, which subsequently has been amended several times, for a compact waste disposal facility and a Federal waste disposal facility.

[4] The Texas compact consists of the states of Texas and Vermont. Waste generators in these states as well as generators in other states (upon approval of applicable import petitions) are authorized to dispose of LLW in the WCS Texas Compact disposal facility.

[5] The license does not authorize disposal of greater-than-Class C LLW.

[6] Texas law requires, among other things, that the licensee submit a written agreement, signed by the Secretary of Energy, to the Texas Commission on Environmental Quality, stating that the Federal government will assume all right, title, and interest in land and buildings for the disposal of Federal facility waste (Texas Administrative Code,

Offsite disposal of WVDP LLW at the Nevada National Security Site is consistent with DOE's February 25, 2000 Record of Decision for the Department of Energy's Waste Management Program: Treatment and Disposal of Low-Level Waste and Mixed Low-Level Waste; Amendment of the Record of Decision for the Nevada Test Site, 65 FR 10061, (February 25, 2000), related to the *Waste Management Programmatic Environmental Impact Statement* (DOE 1997). DOE observed in this Record of Decision that the arid Nevada Test Site (now named the Nevada National Security Site) provides environmental benefits for waste disposal, such as geology, that greatly restrict the potential for any contamination movement into groundwater.

DOE selected offsite disposal of WVDP LLW at DOE facilities or commercial facilities in its Record of Decision on the *Final West Valley Demonstration Project Waste Management Environmental Impact Statement*, DOE/EIS-0337, 70 FR 35073 (June 16, 2005). Therefore, disposal of the Vitrification Melter at either the Nevada National Security Site or the WCS facility is consistent with this Record of Decision.

In June 2006, DOE issued a Supplement Analysis (DOE 2006) to its WVDP Waste Management Environmental Impact Statement to address shipment of components from the Vitrification Facility and shipment of an increased volume of LLW. This Supplement Analysis specifically addressed the Vitrification Melter. The analysis noted that the Vitrification Melter may be shipped to one of four sites that can accept Class C LLW, including the Nevada Test Site (now called the Nevada National Security Site) and the WCS site.

As discussed in subsequent sections of this evaluation, the requirements for disposal of LLW such as the Vitrification Melter waste package at the commercial WCS facility are similar to the requirements for disposal at DOE's Nevada National Security Site. The State of Texas regulations that apply to the WCS facility mirror the NRC regulations for LLW and are comparable to DOE LLW disposal requirements. For example, the provisions concerning performance objectives, solid waste form, waste stability, and Class C concentration limits would be comparable regardless of the site selected by DOE for disposal of the waste.

DOE's decision on the disposal site to be used is not within the scope of this evaluation. The DOE decision on the facility to which the Vitrification Melter waste package will be sent will be made after DOE confers with appropriate State officials in the states where the waste package may be disposed. Prior to disposal, DOE will post notice of its decision concerning the disposal location on DOE's WVDP website (www.wv.doe.gov) and on DOE's Office of Environmental Management website (www.em.doe.gov).

1.4.7 Previous NRC Staff Review

Nuclear Regulatory Commission staff members make routine visits to the WVDP to monitor DOE activities pursuant to the Commission's review responsibilities under the WVDP Act. Two such visits in 2004 focused on vitrification equipment. During these visits, NRC staff members reviewed

---

Title 30, Part 1, §336.909). The DOE and the Texas Commission on Environmental Quality entered into a non-binding Memorandum of Agreement in January 2010, concerning the requirements in §336.909, and recognizing that DOE, in its sole discretion, will decide whether to award a contract for waste disposal to WCS and whether to dispose of LLW in the WCS Federal Facility Waste Disposal Facility. Should DOE decide to dispose of the Vitrification Melter in the WCS Federal Facility Waste Disposal Facility, such disposal would be in accordance with the license, as may be amended, and the WCS waste acceptance procedures and plans approved by the Texas Commission on Environmental Quality.

information on the characterization, classification, and packaging for the Vitrification Melter and concluded that all applicable regulatory requirements had been met (NRC 2004).

1.5 Organization of this Evaluation

Information in the remainder of this evaluation is presented as follows:

Section 2 describes the waste stored in Tanks 8D-2 and 8D-4 at the conclusion of reprocessing and the HLW pretreatment process. It also describes the Vitrification Melter, including the characterization that has been performed.

Section 3 describes DOE Manual 435.1-1 waste-incidental-to-reprocessing waste determination criteria.

Section 4 describes how key radionuclides have been removed from the Vitrification Melter to the maximum extent technically and economically practical.

Section 5 discusses how safety requirements comparable to NRC performance objectives in 10 CFR 61, Subpart C and waste acceptance criteria for the potential disposal sites (the Nevada National Security Site Area 5 Radioactive Waste Management Site or the WCS site) will be achieved.

Section 6 explains that the radionuclide concentrations in the packaged Vitrification Melter are less than Class C concentration limits, and that the Vitrification Melter will be managed in accordance with Chapter IV of DOE Manual 435.1-1.

Section 7 describes consultation with the NRC and the opportunity for public comment.

Section 8 summarizes DOE's conclusions related to the evaluation.

Section 9 identifies the references cited in the evaluation.

Appendix A provides copies of drawings for the Vitrification Melter and its shipping container.

Appendix B describes management controls used to ensure quality in this evaluation.

Appendix C discusses the comparability of DOE, NRC, and State of Texas requirements for LLW disposal.

Appendix D discusses the comparability of DOE, NRC, and State of Texas radiation dose standards.

Appendix E discusses the criteria in Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005.

2.0 BACKGROUND

Section Purpose

The purpose of this section is to provide detailed background information to support the discussions in the sections that follow.

Section Contents

This section describes nuclear fuel reprocessing, the contents of the underground waste storage tanks at the conclusion of spent fuel reprocessing, initial West Valley Demonstration Project activities, the subject Vitrification Melter equipment, radiological characterization of the equipment, and waste management plans.

Key Points

- A salt/sludge separation process was used to treat the high-level waste in Tanks 8D-2 and 8D-4 to produce a high-activity waste mixture to be stabilized by vitrification into a borosilicate glass waste form suitable for geologic disposal.
- The Melter – the last component in the vitrification system – consists of an electrically heated box structure, approximately 10 feet on each side, with a stainless steel outer shell and an interior lined with refractory materials.
- The Vitrification Melter was used from 1996 through 2002 to heat the high-activity waste mixture and glass formers to turn them into homogenized molten glass.
- In September 2002, after completion of vitrification operations, the Vitrification Melter was used to process low-activity decontamination solutions, emptied, and shut down.
- The Vitrification Melter has been characterized for residual radioactivity using measured dose rates, dose conversion factors calculated by two different geometric computer models, and radionuclide scaling factors based on glass sample analytical data and an estimate of residual radioactivity made by another method as a crosscheck, which showed the initial characterization results to be conservative.
- Based on the characterization results, the Vitrification Melter meets concentration limits for Class C low-level waste.
- The Vitrification Melter has been loaded in a custom-built shielded steel shipping container in preparation for offsite disposal.
- The Department plans to ship the packaged Vitrification Melter to an offsite waste disposal facility. For purposes of this evaluation, this facility was assumed to be either the Nevada National Security Site or the licensed WCS Federal waste facility in Texas.

2.1 Introduction

This section establishes the context for the evaluations of the Vitrification Melter that are described in Sections 4, 5, and 6 by providing the following information:

- Section 2.2 provides a brief review of nuclear fuel reprocessing, with emphasis on management of the liquid HLW stream that impacted the equipment that is the subject of this evaluation.
- Section 2.3 provides summary information on the WVDP and on preparations for waste treatment.
- Section 2.4 summarizes how the waste was pretreated and describes how the HLW was stabilized into a vitrified glass form for transport to an appropriate Federal repository for permanent disposal.
- Section 2.5 describes the Vitrification Melter, explains how it was used, and describes how it was characterized for residual radioactivity, providing information important in the evaluations described in Sections 5, 6, and 7.
- Section 2.6 describes DOE plans for disposing of the Vitrification Melter.

Note that the brief descriptions of removal of HLW from the underground storage tanks and its pretreatment and vitrification are provided here solely for information purposes; the effectiveness of removal of key radionuclides from the underground waste tanks is not being evaluated in this evaluation. Section 4 addresses removal of key radionuclides from the Vitrification Melter.

2.2 Nuclear Fuel Reprocessing

Spent nuclear fuel began arriving at the Western New York Nuclear Service Center in 1965. Reprocessing was accomplished in 27 campaigns, 11 of which involved fuel from the N-Reactor at the U.S. Atomic Energy Commission’s Hanford, Washington site. The other spent nuclear fuel came from commercial nuclear reactors. Reprocessing recovered both uranium and plutonium from the fuel, and produced the approximately 600,000 gallons of liquid HLW mentioned previously.

2.1.1 The Basic Process

Reprocessing operations were conducted in the Process Building. Figure 2-1 shows this building, the Vitrification Facility, and other nearby facilities. (The Vitrification Facility, which is described in Section 2.4.2, was built by the WVDP for solidification of the HLW.)

The first step in reprocessing operations involved disassembling fuel assemblies and chopping them into pieces. The small pieces of fuel were transported to vessels where they were dissolved in concentrated nitric acid, which transformed them into an aqueous stream containing uranium nitrate, plutonium nitrate, and fission products.

As noted in Section 1, Nuclear Fuel Services mainly used the PUREX process. This five-stage solvent extraction process used tributyl phosphate/n-dodecane solution to separate the fission products from the uranium and plutonium. (The tributyl phosphate process step generated the majority of the HLW produced in the reprocessing.)

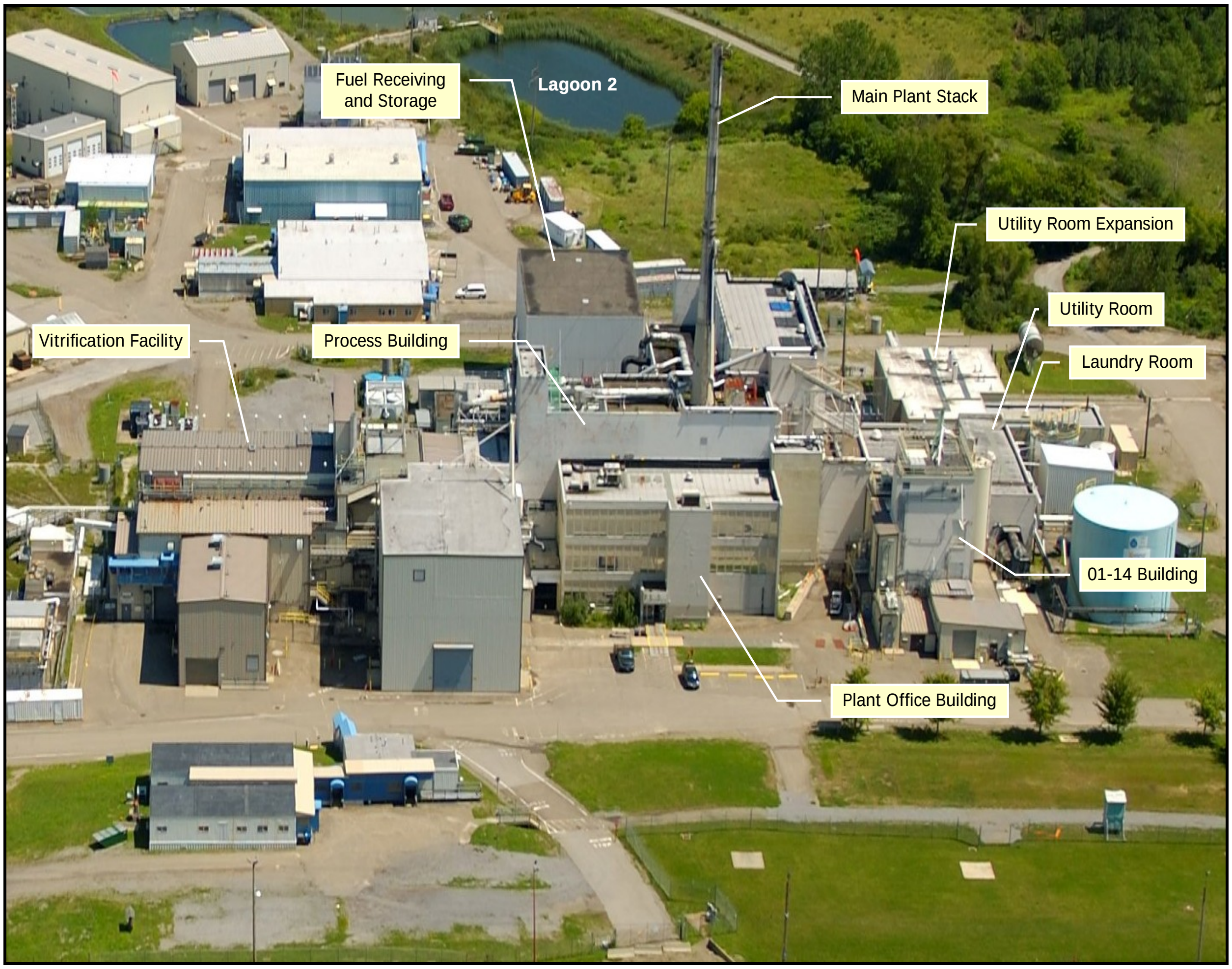


Figure 2-1. The Process Building, the Vitrification Facility, and Ancillary Facilities in 2006 (WVDP photo)

Following initial separation, the uranium-bearing and plutonium-bearing solutions underwent additional purification. The purified product solutions were then concentrated, packaged, stored, and shipped offsite. A simplified diagram representing the PUREX fuel reprocessing operation appears in Figure 2-2. (Note that the West Valley plant did not produce oxide products – UO_3 product and PuO_3 product as shown on the diagram – but rather uranyl nitrate and plutonium nitrate, materials that could be converted to the oxide products.)

2.2.2 Contents of the Waste Storage Tanks

The largest volume of waste (approximately 560,000 gallons) remaining from the normal operation of the plant in reprocessing uranium fuel was neutralized by the addition of sodium hydroxide before transfer to Tank 8D-2. 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. Therefore, the HLW was not homogeneous but was comprised of supernatant (liquid) and sludge (solids).

The approximately 12,000 gallons of acidic high-level radioactive liquid waste produced in reprocessing thorium-enriched uranium fuel using the THOREX process was stored in Tank 8D-4 without being neutralized.

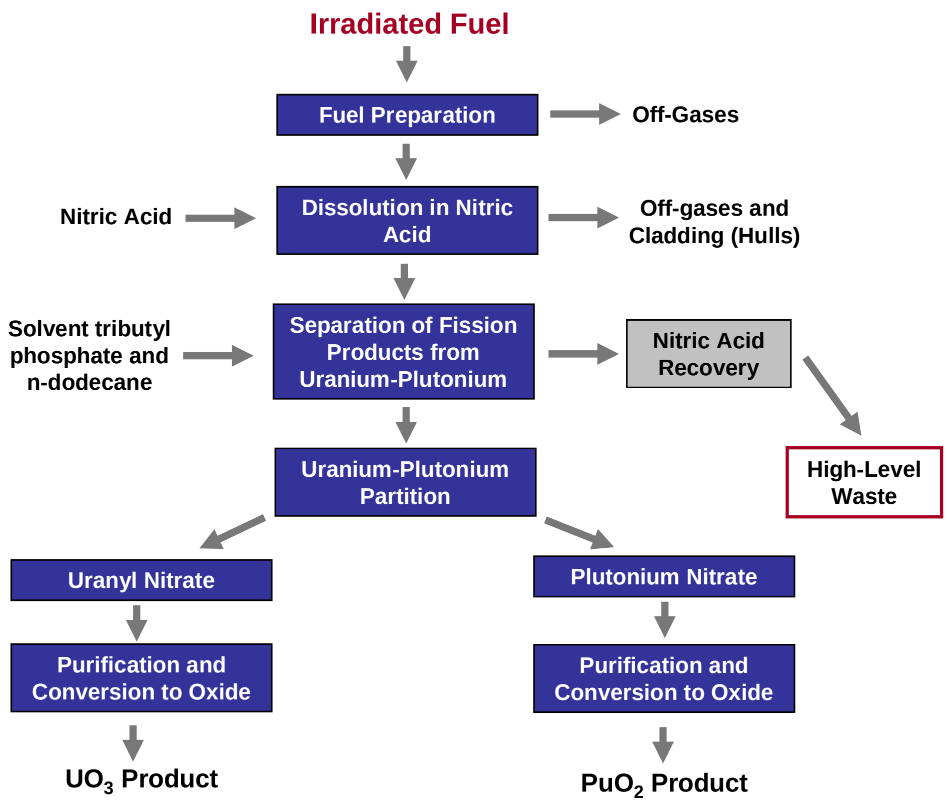


Figure 2-2. Spent Fuel Reprocessing Diagram (PUREX Process)

Table 2-1 shows the estimated radionuclide inventory in Tank 8D-2 and Tank 8D-4 at the completion of reprocessing, adjusted for decay and in-growth to July 1987. Information in this table is based on analytical data from samples collected by the WVDP in the initial project waste characterization program begun shortly after DOE assumed control of the project premises (Rykken 1986 and Eisensatt 1986).

Table 2-1. Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Fuel Reprocessing (from Eisenstatt 1986 Table 6, fission and activation products decay-corrected to July 1987)

Radionuclide	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
H-3	9.5E+01	~0	<2.0E+00	<9.7E+01
C-14	1.4E+02	~0	(1)	1.4E+02
Fe-55	(1)	1.0E+03	(1)	1.0E+03
Ni-59	(1)	8.2E+01	(1)	8.2E+01
Co-60	~0	4.7E+00	1.2E+03	1.2E+03
Ni-63	8.9E+02	6.4E+03	(1)	7.3E+03
Se-79	3.7E+01	~0	□	3.7E+01
Sr-90	2.9E+03	6.9E+06	5.0E+05	7.4E+06
Y-90(2)	2.9E+03	6.9E+06	5.0E+05	7.4E+06
Zr-93	(1)	2.3E+02	(1)	2.3E+02
Nb-93m	(1)	2.3E+02	(1)	2.3E+02

Table 2-1. Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Fuel Reprocessing (Continued) (from Eisenstatt 1986 Table 6, fission and activation products decay-corrected to July 1987) (continued)

Radionuclide	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
Tc-99	1.6E+03	(1)	8.0E+01	1.7E+03
Ru-106	(1)	1.3E+02	<3.1E-01	1.3E+02
Rh-106	(1)	1.3E+02	<3.1E-01	1.3E+02
Pd-107	(1)	1.2E+00	(1)	1.2E+00
Sb-125	4.8E+01	4.5E+03	(1)	4.5E+03
Te-125m	1.1E+01	1.0E+03	(1)	1.0E+03
Sn-126	(1)	4.0E+01	(1)	4.0E+01
Sb-126m	(1)	4.0E+01	(1)	4.0E+01
Sb-126	(1)	5.6E+01	(1)	5.6E+01
I-129	2.1E-01	(1)	<1.5E-01	<3.6E-01
Cs-134	1.4E+04	(1)	2.9E+02	1.4E+04
Cs-135	1.6E+02	(1)	(1)	1.6E+02
Cs-137	7.3E+06	(1)	5.1E+05	7.8E+06
Ba-137m(2)	6.8E+06	(1)	4.8E+05	7.3E+06
Ce-144	2.9E-05	1.4E+01	<2.0E-02	1.4E+01
Pr-144	2.9E-05	1.4E+01	<2.0E-02	1.4E+01
Pm-147	1.7E+02	3.1E+05	4.5E+03	3.1E+05
Sm-151	1.1E+00	2.1E+05	1.5E+01	2.1E+05
Eu-152	4.2E-02	4.2E+02	5.8E+00	4.3E+02
Eu-154	1.4E+01	1.3E+05	2.6E+03	1.3E+05
Eu-155	2.3E+00	2.3E+04	3.1E+02	2.3E+04
Th-232	(1)	(1)	1.6E+00	1.6E+00
U-233	4.9E-01	6.9E+00	2.6E+00	1.0E+01
U-234	2.9E-01	4.0E+00	3.0E-01	4.6E+00
U-235	6.4E-03	8.9E-02	4.9E-03	1.0E-01
U-236	1.9E-02	2.7E-01	1.0E-02	3.0E-01
U-238	5.7E-02	7.9E-01	6.1E-04	8.5E-01
Np-237	(1)	1.1E+01	(1)	1.1E+01
Np-239	(1)	2.4E+03	(1)	2.4E+03
Pu-238	1.3E+02	6.5E+03	5.3E+02	7.2E+03
Pu-239	2.5E+01	1.7E+03	1.7E+01	1.7E+03
Pu-240	1.9E+01	1.3E+03	9.0E+00	1.3E+03
Pu-241	1.5E+03	8.5E+04	9.3E+02	8.7E+04
Pu-242	2.5E-02	1.7E+00	1.3E-02	1.7E+00
Am-241	(1)	7.2E+04	2.7E+02	7.2E+04
Am-242	(1)	2.1E+01	(1)	2.1E+01
Am-242m	(1)	2.1E+01	(1)	2.1E+01

Table 2-1. Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Fuel Reprocessing (Continued) (from Eisenstatt 1986 Table 6, fission and activation products decay-corrected to July 1987) (continued)

Radionuclide	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
Am-243	(1)	2.4E+03	8.8E+00	2.4E+03
Cm-242	(1)	2.2E+00	<1.1E-03	2.2E+00
Cm-243	(1)	1.7E+02	5.0E-02	1.7E+02
Cm-244	(1)	2.2E+04	1.6E+01	2.2E+04
Cm-245	(1)	1.0E+01	1.2E-03	1.0E+01
Cm-246	(1)	4.3E+00	(1)	4.3E+00

NOTES: (1) Not present or undetectable.

(2) The progeny of Sr-90 (Y-90) and Cs-137 (Ba-137m) are included here because they were reported in Table 6 of Eisenstatt 1986.

2.3 The Beginning of the West Valley Demonstration Project

This brief summary begins with a discussion of key points in the WVDP Act, then summarizes the WVDP preparations for waste treatment.

2.3.1 The WVDP Act

After conducting studies and hearings related to dealing with the radioactivity from reprocessing activities that remained at the West Valley facility, the U.S. Congress enacted the WVDP Act. The WVDP Act directed the DOE to carry out the following activities:

(1) Solidify the HLW;

(2) Develop containers suitable for permanent disposal of the solidified HLW waste;

(3) Transport the waste to an appropriate Federal repository for permanent disposal;

(4) Dispose of LLW and transuranic waste produced in the solidification of the HLW; and

(5) Decontaminate and decommission the tanks, facilities, materials, and hardware used in the project in accordance with requirements prescribed by the NRC.

The Act directed DOE to enter into a cooperative agreement with the State (NYSERDA) for the State to make available the facilities and HLW necessary to carry out the project, without transfer of title, with DOE providing technical assistance in securing required license amendments. This cooperative agreement became effective on October 1, 1980 (DOE and NYSERDA 1981).

The Act also directed DOE to enter into an agreement with the NRC for informal review and consultation on the project by NRC and to afford NRC access to the site to monitor activities under the project to assist DOE in protecting health and safety. This agreement was formalized in a memorandum of understanding signed in September 1981 (DOE and NRC 1981).

In addition, pursuant to the WVDP Act, applicable decontamination and decommissioning activities shall be in accordance with such requirements as the NRC may prescribe. The review and consultation by the NRC shall not include or require formal procedures or activities by the NRC pursuant to the Atomic Energy Act or any other law.

In accordance with the WVDP Act, and under the cooperative agreement with DOE, NYSERDA made available to DOE, without transfer of title, the 156.4-acre area known as the Project

Premises[7]. DOE assumed operational responsibility for this area in February 1982 and employed the West Valley Nuclear Services Company (WVNSCO) as the managing and operating contractor for the WVDP[8].

2.3.2 Waste Treatment Preparations

To manage the HLW, DOE selected onsite processing using a salt/sludge[9] separation process (47 FR 40705 (September 15, 1982)). This approach involved use of a chemical pretreatment method to:

(1) Separate the major radioactive species (i.e., cesium 137) from the liquids held in Tanks 8D-2 and 8D-4,

(2) Combine the separated cesium 137 with the sludge to produce a high activity waste mixture, and

(3) Vitrify the resulting high activity waste mixture into an approved glass waste form.

DOE refined the approach as details of plans for separation of the waste streams and vitrification of the HLW were developed. Preparations for these activities included:

- Constructing the Supernatant Treatment System, including a new building to house a valve aisle and other equipment;
- Constructing the HLW Transfer Trench and associated piping to transport waste from the underground tanks to the Vitrification Facility;
- Modifying Tank 8D-1 for use as a treatment tank, including installation of ion exchange[10] columns containing zeolite;
- Adapting existing tanks within the Process Building for use with the Integrated Radwaste Treatment System;
- Removing equipment from other areas of the Process Building to provide room for Integrated Radwaste Treatment System equipment;
- Removing equipment from the Chemical Process Cell in the Process Building and setting up this shielded area for interim storage of the HLW canisters;
- Installing the waste mobilization pumps, waste transfer pumps, and other necessary hardware in Tanks 8D-1, 8D-2, and 8D-4 to facilitate removal of tank contents; and
- Constructing the Vitrification Facility and installing the vitrification equipment, including the Melter.

---

[7] Two other small parcels of land were transferred to DOE in 1986, bringing the actual total to approximately 167 acres. The Project Premises is commonly referred to as being 200 acres in size.

[8] In October 2007, West Valley Environmental Services LLC (WVES) superseded WVNSCO as DOE's site contractor. In September 2011, CH2MHill-B&W West Valley LLC became the site contractor for Phase 1 decommissioning and facility disposition activities.

[9] Salt in this context means the liquid portion of the stored waste, i.e., the supernatant.

[10] Although the zeolite-loaded columns are commonly referred to as ion exchange columns, the zeolite actually functions as a molecular sieve characterized by pores and crystalline cavities of uniform dimensions that adsorb certain molecules.

2.4 HLW Processing

Processing of HLW involved two major programs: pretreatment, followed by vitrification.

2.4.1 Pretreatment of the Waste

The pretreatment program consisted of four major tasks: (1) supernatant processing, (2) PUREX sludge washing, (3) PUREX/THOREX sludge washing, and (4) zeolite transfer to Tank 8D-2. DOE consulted with NRC on the treatment processes, consistent with provisions of the DOE/NRC Memorandum of Understanding.

The major steps involved: (1) decontaminating PUREX supernatant from Tank 8D-2 in the Supernatant Treatment System columns inside Tank 8D-1, (2) transferring the decontaminated liquid to the Liquid Waste Treatment System evaporator, and (3) transferring the evaporator concentrates to the Cement Solidification System set up in the 01-14 Building, where they were solidified in cement in 71-gallon steel drums.

The Integrated Radwaste Treatment System was operated from May 1988 until November 1990, pretreating approximately 600,000 gallons of PUREX supernatant. Cesium 137 was removed from this liquid at a decontamination effectiveness of greater than 99.99 percent and adsorbed on zeolite, which was stored under liquid in Tank 8D-1. Some Pu removal was also accomplished (Kelly and Meess 1997).

The PUREX sludge in Tank 8D-2 was washed from October 1991 to January 1992. Washing consisted of adding a sodium hydroxide solution to increase the alkalinity of the liquid waste and adding additional water.

The washing process dissolved the hard layer of sludge present in the tank, solubilized the sulfate and other undissolved salts present in the sludge, and mixed the interstitial liquid trapped in the sludge with the wash solution. This sludge washing was performed in conjunction with sequential operation and lowering of the five mobilization pumps in Tank 8D-2 to thoroughly mix the contents.

A second wash of the PUREX sludge was performed from May to June 1994 to further reduce the amount of sulfates in the high activity waste prior to vitrification. As with the first sludge wash, sodium hydroxide and water were added to Tank 8D-2 while the mobilization pumps mixed the contents of the tank. Following the second wash, the wash solution was again processed through the Integrated Radwaste Treatment System from June to August of 1994.

Following the completion of sludge washing, final preparations were made to complete the installation of the HLW transfer system which links all three underground waste storage tanks that contained HLW (Tanks 8D-1, 8D-2, and 8D-4) to the Vitrification Facility using double-contained piping run in underground concrete trenches and pits. To facilitate waste removal, waste transfer pumps were installed in Tanks 8D-1, 8D-2, and 8D-4. Tank 8D-2 was prepared for the acidic THOREX addition from Tank 8D-4 during November and December 1994 by increasing its alkalinity with sodium hydroxide. The acidic THOREX was transferred from Tank 8D-4 to Tank 8D-2 and neutralized during January 1995. (Kelly and Meess 1997)

Following neutralization, sodium nitrite was added to Tank 8D-2 to minimize pitting corrosion that could result from the large amount of nitrates in the THOREX solution (Kelly and Meess 1997).

After mixing the contents of Tank 8D-2 – which included washed PUREX sludge, sludge wash liquid, THOREX precipitates, and THOREX solution – using the waste mobilization pumps, the THOREX/ PUREX wash liquid was processed through the Integrated Radwaste Treatment System.

2.4.2 Vitrification of the HLW

The Vitrification Facility was designed and used to stabilize the following waste streams in a borosilicate glass matrix: (1) the radioactive high activity sludge that had been generated during PUREX reprocessing of spent uranium fuel, (2) THOREX waste that resulted from the reprocessing of thorium-uranium fuel, and (3) contaminated cesium-loaded zeolite generated during Supernatant Treatment System operations. Figure 2-3 shows the general arrangements in the facility.

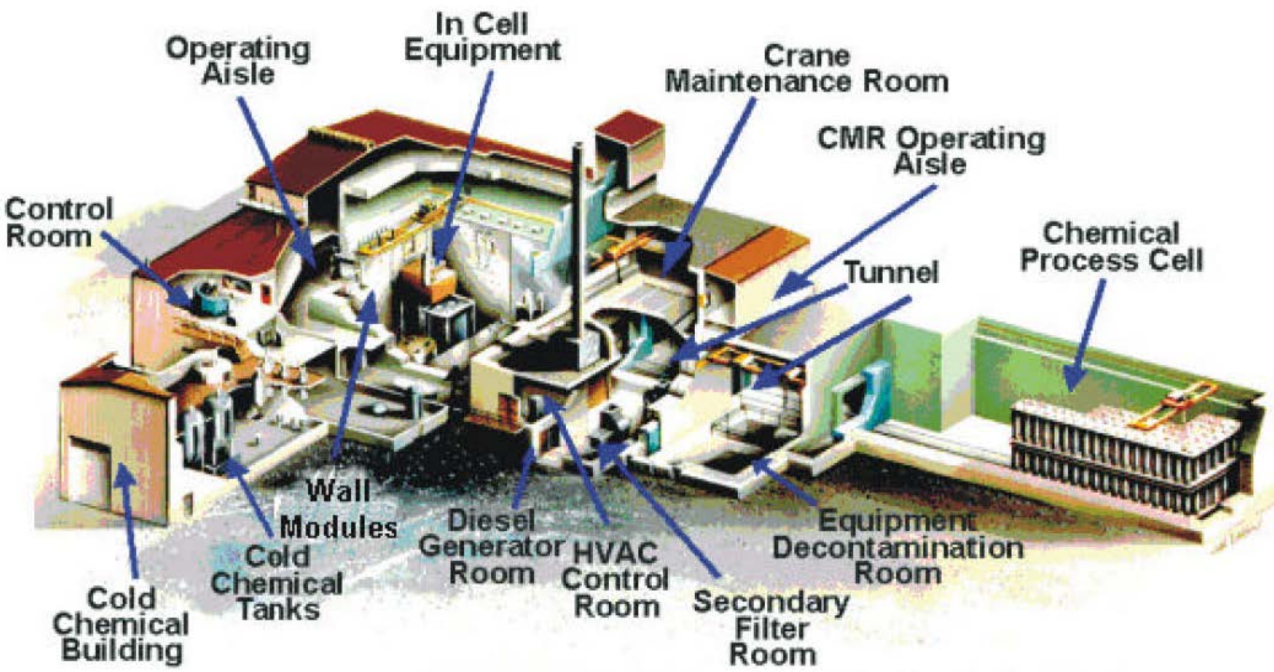


Figure 2-3. Vitrification Facility General Arrangement. *(The Equipment Decontamination Room and the Chemical Process Cell, where the HLW canisters were temporarily stored, were part of the original Process Building.)*

The Vitrification Facility building housed the Vitrification Cell, operating aisles, and a control room. The shielded Vitrification Cell contained the equipment used to concentrate the high activity waste slurry, mix it with glass formers (oxide additives), melt this mixture to form borosilicate glass, pour the molten glass into the stainless steel canisters, seal the canisters, and decontaminate the canister exteriors. Among this equipment were the Concentrator Feed Makeup Tank, the Melter Feed Hold Tank, and the Vitrification Melter.

Figure 2-4 illustrates the general process flow, and shows the location of the Melter in the vitrification process.

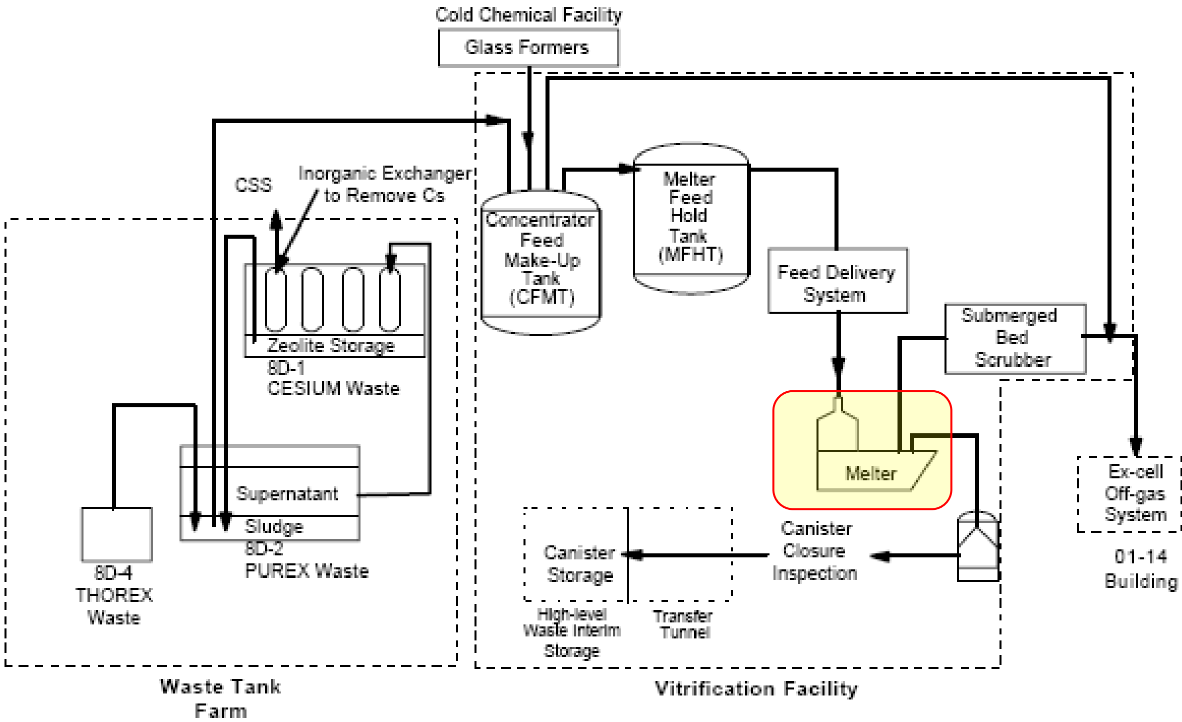


Figure 2-4. Vitrification Process Flow Diagram (for information, not to scale)

Between 1996 and 2002, the WVDP retrieved the high activity waste from the tanks and stabilized it by vitrification. Deactivation of the Vitrification Facility, which included removal of all of the process equipment, was completed in July 2005. The WVDP is currently focusing on facility decontamination and deactivation, waste management, and plans for decommissioning. DOE plans for waste shipment are summarized in Section 2.6 below.

2.5 Vitrification Melter Description, Operation, and Characterization

This subsection describes the Vitrification Melter, summarizes the radiological characterization process, and provides the characterization results.

2.5.1 Vitrification Melter Description

Also known as the Slurry-Fed Ceramic Melter, the Vitrification Melter consists of an electrically heated box structure approximately 10 feet on each side. The outer shell is formed of stainless steel. The interior is lined with a composite of various refractory materials[11] to with-stand high temperatures. Figures 2-5 shows the Melter general assembly.

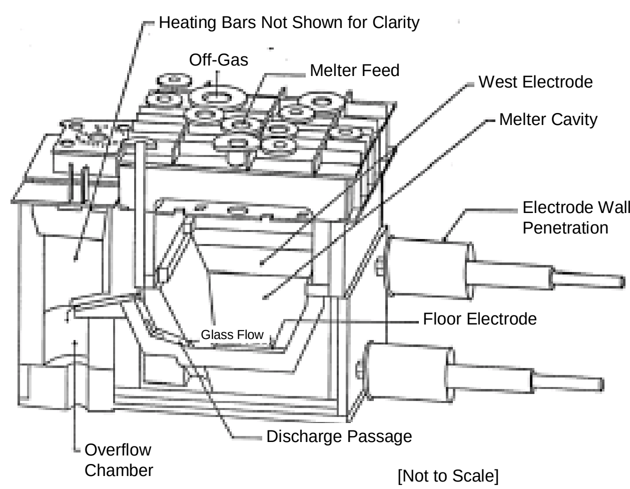


Figure 2-5. Vitrification Melter Design Features

[11] The surface in contact with the molten glass is composed of Monofrax™ K-3. This material is a chrome alumina fusion cast refractory with high corrosion resistance used in applications such as fiberglass furnaces and nuclear encapsulation melters (RHI 2011).

The sides and bottom of the outer shell are covered with a cooling water jacket. The Melter weighs approximately 106,000 pounds (Vance, et al. 1997). Figure 2-6 show the installed Melter.[12] Appendix A provides copies of Melter drawings.

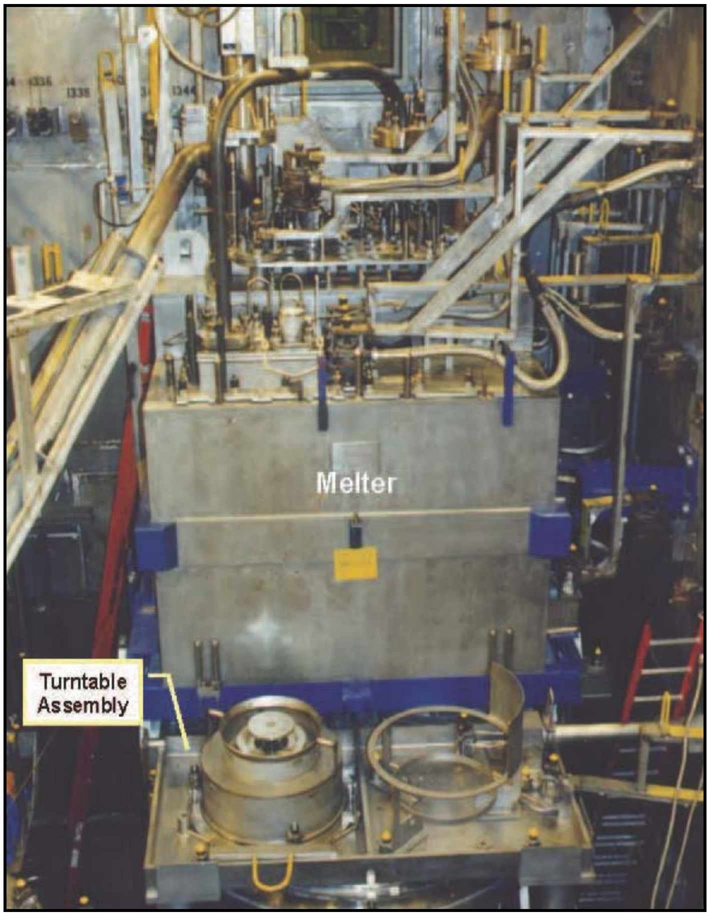


Figure 2-6. Melter Installed in Vitrification Cell
WVDP photo)

The Vitrification Melter is divided into two sections. The main section contains the Melter cavity, which has an overall height of approximately 4.5 feet. The upper part of the cavity is rectangular in shape, with the lower part in the form of an inverted truncated rectangular pyramid. During normal operation, the Vitrification Melter would accommodate 227 gallons (approximately 30 cubic feet) of slurry. The slurry was heated with three electrodes, one of which served as the floor of the vessel.

The discharge section of the Vitrification Melter contains a primary and a secondary pour chamber, each with spouts and silicone carbide radiant heaters. The Vitrification Melter was mounted on a track system in the northeast corner of the Vitrification Cell.

2.5.2 Operational History

The Melter was installed by the WVDP along with the other vitrification equipment and used throughout vitrification operations. During operation, batches of HLW slurry feed material were transferred from the Melter Feed Hold Tank to the Melter. Inside the Vitrification Melter, calcined wastes and glass formers were melted and fused into a glass pool where they homogenized.

Homogenized molten glass in the Vitrification Melter was periodically transferred into a stainless steel canister held in position beneath the Melter by the canister turntable. After the glass was allowed to cool, the loaded canister was transferred from the turntable to the weld station, where it was sealed with a welded lid. Figure 2-7 illustrates how the Vitrification Melter was positioned over the turntable to fill a HLW canister.

[12] Note that the unit contained three external electrode assembles – rather than two as indicated in Figure 2-5 – which were removed flush with the side of the outer box surface before the unit was placed into its shipping container, which is shown in Figure 2-8 below.

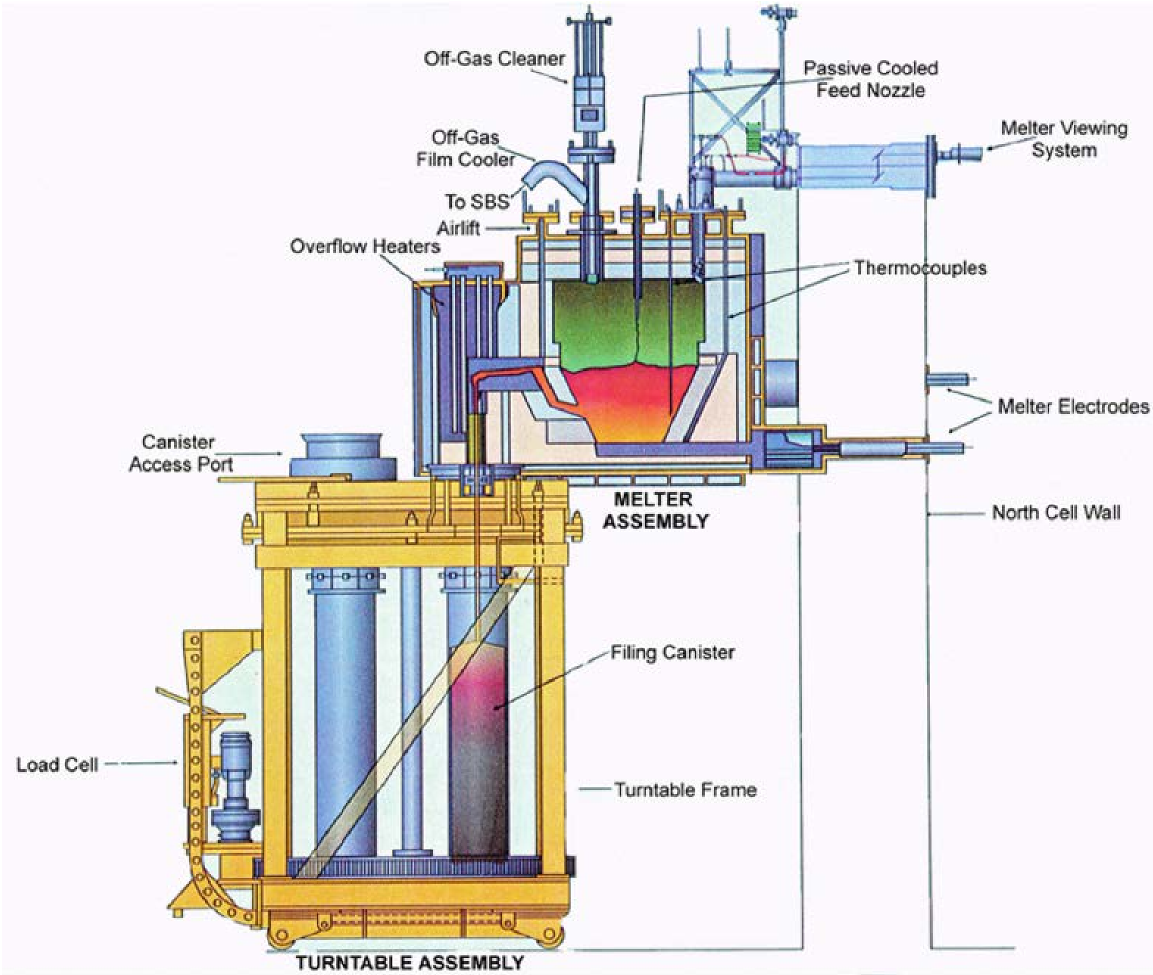


Figure 2-7. Melter and Turntable Assemblies During Vitrification.

During the glass melting process, volatile elements evaporating from the glass pool were vented to the off-gas treatment system, along with feed particles entrained in the process off-gas.

The silicon carbide heaters used in the discharge section of the Vitrification Melter were expected to have limited service life based on system testing, and two heater assemblies failed during use. Another operating problem was encountered when the glass exit port plugged with glass near the end of vitrification operations. This condition was not unexpected and this possibility led to the twin pour chamber design. The secondary pour chamber was brought on line after this event. (Petkus, et al. 2003)

In September 2002, after completion of HLW vitrification, the Vitrification Melter was used to process decontamination solutions, emptied using two evacuated canisters, and shut down, as described in Section 4. Based on recorded data, approximately 2200 kg of molten residual glass – about 88 percent of the estimated amount present – were removed from the Vitrification Melter during this process.

The residual material which could not be removed by these processes – approximately 12 percent of the original amount – consists of the glass in the plugged discharge port and in the bottom of the Melter cavity, along with a thin layer on the sides of the Melter cavity. It was estimated that approximately 300 kg of glass remained in the Melter cavity, a heel of approximately eight inches. This estimate was based on a combination of level detector responses, preliminary canister weights, and thermocouple responses during the evacuation evolution. Visual observation indicated that the residual glass was present in portions of the Melter at the 10-inch level.[13] (Lachapelle 2003)

Section 4 discusses the efficiency of the decontamination processes.

In 2004, the Vitrification Melter was removed from the Vitrification Cell and loaded into its shipping container, which is shown in Figure 2-8. Container drawings appear in Appendix A.

This Industrial Package (IP)-2 type shipping container is formed of steel that is six inches thick on the sides and four inches thick on the top and bottom. The empty container weighs approximately 210,000 pounds. The total package weight with the Vitrification Melter and low-density cellular concrete, which will be used to fill both the Melter cavity and the container, will be approximately 360,000 pounds. (WMG 2004b)

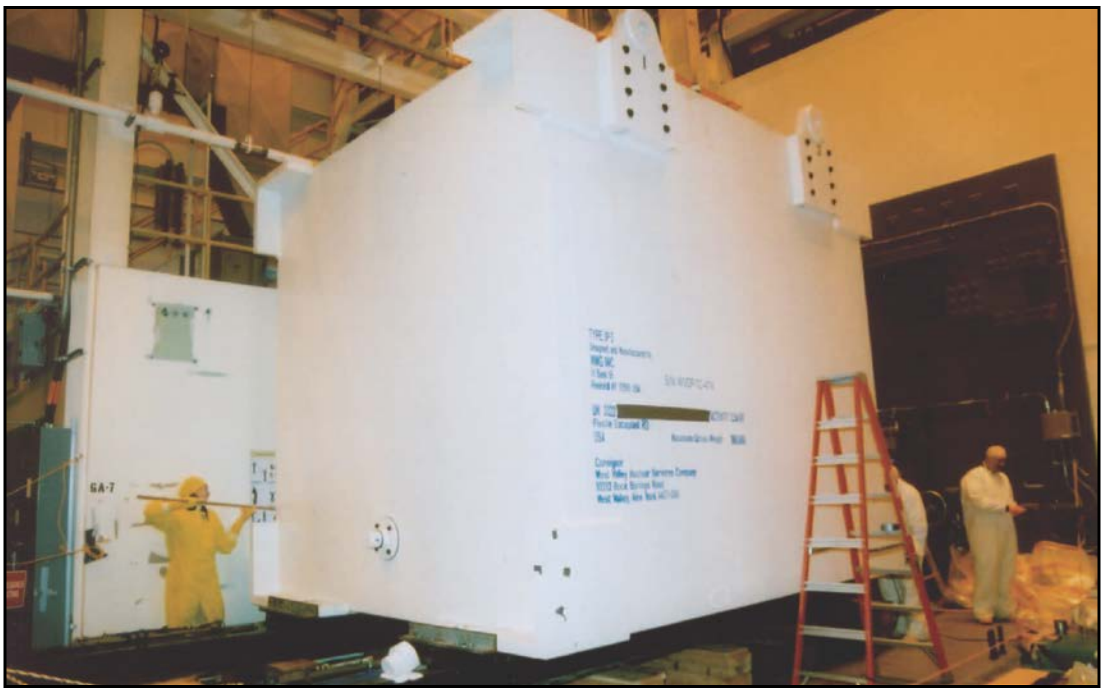


Figure 2-8. Vitrification Melter Shipping Container *(WVDP photo)*

2.5.3 CHARACTERIZATION

Details of the Vitrification Melter characterization appear in the characterization report (WMG 2004a). The characterization process took into account three different sources of residual radioactivity:

- The Vitrification Melter cavity;
- The plugged discharge port, which is separated from the cavity by 18 inches or more of refractory material; and
- The exterior of the Vitrification Melter.

[13] An assessment made using actual vitrification program data showed the eight-inch estimate to be conservative as discussed below.

Data Used in Characterization

The characterization for residual activity inside the Vitrification Melter was based on measured gamma dose rates and analytical data from two samples of residual glass.

For the Vitrification Melter cavity, gamma dose rates were measured in January 2003 with an unshielded Ludlum Model 133-7 Geiger-Mueller detector probe lowered into nozzles located above the residual glass. These dose rates ranged from 330 R/h to >1,000 R/h, with levels of 748 and 749 R/h measured through a nozzle located directly above the residual glass. A level of 2.1 R/h measured one foot above the lid of the Vitrification Melter assembly was also used in the calculations. A dose rate of 40 R/h measured at the southernmost lip of the west discharge cavity was used in calculations for the discharge cavity source. (Lachapelle 2003)

The two glass samples were obtained from the evacuated canisters. They were analyzed in the WVDP Analytical and Process Chemistry Laboratory. To promote accuracy, three separate analyses (the original and two replicates) were performed on material from each canister.

Copies of both the radiation survey records and the laboratory analysis reports appear in the characterization report (WMG 2004a).

The exterior surfaces of the Vitrification Melter were characterized based on measured surface contamination levels. The maximum beta-gamma contamination level on portions of the surface impacted by accidental spills of slurry was determined to be 16.7 $\mu Ci/cm^2$; the maximum alpha surface activity from airborne contamination was determined to be 0.0443 $\mu Ci/cm^2$ (WMG 2004a). Both types of contamination are now considered to be fixed because Polymeric Barrier System, a latex fixative coating, was applied to the outside of the Vitrification Melter before it was placed in the shipping container.

Models Used in Characterization of the Vitrification Melter Interior

Two geometry models were used to calculate dose conversion factors, which were combined with measured dose rates to estimate residual Cs-137 activity in the two areas of the Vitrification Melter.

A model prepared using the QAD-CGGP-A computer code[14] was used to represent the complex geometry of the Vitrification Melter cavity. This model accounted for a layer of contamination 0.125-inch thick on the cavity surfaces to its full height of 28 inches, as well as the eight inches of solidified glass in the bottom of the cavity. The basis for the thickness of the contamination layer is previous WVDP experience in Vitrification Melter testing described in the *Slurry Fed Ceramic Melter Disassembly Report* (Brooks 1993)[15].

It was also assumed that glass had penetrated fissures in the refractory material to a depth of approximately 0.5 inch, based on experience in Vitrification Melter testing (Brooks 1993). This factor was taken into account in the model (WMG 2004a).

[14]QAD-CGGP-A is a point-kernel computer code for calculating fast neutron and gamma ray penetration through various shield configurations. It was developed by Atomic Energy of Canada, Ltd.

[15] The WVDP conducted a five-year test program producing glass from simulated waste slurry with an actual vitrification melter that was disassembled and examined after completion of the testing (Brooks 1993).

A model prepared with the Megashield™ code was used for the plugged discharge port. Solidified glass was assumed to completely fill this discharge tube and the discharge cavity itself to a level of four inches.

Assumptions Used in Characterization of the Vitrification Melter Interior

Two assumptions added conservatism to the calculations:

- It was assumed that the measured gamma radiation levels were due entirely to Cs-137, even though other gamma emitters such as Am-241 are present and contribute to the measured gamma radiation levels[16]; and
- It was assumed that radioactivity in the Vitrification Melter cavity did not contribute to the 40 R/h dose rate used to characterize radioactivity in the discharge port, although the detector was positioned such that radiation from the cavity influenced this measurement.

Results for the Vitrification Melter Interior

These calculations – which were performed using the data, models, and assumptions identified above – produced an estimate of 4062 curies of cesium 137 in the Vitrification Melter cavity and an additional 252 curies of cesium 137 in the plugged discharge tube. Scaling factors based on the glass sample analytical data were used with the RADMAN™ computer code to estimate the amounts of other radionuclides in the Vitrification Melter. Table 2-2 shows the estimates.

Table 2-2. Vitrification Melter Total Activity Estimates(1)

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
C-14	2.12E-02	U-235	3.76E-04
K-40	8.19E-02	U-238	2.25E-03
Mn-54	8.57E-02	Np-237	6.20E-03
Co-60	8.33E-02	Pu-238	6.84E-01
Sr-90	2.47E+02	Pu-239	1.59E-01
Zr-95	1.65E+00	Pu-241	3.12E+00
Tc-99	1.11E-02	Pu-242	1.12E-05
I-129	(2)	Am-241	3.00E+00
Cs-137	4.31E+03	Am-242m	9.16E-05
Eu-154	1.21E+00	Am-243	3.50E-02
Th-228	4.09E-02	Cm-242	7.33E-02
Th-229	(2)	Cm-243	1.68E-02
Th-230	3.65E-04	Cm-244	2.13E-01
Th-232	4.01E-04	Cm-245	1.55E-02
U-232	5.01E-04	Cm-246	1.77E-03
U-234	9.81E-03		

NOTES: (1) From WMG 2004a, Table 3, as of October 1, 2004, except for Pu-242, Am-242m, Cm-244, Cm-245, and Cm-246, which were estimated using the alternate process discussed below.
(2) The amounts of these radionuclides are insignificant based on sample analytical data.

[16] This approach is conservative because less than 100 percent of the gamma radiation levels used in the calculation were produced by cesium 137, a condition that results in slightly overestimating the amount of cesium 137 present.

The estimates in Table 2-2 total approximately 4570 Ci as of October 1, 2004. The estimated amounts of residual glass in the Vitrification Melter are as follows:

- Melter heel, 300 kg[17];
- On Melter cavity surfaces and in refractory material fissures, 26 kg; and
- The plug in the discharge port, 99 kg.

These amounts total approximately 425 kg. These estimates were based on the west discharge port containing a solid plug of glass, the discharge tube being completely filled, and the discharge cavity being filled to a height of four inches. (WMG 2004a)

As a crosscheck, DOE also prepared an estimate using an alternative method. This method involved multiplying the estimated mass of the residual glass (425 kg) by the radionuclide concentrations measured in the glass samples taken from the two evacuated canisters. This alternate method produced a total estimate of approximately 2,240 Ci, which demonstrated that the estimates in Table 2-2 are conservative. (DOE 2011c).

Radioactivity on the Outside Surfaces

The amount of contamination on the outside surfaces of the Vitrification Melter was estimated by using the maximum measured surface activity. The total surface area used in the calculations was 9.08E+05 cm^2 (approximately 980 ft^2). The slurry contamination level of 16 $\mu Ci/cm^2$ was used for 10 percent of the surface area and the maximum measured removable activity of 1.0E+08 dpm/100 cm^2 was used for the rest of the surface area, with the slurry and airborne contamination radionuclide distributions used in the calculations. The calculations yielded a total estimate of 5.2 Ci as of October 1, 2004, with approximately 4.1 Ci of this amount associated with Cs-137. This activity represents approximately 0.11 percent of the total estimated residual radioactivity associated with the Vitrification Melter. (WMG 2004a)

NRC Staff Monitoring Visit Assessment

As explained in Section 1.4.7, NRC representatives concluded in an assessment performed in connection with two 2004 monitoring visits that the Vitrification Melter was appropriately characterized, packaged, and prepared for offsite disposal in accordance with regulatory requirements. The purpose of these visits was to evaluate preparations for packaging and disposal of three vitrification process components, including the Melter. (NRC 2004)

The NRC representatives evaluated the characterization and waste profile methodologies, the design and fabrication of the waste packages, and verification that the packages were prepared for shipment and disposal in accordance with applicable requirements. They reviewed the characterization data, the methods used to determine activity amounts, and the sample analytical data used to develop radionuclide scaling factors, and interviewed cognizant site personnel. (NRC 2004)

[17]The 300 kg estimate was based on eight inches of glass remaining in the Vitrification Melter cavity. An independent assessment was performed to help establish the validity of the eight-inch estimate. The assessment utilized recorded data from vitrification records and drawing dimensions to determine the amount of glass removed by the two evacuated canisters and compared this amount to the amount of molten glass in the cavity before the first of the evacuated canisters was used. This calculation showed a residual glass level of about 6.5 inches. This level equates to approximately 4,500 cubic inches of glass remaining in the cavity, or approximately 177 kg of glass at a density of 2.4 g/cm^3. This result shows the 300 kg estimate to be conservative. (DOE 2011c)

The NRC conclusion about characterization described in Section 1.4.7 independently confirms the validity of the characterization process used by DOE for the Vitrification Melter.

2.6 WVDP Waste Management Plans

This section briefly summarizes DOE plans for managing WVDP LLW, including the Vitrification Melter.

The Department evaluated management of radioactive waste at West Valley in its WVDP Waste Management Environmental Impact Statement (DOE 2003). In its Record of Decision, 70 FR 35073 (June 16, 2005), DOE decided that, for WVDP LLW and mixed LLW that is currently in storage at the site or that will be generated at the site over the next ten years (i.e., through 2015), DOE will ship such WVDP LLW and mixed LLW offsite for disposal, in accordance with all applicable requirements, at commercial sites (such as EnergySolutions [formerly known as Envirocare], a commercial radioactive waste disposal site in Clive, Utah), one or both of two DOE sites, the Nevada Test Site [now called the Nevada National Security Site] in Mercury, Nevada, or the Hanford Site in Richland, Washington, or a combination of commercial and DOE sites. This Record of Decision included wastes that DOE may determine in the future to be LLW or mixed LLW pursuant to a waste-incidental-to-reprocessing determination using the evaluation process (this evaluation, for example).

As noted previously, in June 2006, DOE issued a Supplement Analysis (DOE 2006) to its WVDP Waste Management Environmental Impact Statement to address shipment of components from the Vitrification Facility and shipment of an increased volume of LLW. This Supplement Analysis specifically addressed the Vitrification Melter. The analysis noted that the Vitrification Melter may be shipped to one of four sites that can accept Class C LLW, including the Nevada Test Site (now called the Nevada National Security Site) and the WCS site.

In 2001, after completing the required approval process, the WVDP received approval to ship LLW to the Nevada Test Site (now called the Nevada National Security Site) and has been shipping LLW to that facility since that time (CHBWV 2011). DOE plans to ship the Vitrification Melter waste package to an offsite LLW disposal facility. For the purposes of this evaluation, this facility is assumed to be either the Nevada National Security Site or the WCS disposal facility for Federal LLW in Texas. A final decision on the facility to which the Vitrification Melter waste package will be sent will be made after DOE confers with appropriate State officials for the states where the waste package may be disposed.[18]

Section 6 demonstrates that the Vitrification Melter waste package does not exceed concentration limits for Class C LLW.

[18] DOE also will comply with the provisions in DOE Manual 435.1-1, Section I.2.F(4), concerning approval of exemptions for use of non-DOE disposal facilities, should DOE decide to dispose of the Vitrification Melter in the WCS facility.

3.0 WASTE DETERMINATION CRITERIA

Section Purpose

The purpose of this section is to describe the criteria applicable to this waste- incidental-to-reprocessing evaluation.

Section Contents

This section provides brief background information on Department of Energy criteria that apply to this waste-incidental-to-reprocessing evaluation that have been considered and then describes the Department's criteria that apply to management of the Vitrification Melter.

Key Points

- Applicable criteria appear in Department of Energy Manual 435.1-1, *Radioactive Waste Management Manual*.

3.1 Waste Determination Criteria Background

The WVDP is required to comply with two separate and distinct sets of criteria to determine whether waste from reprocessing is incidental to reprocessing, is not HLW and may be managed as other than HLW through a demonstration of compliance with the appropriate waste determination criteria:

- DOE Manual 435.1-1, *Radioactive Waste Management Manual*, applies to WVDP wastes that DOE disposes of offsite.
- The NRC's *Final Policy Statement on Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site* (NRC 2002) describes criteria for classification of "any residual wastes present after cleaning of the high-level radioactive waste (HLW) tanks at West Valley."

Because the NRC West Valley decommissioning criteria policy statement (NRC 2002) does not apply to waste shipped offsite for disposal, as explained in Section 1.3.1, this evaluation for the Vitrification Melter was performed in accordance with DOE Manual 435.1-1. DOE's waste determination criteria are described in Section 3.2.[19]

3.2 Applicable Waste Determination Criteria

Section I.1.C of DOE Manual 435.1-1 provides that all radioactive waste subject to DOE Order 435.1 be managed as HLW, transuranic waste, LLW, or mixed LLW. DOE Manual 435.1-1, Section

[19] Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 sets forth criteria which apply only to certain waste from reprocessing of spent nuclear fuel that under certain conditions can be disposed of in Idaho or South Carolina. The WVDP is not within a covered state included in the scope of Section 3116. Although these criteria do not apply to the WVDP, DOE has, for perspective and information, considered whether disposal of the Vitrification Melter would be consistent with the relevant provisions of Section 3116. This matter is addressed in detail in Appendix E of this evaluation.

II.B, also states that waste resulting from reprocessing spent nuclear fuel determined to be incidental to reprocessing is not HLW and shall be managed in accordance with the requirements for transuranic waste or LLW, as appropriate. The determination that waste is incidental to spent nuclear fuel reprocessing, and therefore not HLW, is called a "waste-incidental-to-reprocessing determination," which is also referred to in this evaluation as a waste determination.

DOE Manual 435.1-1, Section II.B.2(a), lists three criteria to be used to demonstrate, using the evaluation method, that wastes resulting from spent nuclear fuel reprocessing are not HLW and should be managed as LLW:

(1) Criterion 1 – the wastes have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical;

(2) Criterion 2 – the wastes 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) Criterion 3 – 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 DOE Manual 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 low-level waste as set out in 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize.[20]

As will be demonstrated in the next three sections of this evaluation, DOE has evaluated the Vitrification Melter against these criteria, and, for the reasons presented, the evaluation shows that the Vitrification Melter meets the applicable criteria and can be managed and disposed of as LLW.

[20] DOE did not authorize alternative requirements for the Vitrification Melter.

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

Section Purpose

The purpose of this section is to evaluate whether the waste (i.e., the Vitrification Melter) has been processed to remove key radionuclides to the maximum extent that is technically and economically practical.

Section Contents

This section describes the process used in determining the key radionuclides in the Vitrification Melter and identifies those radionuclides. It then describes the technical and economic practicality evaluations that have been performed and their results.

Key Points

- The evaluations show that key radionuclides have been removed from the Vitrification Melter to the maximum extent that is technically and economically practical.
- The key radionuclides in the Vitrification Melter are those long-lived and short-lived radionuclides listed in Tables 1 and 2 of the Nuclear Regulatory Commission's regulations in 10 CFR 61.55, three of which are important to the performance assessment of the WCS low-level waste disposal facility, along with four other radionuclides that are important to the results of the performance assessment of the Nevada National Security Site low-level waste disposal facility.
- Evaluation of representative potential methods of removing key radionuclides showed that processing decontamination solutions in the Vitrification Melter, using the Evacuated Canister System, and dismantling the Melter were the only methods technically practical.
- Processing of decontamination solutions and the Evacuated Canister System were used and proved to be effective in removing key radionuclides.
- The economic practicality assessment evaluated additional decontamination solution processing while the vitrification process was still operational and concluded that this would not have been economically practical.
- This assessment also evaluated Vitrification Melter dismantlement and concluded that this approach also would not have been economically practical because of increased worker radiation dose and other factors.
- The economic practicality assessment compared the impacts of processing additional decontamination solutions and Vitrification Melter dismantlement with the potential benefits.
- This assessment demonstrated that further efforts to remove key radionuclides would have substantially increased costs and produced negligible benefits, if any, in terms of improved worker or public health and safety.

The first criterion of DOE Manual 435.1-1, Section II.B.2(a) is evaluated in this section. It reads:

"[The subject wastes] have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical."

4.1 Key Radionuclides

This section begins with a brief introduction that describes the various factors considered, provides additional information on these factors, discusses their relevance to key radionuclide selection, and concludes with the identification of key radionuclides for this evaluation.

4.1.1 Introduction

The key radionuclides in this evaluation are based on consideration of the following information:

- Guidance in DOE Guide 435.1-1 on identification of key radionuclides;
- NRC requirements for classification of radioactive waste for near-surface disposal that appear in 10 CFR 61.55;
- Radionuclides known to be present in the West Valley HLW;
- The relationship between DOE disposal site waste acceptance criteria and the performance of DOE LLW disposal sites in meeting objectives for protecting individuals and the environment;
- The radionuclides of importance in the performance assessment of the Nevada National Security Site Area 5 LLW disposal area, although such consideration is not required by DOE Manual 435-1 or DOE Guide 435.1-1;
- The State of Texas requirements for classification of radioactive waste in the Texas Administrative Code, which mirror the NRC requirements in 10 CFR 61.55[21];
- Radionuclides specifically limited in the WCS radioactive material license; and
- Radionuclides important to meeting the State of Texas performance objectives – which mirror the NRC performance objectives in 10 CFR Part 61, Subpart C – based on the radionuclides of importance in the performance assessment of the WCS LLW disposal facility.

Consideration of this information will ensure that those radionuclides in the Vitrification Melter that could contribute significantly to radiological risks to workers, the public, and the environment are identified and taken into account.

4.1.2 DOE Guidance on Key Radionuclides

DOE guidance on selection of key radionuclides is provided in Section II.B of DOE Guide 435.1-1, with the applicable portion reading as follows:

"... it is generally understood that [the term] key radionuclides applies to those radionuclides that are controlled by concentration limits in 10 CFR 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

[21] The Texas requirements, license limits, and performance assessment information are considered for completeness and additional information, although such consideration is not specifically required by DOE Manual 435-1 or DOE Guide 435.1-1.

those that are important to satisfying the performance objectives of 10 CFR Part 61, Subpart C [for near-surface radioactive waste disposal facilities]."

This guidance considers both the waste classification requirements in 10 CFR 61.55[22] for radioactive waste destined for near-surface disposal and achieving the waste disposal site performance objectives.

4.1.3 Requirements of 10 CFR 61.55

The radionuclides listed in the guidance found in DOE Guide 435.1 appear in 10 CFR 61.55 in the form of two tables, which are reproduced here as follows.

Table 4-1. 10 CFR 61.55, Table 1 (Long-Lived Radionuclides)

Radionuclides	Concentration (Ci/m^3)
C–14	8
C–14 in activated metal	80
Ni–59 in activated metal	220
Nb–94 in activated metal	0.2
Tc–99	3
I–129	0.08
Alpha Emitting Transuranic (TRU) nuclides with half-life greater than 5 years	100(1)
Pu–241	3,500(1)
Cm–242	20,000(1)

NOTES: (1) These values are in units of nanocuries per gram.

Table 4-2. 10 CFR 61.55, Table 2 (Short-Lived Radionuclides)

Radionuclides	Concentration (Ci/m^3)		
	Column 1 [Class A]	Column 2 [Class B]	Column 3 [Class C]
Total of all nuclides with less than 5 y half-life	700	(1)	(1)
H–3	40	(1)	(1)
Co–60	700	(1)	(1)
Ni–63	3.5	70	700
Ni–63 in activated metal	35	700	7,000
Sr–90	0.04	150	7,000
Cs–137	1	44	4,600

NOTE: (1) There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in the table determine the waste to be Class C independent of these nuclides.

The concentrations given in these tables are used for waste classification purposes. Classification is determined by concentrations of long-lived radionuclides, by concentrations of short-lived radionuclides, or by both in those cases where the waste contains both types of radionuclides. The tables in the Texas Administrative Code mirror the 10 CFR 61.55 tables in Table 4-1 and 4-2 (Rule §336.362, Appendix E, Table I and II).

[22] Title 30 of the Texas Administrative Code has similar requirements (Rule §336.362, Appendix E).

4.1.4 Radionuclides in the West Valley HLW

The West Valley HLW contained a mixture of both long-lived and short-lived radionuclides. Table 2-1 of this evaluation includes, for example, long-lived radionuclides listed in Table 4-1 such as technetium 99 and short-lived radionuclides listed in Table 4-2 such as strontium 90 and cesium 137[23].

The classification requirements of 10 CFR 61.55 for waste containing both long-lived and short-lived radionuclides are as follows:

(1) 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 that determined by the concentration of nuclides listed in Table 2.

(2) 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.

For mixtures of radionuclides, 10 CFR 61.55 specifies that the sum of fractions rule will be used in determining waste classification. This rule entails dividing each radionuclide's concentration by the appropriate limit, adding the resulting fractions, and comparing their sum to 1.0. A sum of fractions less than 1.0 indicates compliance of the radionuclide mixture with the relevant classification criteria.

As noted previously, DOE Guide 435.1-1 indicates that one criterion for determining key radionuclides in waste is their importance in satisfying safety requirements comparable to the performance objectives of 10 CFR Part 61, Subpart C for the waste disposal facility. These performance objectives are described in Section 5.2.2 below.[24]

[23] The approximate half-lives of these radionuclides are as follows: strontium 90, 28 years; cesium 137, 30 years; and technetium 99, 212,000 years (HEW 1970).

[24] In practice, meeting the waste acceptance criteria for the disposal facility ensures that the facility performance objectives will be achieved. The rationale for this conclusion for a DOE LLW disposal facility such as the Nevada National Security Site may be briefly summarized as follows:

- DOE performance objectives for its LLW disposal facilities are comparable with those of 10 CFR 61, Subpart C;
- Disposal site performance in compliance with the performance objectives is determined by a performance assessment of the facility and by a composite analysis that considers other radioactivity sources in the area along with the radioactivity in the disposal site;
- These analyses are based on a projected total radionuclide inventory for the full, closed disposal site;
- This projected total inventory is based on the waste acceptance criteria, thus linking these criteria directly to the calculated disposal site performance;
- The subject LLW stream (the Vitrification Melter) will meet the waste acceptance criteria; and
- Meeting the waste acceptance criteria will therefore ensure that the performance objectives will be achieved, because waste meeting these criteria would not increase the assumed waste inventory used in the performance assessment analyses.

These matters are addressed in more detail in Section 5.2. The link between waste acceptance criteria and disposal site performance described in this footnote is similar for the commercial WCS LLW waste facility. Appendix C shows that the State of Texas performance objectives in Texas Administrative Code, Title 30, Part 1, Chapter 336, Subchapter H, Rules §336.723-727 mirror the NRC performance objectives.

4.1.5 Radionuclides Important to the Disposal Site Performance Assessments

Because meeting the waste acceptance criteria for a given disposal facility ensures that the facility performance objectives will be achieved, those radionuclides that are of particular importance in the disposal site performance analyses are considered in identifying key radionuclides. These radionuclides are Tc-99, Th-229, U-233, U-234[25], U-238, and Pu-239 for the Nevada National Security Site Area 5 waste disposal area (DOE 2010b, NST 2011). The results of the latest performance assessments of the Nevada National Security Site LLW disposal areas are discussed in Section 5.2 of this evaluation.

For the commercial WCS LLW disposal facility, the Radioactive Material License (TCEQ 2011) identifies total radioactivity limits for three radionuclides for disposal in the Federal Facility Waste Disposal Facility: C-14, Tc-99, and I-129. These radionuclides (which also are included in Table 1 of 10 CFR 61.55) contribute most to predicted dose according to the WCS performance assessment submitted with the WCS license application (WCS 2007) and are therefore important to meeting the performance objectives for the WCS facility. The WCS performance assessment is discussed in Section 5.2[26]. The Texas Administrative Code in §336.723-727 sets forth performance objectives for LLW disposal facilities, which track the NRC performance objectives in 10 CFR Part 61, Subpart C, as further discussed in Section 5.2 and Appendix C of this evaluation.

4.1.6 Conclusions About Key Radionuclides in the Vitrification Melter

Based on consideration of the factors discussed above, DOE considers all radionuclides listed in Tables 1 and 2 of 10 CFR 61.55 to be key radionuclides for the purposes of this evaluation, with the caveat that some are of lesser importance due to their low concentrations in the waste, their small dose conversion factors, or both. Table 4-3 shows these radionuclides.

Table 4-3. Key Radionuclides for this Evaluation

Radionuclide	10 CFR 61.55 Long-Lived Radionuclides	10 CFR 61.55 Short-Lived Radionuclides	Radionuclides Important to PA
H-3		X	
C-14	X		X(1)
Co-60		X	
Ni-59	X		
Ni-63		X	
Sr-90		X	
Nb-94	X		
Tc-99	X		X(1)(2)
I-129	X		X(1)
Cs-137		X	

[25] U-234 present at the time of disposal is the predominant source of Pb-210 (DOE 2010b). Pb-210 was not identified as a key radionuclide because its presence at the time of estimated maximum dose is due to U-234 in the disposed of waste, rather than Pb-210 in the waste.

[26] WCS is required to have a performance assessment maintenance plan and to update the performance assessment, consistent with this plan, prior to receipt of waste and annually thereafter (License condition 87, TCEQ 2011).

Table 4-3. Key Radionuclides for this Evaluation (continued)

Radionuclide	10 CFR 61.55 Long-Lived Radionuclides	10 CFR 61.55 Short-Lived Radionuclides	Radionuclides Important to PA
Th-229			X(2)
U-233			X(2)
U-234			X(2)
U-238			X(2)
Np-237(3)	X		
Pu-238(3)	X		
Pu-239(3)	X		X(2)
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		

NOTES: (1) Radionuclides important to the performance assessment of the WCS Federal Facility Waste Disposal Facility (WCS 2007).
(2) Radionuclides important to the performance assessment of the Area 5 Radioactive Waste Management Site.
(3) Alpha emitting transuranic radionuclides with half-life greater than five years (NRC 1982, Table 4.2).

Performance of the Area 5 Radioactive Waste Management Site was last evaluated at the end of Fiscal Year 2010 as described in the *2010 Annual Summary Report for the Area 3 and Area 5 Radioactive Waste Management Sites at the Nevada Test Site Nye County, Nevada: Review of the Performance Assessments and Composite Analyses* (NST 2011, DOE 2011d), using the estimated closure inventory. This evaluation showed that:

- The maximum member of public all-pathways dose was predicted to occur at 1,000 years for the resident farmer scenario. The resident farmer dose was predominately due to Tc-99 (82 percent) and Pb-210 (13 percent). Lead-210 present at 1,000 years would be produced predominately by radioactive decay of U-234 present at the time of disposal. (DOE 2011d)
- The maximum acute intruder dose was predicted to occur at 1,000 years for the shallow land burial disposal units under the acute construction scenario. The acute intruder dose is caused by multiple radionuclides including U-238 (31 percent), Th-229 (28 percent), Pu-239 (8.6 percent), U-233 (7.8 percent) and U-234 (6.7 percent). The maximum chronic intruder dose occurred at 1,000 years for the intruder-agriculture scenario. The intruder-agriculture dose is due predominantly to Tc-99 (75 percent) and U-238 (9.5 percent). (DOE 2011d)

Performance of the WCS Federal Facility Waste Disposal Facility was evaluated by WCS in connection with the WCS license application (WCS 2007). The three radionuclides limited in quantity in the radioactive material license (TCEQ 2011) – C-14, Tc-99, and I-129 – are the radionuclides that contribute most significantly to estimated dose in the performance assessment.

Three other alpha-emitting radionuclides known to be present in the West Valley HLW – Am-242m, Cm-245, and Cm-246 – were also considered. However, these radionuclides were determined not to be key radionuclides because they are not present in the waste package in amounts sufficient to be important to disposal site performance. (DOE 2011c)

4.2 Removal to the Maximum Extent Technically and Economically Practical

Removal to the maximum extent "technically and economically practical" is not removal to the extent "practicable" or theoretically "possible."[27] Nor does the criterion connote removal which may be notionally capable of being done.[28] Rather, the adverbs "technically" and "economically" modify and add important context to that which is contemplated by the criterion. Moreover, a "practical" approach as specified in the criterion is one that is "adapted to actual conditions" (Fowler 1930); "adapted or designed for actual use" (Random House 1997); "useful" (Random House 1997); selected "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure" (Random House 1997); fitted to "the needs of a particular situation in a helpful way" (Cambridge 2004); "effective or suitable" (Cambridge 2004). Therefore, the evaluation as to whether a particular key radionuclide has been or will be removed to the "maximum extent that is technically and economically practical" will vary from situation to situation, based not only on reasonably available technologies but also on the overall costs and benefits of deploying a technology with respect to a particular waste stream. The "maximum extent that is technically and economically practical" standard contemplates, among other things: consideration of expert judgment and opinion; environmental, health, timing, or other exigencies; the risks and benefits to public health, safety, and the environment arising from further radionuclide removal as compared with countervailing considerations that may ensue from not removing or delaying removal; life cycle costs; net social value; the cost (monetary as well as environmental and human health and safety costs) per curie removed; radiological removal efficiency; the point at which removal costs increase significantly in relationship to removal efficiency; the service life of equipment; the reasonable availability of proven technologies; the limitations of such technologies; the usefulness of such technologies; and the sensibleness of using such technologies. What may be removal to the maximum extent technically and economically practical in a particular situation or at one point in time may not be that which is technically and economically practical, feasible, or sensible in another situation or at a prior or later point in time. In this regard, it may not be technically and economically practical to undertake further removal of certain radionuclides because further

[27] In evaluating whether key radionuclides have been removed to the maximum extent that is "technically and economically practical", DOE has considered the guidance in DOE Guide 435.1-1 as well as the plain meaning of the phrase "technically and economically practical." DOE's evaluation also reflects a risk-based approach, and is consistent with the NRC Policy Statement concerning WVDP decommissioning criteria for waste to remain at the WVDP (NRC 2002), NRC staff guidance for NRC consultation activities related to DOE waste determinations (NRC 2007), and the approach taken pursuant to the similar criterion in Section 3116(a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (see e.g., Basis for Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center Tank Farm Facility (DOE 2006)).

[28] Compare the meaning of practical with the meaning of *practicable*. As commonly understood, *practicable* refers to something that can be put into effect. *Practical* refers to something that is also sensible and worthwhile. Thus, it might be *practicable* to transport children to school by balloon, but it would not be *practical*. (Webster's 1984)

removal is not sensible or useful in light of the overall benefit to human health and the environment.[29]

4.2.1 Technical Practicality Assessment

This section identifies different methods to remove key radionuclides from the Vitrification Melter that were considered before the Melter was shut down and describes the evaluations performed for four representative removal methods to determine whether they would have been technically practical. It also addresses the technical practicality of dismantling the Vitrification Melter in its current condition.

As noted above, HLW vitrification was completed in September 2002. At this point, increasingly dilute liquids were being retrieved from the underground storage tanks and the Vitrification Melter's design life was nearing its end. Before shutting down the vitrification process – which would eliminate the only available means of effectively managing additional removed radionuclides – DOE evaluated various methods of removing key radionuclides and utilized two effective methods as described below.

4.2.2 Methods Considered

The design and operation of the Vitrification Melter posed unique challenges from a decontamination standpoint. If the Melter were to be shut down without first being decontaminated, the molten glass containing HLW would immediately harden. The resulting condition would amount to a steel box containing a substantial amount of key radionuclides incorporated into hardened glass that would be exceedingly difficult to remove.

With this in mind, DOE considered a variety of methods to remove key radionuclides from the Vitrification Melter as discussed below. The best methods would involve reducing key radionuclide concentrations in the Melter molten glass pool and then extracting as much of the molten glass as practical before Melter shutdown.

Other methods considered included those discussed in DOE's Decommissioning Handbook (DOE 1994). This handbook describes a wide range of technologies used for decontamination at DOE sites. Some of these methods – such as vacuuming, flushing with water, grinding, grit blasting, and milling – are widely used industrial technologies. Others are innovative technologies developed with the support of the DOE Office of Science as part of a continuing program to improve methods used in decontamination and decommissioning work. Some innovative decontamination technologies have been tested in field demonstration projects sponsored by DOE as described in innovative technology summary reports issued by the DOE Office of Science and Technology (now Office of Science), and these were considered. However, none of these technologies was identified as specifically applicable to a vitrification melter.

DOE's Decommissioning Handbook identifies advantages and disadvantages of decontamination technologies in various applications. The most relevant application for decontaminating the Vitrification Melter internals was "Embedded Material and Some Oxide Surfaces." The DOE Handbook considered five technologies to be highly effective in this

[29] As a general matter, such a situation may arise if certain radionuclides are present in such extremely low quantities that they make an insignificant contribution to potential dose to workers, the public, and the hypothetical human intruder.

application: (1) hydroblasting, (2) ultra high pressure water grit blasting, (3) drill and spall, (4) paving breaker/chipping hammer, and (5) expansive grout.

DOE developed a vacuum extraction system using evacuated canisters to remove the majority of residual dilute molten glass from the Vitrification Melter prior to shutdown. This system could only be utilized before the contents of the Vitrification Melter were allowed to solidify after the heaters were secured and the Melter allowed to cool.

After considering available options, the WVDP determined that three fundamental approaches existed for potential removal of additional key radionuclides from the Vitrification Melter: (1) processing solutions with lower radionuclide concentrations, (2) mechanical means (including the five technologies the DOE Handbook identified as high effective in the application just discussed) , and (3) dismantlement.

The methods evaluated in detail were:

- Processing vitrification system decontamination solutions;
- Mechanical – evacuated canisters, ball milling; and
- Dismantlement – size reduction and waste segregation.

The objective of each of these potential methods was to remove residual material in the equipment, including key radionuclides, to the extent that was technically and economically practical. The remaining available options were considered to be experimental[30] on such an application as a vitrification melter. In 2008, DOE reconsidered use of a dismantlement approach to determine whether using this approach with present-day technologies would be technically practical as discussed in Section 4.2.6.

4.2.3 Processing Vitrification System Decontamination Solutions

A technically and economically practical method for removing key radionuclides from the Vitrification Melter involved processing decontamination solutions from other vessels. This method was used successfully while the Vitrification Facility was still operational[31].

Vitrification System Flushing

In late 2000, DOE commissioned a Vitrification Completion Team composed of representatives from DOE, NYSERDA, West Valley Nuclear Services, and NRC to review issues surrounding the ability to complete vitrification operations (VCT 2001). This team developed an approach to retrieving waste from the underground waste tanks, washing and characterizing the residual tank materials, and flushing the vitrification system prior to completing a controlled shutdown of the Melter.

The vitrification system flushes involved flushing Tank 8D-1, Tank 8D-2, the waste header, Tank 8D-4, Sludge Mobilization System piping, portions of the Supernatant Treatment System, the Low-Level Waste Evaporator, the Concentrator Feed Makeup Tank, the Melter Feed Hold Tank,

[30] DOE considered it necessary to use a proven, mature technology to avoid unnecessary development costs and the inherent uncertainties that would have been associated with an experimental technology.

[31] Nitric acid and demineralized water were used in the flushes. Note that the accessible portions of the exterior of the Vitrification Melter were also decontaminated by flushing with pressurized water as required by the *HLW Processing Systems Flushing Operations Run Plan* (WVNSCO 2002a)

the Submerged Bed Scrubber, and the Vitrification Melter. The Concentrator Feed Makeup Tank and the Melter Feed Hold Tank were required to be flushed by two passes using high-pressure spray. For the Vitrification Melter, this process involved feeding material with lower than normal radioactivity concentrations to reduce the Melter radionuclide inventory. (WVNSCO 2002a)

The flushing plan (WVNSCO 2002a) acknowledged concerns over limited Vitrification Melter life but stated that "flushes may be repeated based on the results obtained." Based on this criterion, the Concentrator Feed Makeup Tank and the Melter Feed Hold Tank were each flushed three times instead of the required two times[32]. A deliberate decision was made that the flushing had been satisfactorily completed before the evacuated canisters were used and the Vitrification Melter shutdown (WVNSCO 2002c and WVNSCO 2002d).

A detailed description of the vitrification system flushing accomplishments appears in the *Report on Deployment of Miscellaneous Tanks and Piping Cleaning Equipment and Methodology* (WVNSCO 2002b). This report includes photographs that demonstrate the effectiveness of the flushing on the internals of the Concentrator Feed Makeup Tank and the Melter Feed Hold Tank. Additional flushing of these vessels and other vitrification system equipment was determined to be unnecessary based primarily on the visual inspections that showed the vessel interiors to be clean and free of visible waste and on dose rate data (WVNSCO 2002c). The flush solutions were reduced in volume by evaporation and fed to the Vitrification Melter as discussed below.

Vitrification Melter

It was not feasible to literally flush the Vitrification Melter like the vessels used to feed slurry to the Melter because the Melter had to remain in operation to avoid hardening of the molten glass in the Melter cavity. A high-pressure water spray apparatus could not be used inside the Melter while it was in operation, as had been done with the Concentrator Feed Makeup Tank and Melter Feed Hold Tank.

Rather, decontamination solutions used to flush these two vessels (the Concentrator Feed Makeup Tank and Melter Feed Hold Tank) were slowly fed to the Melter and thereby significantly reduced the radionuclide concentrations in the Melter. This concentration reduction substantially reduced the amount of residual radioactivity in the glass that could not be removed from the Vitrification Melter by the second decontamination method used – the evacuated canister system, which is discussed below.

The effectiveness of this method in decontaminating the Melter – that is, in reducing the concentrations of key radionuclides in the molten glass pool before use of the evacuated canisters – can be determined by comparing canister dose rates. These dose rates are proportional to the Cs-137 concentration in the material inside the canister[33]. Figure 4-1 shows how canister dose rates dropped after the decontamination flush solutions entered the Melter glass pool.

[32]Each flush took 55-60 minutes and involved two separate passes with the spray nozzle (WVNSCO 2002b).

[33] The measured concentrations of feed material in the Concentrator Feed Makeup Tank are not used for this purpose because of the complex relationship between melter input and melter output. Melter input consisted of HLW slurry – and, for the last two batches, vitrification system decontamination solutions consisting of nitric acid, water, removed waste, and glass formers – that was slowly feed into the melter on a continuous basis. Melter output consisted of molten glass airlifted from the glass pool multiple times to fill the stainless steel canisters. Given the input-output differences, the molten glass pool in the Vitrification Melter typically contained material from more than one feed material batch.

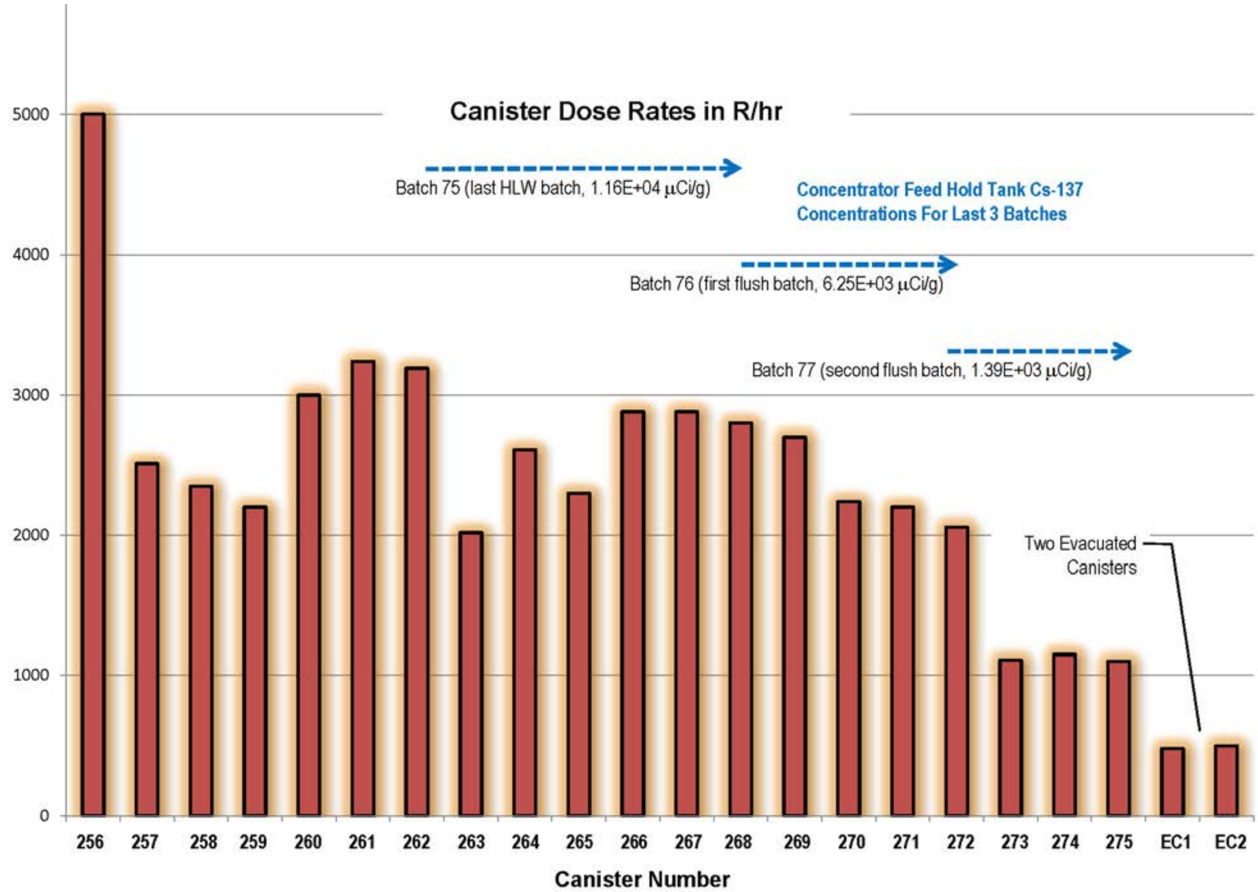


Figure 4-1. Relationship Between Feed Material Cs-137 Concentrations and Canister Dose Rates

As can be seen in Figure 4-1, canister dose rates started falling while batch 76, the first of the two batches of decontamination solutions, was being fed to the Vitrification Melter. They dropped further as batch 77 was being fed to the Melter.

The decontamination factor associated with this process can be estimated by comparing dose rates on canisters 266 – 270 (which averaged 2,716 R/h) to the dose rates on the two evaluated canisters (which averaged 490 R/h). This calculation shows a decontamination factor of approximately 5.5.

This decontamination process – reducing radionuclide concentrations in the molten glass pool before use of the evacuated canisters – would be expected to remove key radionuclides in approximately the same proportions. Table 4-4 compares scaling factors based on analytical data from the last two batches of HLW slurry feed material with analytical data from samples of glass taken from the evacuated canisters, which represent the material in the Vitrification Melter at the time it was shutdown. Comparing the glass sample scaling factors to the average of scaling factors from batch 74 and batch 75 shows that Sr-90 was removed in essentially the same proportion as Cs-137 and that other key radionuclides were removed in slightly higher proportions than Cs-137. Analysis of input and output for the Vitrification Melter suggests that scaling factors based on batch 75 are more representative of the radionuclide distribution in the glass in the plugged exit port as can be seen in Table 4-4 below.

Table 4-4. Radionuclide Scaling Factors (Ratios to Cs-137)

Radionuclide	Scaling Factor Basis		
	Glass Sample Data(1)	Batch 74 Data(2)	Batch 75 Data(3)
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	3.23E-04

NOTES: (1) From WMG 2004b, Exhibit 1.
(2) From Concentrator Feed Makeup Tank analytical data shown in the spreadsheet provided to NRC (WVES 2011).
(3) Based on Verification Analytical System Tracking (VAST) sample 01-2498 (WVES 2011).

The economic practicality of maintaining the Vitrification Melter operational to process additional vitrification system decontamination solutions is discussed below.

4.2.4 Evacuated Canisters

The WVDP developed the Evacuated Canister System in parallel with the overall Vitrification Facility design to enable removal of the residual dilute molten glass expected to remain in the Vitrification Melter prior to its shutdown and cooling. The Evacuated Canister System consisted of two stainless steel canisters the same size as a standard HLW canister but equipped with a special L-shaped "snorkel" assembly, both of which were designed to be positioned within a steel cage and positioned over the Vitrification Melter. The residual glass was retrieved into the canisters under vacuum after the plug within the snorkel melted from the heat of the Vitrification Melter after its insertion.

Figure 4-2 shows the snorkel portion of the evacuated canister in the process of removing molten glass from the Vitrification Melter. The metal has been heated by the molten glass to the point where it produced the orange glow shown in the figure. The lower portion of the snorkel was designed and positioned to reach approximately six inches from the bottom of the Vitrification Melter cavity for the first evacuated canister and approximately two inches from the bottom for the second one.

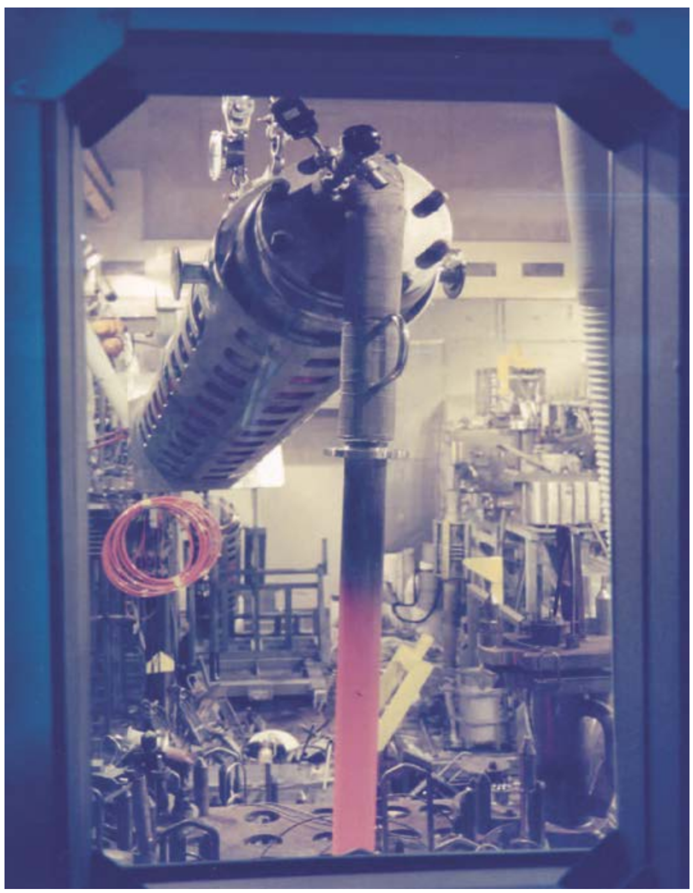


Figure 4-2. Evacuated Canister Removing Residual Glass from the Vitrification Melter

During deployment, approximately 88 percent of the residual molten glass volume was removed using this system. This technology was technically practical only for removal of key radionuclides from the Vitrification Melter while they were contained in the molten glass. As a bulk waste removal method that simply removed molten glass from near the bottom of the homogeneous glass pool[34], the Evacuated Canister System removed all key radionuclides from the Vitrification Melter with approximately equal efficiency.

4.2.5 Mechanical Decontamination

This process would have used a combination of grinding and impact in a ball mill to separate radioactive contamination from the base material.[35] This process was evaluated as representative of decontamination by mechanical means.

This process was initially evaluated for potential use in decontamination of small pieces of equipment and other materials contaminated with HLW during the vitrification process. In that application, it was envisioned that equipment pieces would tumble against each other within a closed system, possibly using zirconia as an abrasive to promote grinding action. This process would dislodge the contamination, which would be returned to the vitrification process for solidification as HLW. (WVNSCO 2001)

The WVDP evaluation of this process included small-scale tests performed using a laboratory jar mill. Tests were performed on specimens consisting of Inconel and stainless steel plates with glass annealed to their surfaces, short lengths of pipe containing dried slurry, and glass chunks. These tests showed that the process could decontaminate metal and reduce the size of glass pieces. (WVNSCO 2001)

Disadvantages evident from the laboratory tests included production of fine glass powder, which could cause problems in the process, and a propensity for embedding contamination in the metal. These characteristics would have limited the effectiveness of the process.

DOE concluded that the process could be optimized, but that development for the Vitrification Melter application would be costly and the results uncertain, resulting in another disadvantage since DOE was seeking a proven, mature technology to avoid the uncertainties inherent in unproven technologies. The WVDP determined that this process was not technically practical because of such disadvantages (WVNSCO 2001). Given this conclusion, details of how to use the process on the Vitrification Melter and how to dispose of the resulting waste stream were not developed.

4.2.6 Dismantlement

Another mechanical option considered involved dismantling the Vitrification Melter. Dismantling the Vitrification Melter would involve cutting it up, removing and segregating residual glass and refractory brick, characterizing and packaging the waste into disposal-ready containers, storing the wastes in an interim onsite location, and eventually shipping the containers offsite for

[34] Natural convection currents within the Vitrification Melter molten glass pool from heat input ensured a homogeneous mixture of the calcined waste and glass formers (Petkus 2002, DesCamp and McMahon 1996).

[35] In a commercial ball mill, balls tumble around within a rotating cylinder breaking material that becomes sandwiched between the balls and the cylinder walls. Such systems typically use media ½ inch or larger as the grinding instruments. A laboratory jar mill is a small version of a ball mill.

disposal. Because of the radiation levels associated with the Vitrification Melter, such dismantlement would need to be performed remotely in an area such as the Vitrification Cell.

Three disadvantages in this approach are evident:

- It would not remove key radionuclides, but simply segregate the residual glass into a separate waste stream;
- Dismantlement of the Vitrification Melter would result in the pieces being placed in multiple containers for disposal, rather than the intact Vitrification Melter being placed in a single shipping and disposal container, thus requiring more handling and increased worker radiation exposure; and
- The potential generation of orphan waste because it is unlikely that the separated residual glass could be disposed of as LLW and it would not be in a form (e.g., encased in stainless steel) suitable for disposal as HLW.

Despite these disadvantages and the practical difficulties that would be associated with Vitrification Melter dismantlement, this process was considered in 2002 to be technically practical. It would still be technically practical today, albeit with the same major disadvantages.

4.2.7 Other Potential Decontamination Methods

DOE considered whether other decontamination methods developed since 2002 could be technically practical for removing key radionuclides from the Vitrification Melter in its present condition with residual radioactivity in the hardened glass. More recent compilations of decontamination methods such as those in *The Decommissioning Handbook* (ASME 2004) were considered, but DOE identified no new methods that warranted evaluation for possible application to the Vitrification Melter.

4.2.8 Summary and Conclusions

In the technical practicality assessment, numerous methods that might have been used to remove key radionuclides from the Vitrification Melter were considered and four representative methods evaluated. Of these four methods, processing of decontamination solutions, the Evacuated Canister System, and Vitrification Melter dismantlement were determined to be technically practical.

4.3 Economic Practicality Assessment

Economic practicality includes consideration of total lifecycle costs, the cost per curie removed, the relationship between costs and removal of the key radionuclides, and the point in this relationship at which removal costs increase significantly and thus become impractical (DOE Guide 435.1). In this regard, removal of key radionuclides to the “maximum extent . . . economically practical” includes consideration of net social benefit, the remaining service life of equipment, expert judgment, and whether the benefits to health and safety outweigh the disadvantages, that is, whether further radionuclide removal would be useful and sensible in light of the overall benefit to human health and safety.

The evaluation of the economic practicality of additional radionuclide removal from the Vitrification Melter focused on:

(1) Processing of additional decontamination solutions before shutdown of the Vitrification Melter, considering the effectiveness of the processing already accomplished;

(2) Use of a third evacuated canister to remove additional molten glass from the Vitrification Melter Cavity; and

(3) Vitrification Melter dismantlement, in comparison with intact removal and shipment of the intact Vitrification Melter for offsite disposal.

In each case, the evaluation compared the potential benefits in improved worker and public health and safety (i.e., reduced worker and public risk from radiation exposure) with the expected impacts.

A cost-benefit analysis was performed using methodology consistent with guidance issued by NRC in NUREG/BR-0058, *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission* (NRC 1995b) and the companion handbook, NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook*, (NRC 1997). This analysis is documented in a detailed report (Perdue 2004). The following discussion makes use of some information from that report.

4.3.1 Evaluation of Additional Decontamination Solutions Processing

This subsection discusses the results of an evaluation performed to determine the economic practicality of additional vitrification system flushing with regard to decontamination of the Vitrification Melter.

Plans for the Vitrification System Flushes

The flushes were performed in accordance with the flushing plan (WVNSCO 2002a) as part of the system shutdown program developed by the Vitrification Completion Team. The flushing plan (WVNSCO 2002a) stated that "the primary objectives of the flushing are to transfer to the Vitrification Melter as much HLW as technically and economically practical and at the earliest opportunity so it could be vitrified." It also identified expected post-flushing conditions for each component.

The plan provided for flushing Tank 8D-1, Tank 8D-2, the waste header, Tank 8D-4, Sludge Mobilization System piping, portions of the Supernatant Treatment System, the Low-Level Waste Evaporator, the Concentrator Feed Makeup Tank, the Melter Feed Hold Tank, the Submerged Bed Scrubber, and the Vitrification Melter. The flushing media included nitric acid, water, simulated waste, and glass formers. The Concentrator Feed Makeup Tank and the Melter Feed Hold Tank were required to be flushed by at least two passes using high-pressure spray.

The plan described plans for the Vitrification Melter as follows:

"<u>Pre-Flush Condition</u> - The Melter is full of molten HLW glass. Since the Melter was started on non-radioactive glass, the radioactivity may not have penetrated very far into the refractory.

<u>Flushing Operations</u> - After completion of HLW transfers from the Waste Tank Farm, feed with lower than normal radioactivity will be fed to the Melter. The activity in these batches comes from normal recycle streams, such as, the LWTS [Liquid Waste Treatment System] Evaporator, flushing activities, canister decontamination and the SBS [Submerged Bed Scrubber]. The lower activity batches will serve to dilute the Melter inventory of radioactive species. Although the recycle streams continue to be processed, and tank heels are adsorbed into the next batch, the final batch will consist primarily of simulated waste and glass formers.

Expected Condition After Flushes - It is expected that 3 to 6 inches of glass including a "noble metal sludge" will be left in the bottom of the Melter after the vacuum canisters.

Data Collection / Sampling - Sampling of glass shards from canisters are taken per SOP [standard operating procedure]. Videotaping of evacuated canister evolution is expected to occur."

The flushing plan acknowledged concerns over limited Vitrification Melter life but stated that "flushes may be repeated based on the results obtained." The NRC staff was consulted in connection with developing plans to complete the vitrification program, including the system flushes, as noted previously.

The Concentrator Feed Makeup Tank and the Melter Feed Hold Tank were each flushed three times instead of the required two times. A decision was then made that the flushing had been satisfactorily completed before it was concluded (WVNSCO 2002h and WVNSCO 2002i).

The flush solutions were sent to the Vitrification Melter in two batches, batch 76 and batch 77 as shown in Figure 4-1. The flush solutions totaled approximately 220,000 gallons, although more than 95 percent of this amount was evaporated. The flushes effectively decontaminated the Melter by reducing radionuclide concentrations in the molten glass pool by a factor of approximately 5.5, as noted previously.

The flushing plan, as noted previously, provided that "flushes may be repeated based on the results obtained." Although the results obtained were satisfactory after a third flush of both vessels, additional flushes of the Concentrator Feed Makeup Tank, the Melter Feed Hold Tank, and/or the entire Vitrification System could have been performed before use of the evacuated canisters.

Conceptual Model for an Additional Flush

A simplified model with the following characteristics (Kurasch 2011) was developed to evaluate the economic practicality of processing additional flush solutions in the Vitrification Melter:

- Actual recorded data are used, including radionuclide concentrations from samples of materials in the Concentrator Feed Makeup Tank, Melter cavity glass levels, canister weights, and canister glass heights.
- Total activity is considered, rather than just Cs-137.
- Progressive cumulative results are produced on a canister-by-canister basis.
- The baseline starting point is canister 267, which contained 31,804 curies and 1,992 kg of glass, with an activity concentration of 15.97 curie/kg of glass. (Canister 267 was used as the baseline starting point for this analysis because canister 267 was filled at the time the last part of batch 75 was being fed to the Melter, which indicates that the radionuclide concentrations in in the Melter glass pool were approximately the same as the radionuclide concentrations in this feed material batch.)
- A glass density of 2.4 g/cm^3 is used consistent with the density value used in the vitrification program records.
- One additional hypothetical batch of flush solutions with glass formers is fed to the Melter before use of the evacuated canisters and this feed material batch is nonradioactive for conservatism.

- The actual airlift numbers are used, with fifteen airlifts assumed for filling the additional hypothetical canister, a value consistent with vitrification records.

Figure 4-3 illustrates the conceptual model. Microsoft Excel was used with these inputs to calculate the total activity in the Melter glass pool over time.

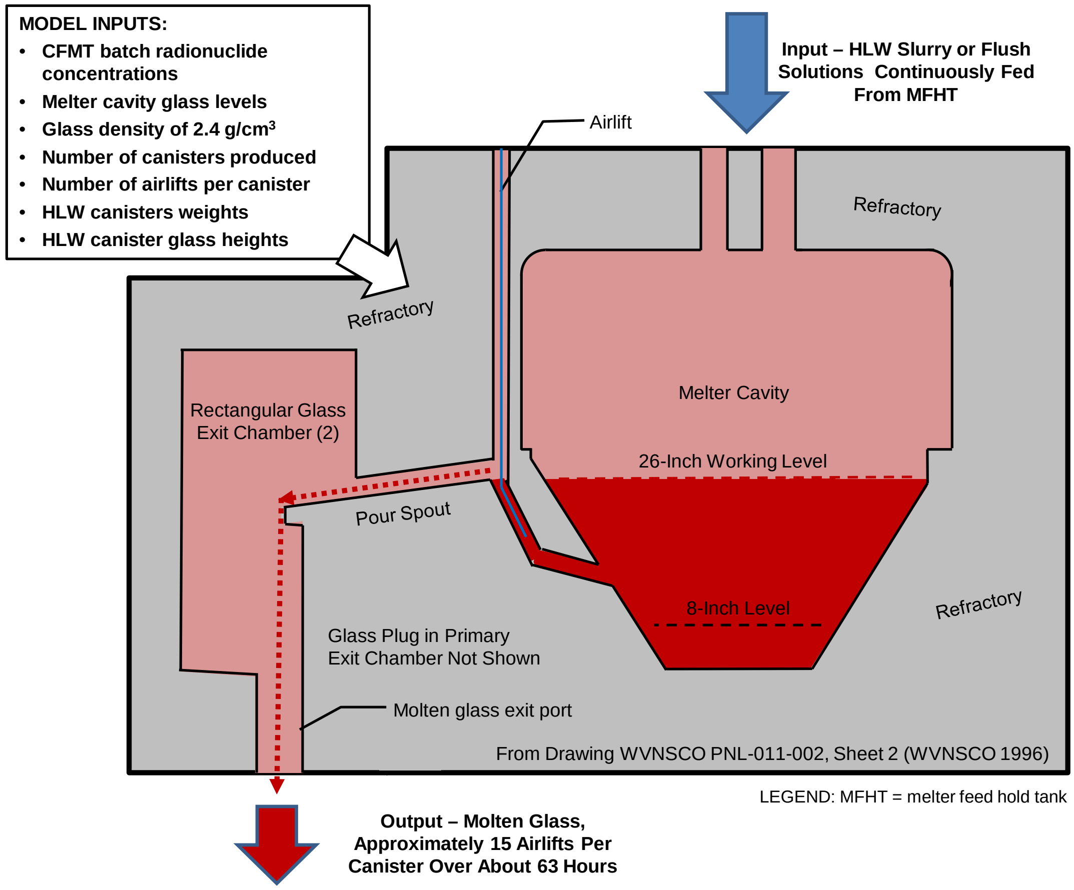


Figure 4-3. Conceptual Model

Results of Implementing the Model

The results predict a radioactivity concentration in the glass pool at the time the actual evacuated canisters were used – that is, after batch 77 was fed to the Melter – of 2.4 curies/kg. This value is in close agreement with the average of 2.51 curies/kg based on analytical data from glass shard samples taken from the two evacuated canisters. Table 4-5 shows the principle results.

Table 4-5. Estimated Effectiveness of Processing Another Flush Solution Batch(1)

Concentrator Feed Makeup Tank Batch and Last Canister Filled While this Batch Was Being Fed to the Melter	Glass Pool Activity (Ci)
75 (last HLW batch, canister 267)	31,908
76 (first flush solutions batch, canister 272)	16,012
77 (second flush solutions batch, canister 275)	4,303
78 (Hypothetical flush solutions third batch, one hypothetical additional canister)	1,814

NOTE: (1) From Kurasch 2011.

Figure 4-4 displays the model results.

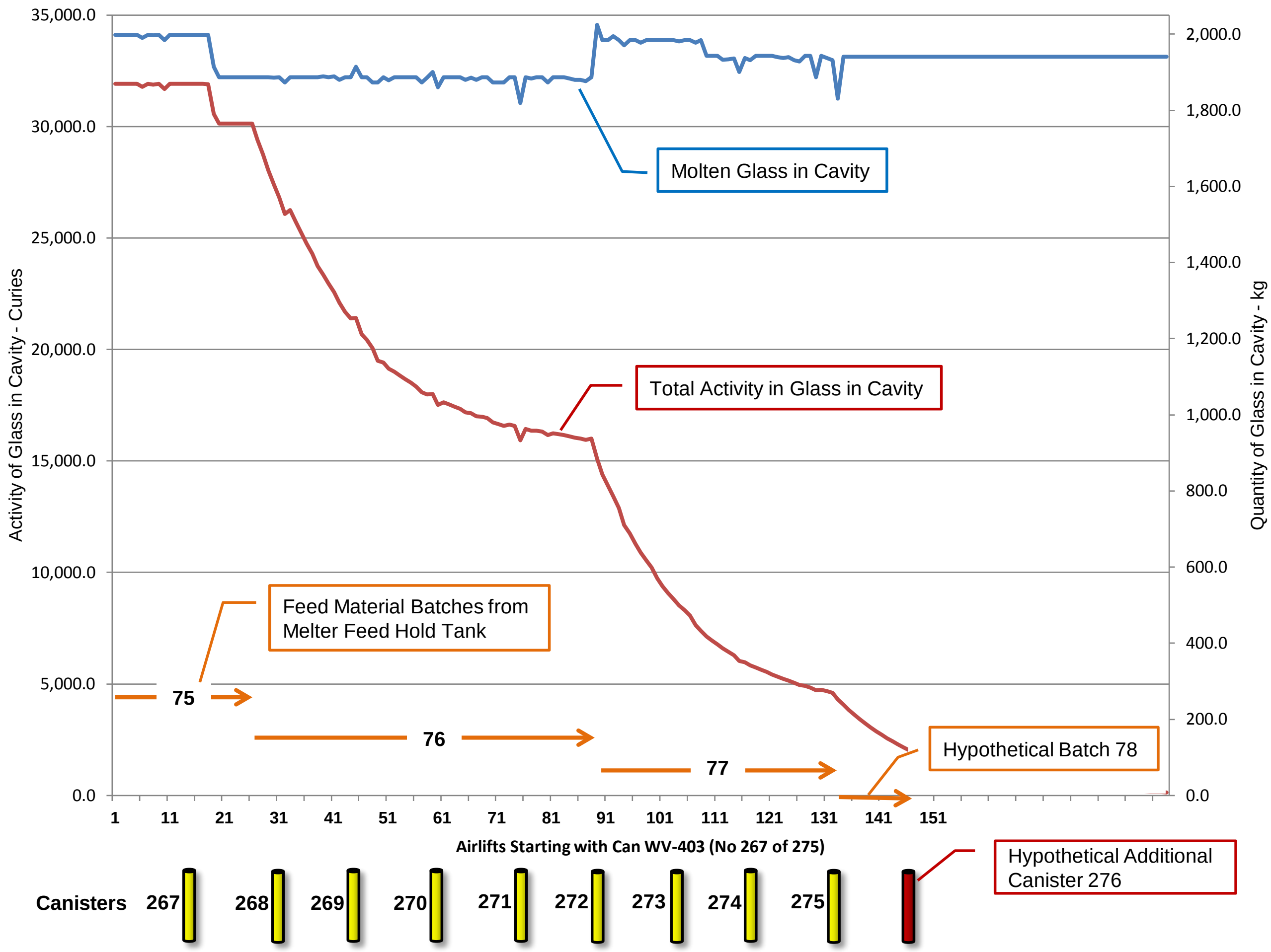


Figure 4-4. Predicted Total Activity in Vitrification Melter Glass Pool

Table 4-5 and Figure 4-4 show that the total activity in the molten glass pool in the Vitrification Melter cavity would have been approximately 1,814 Ci if an additional batch of nonradioactive feed material sufficient to produce a single additional canister had been processed in the Melter. If 88 percent of this amount had been extracted by the evacuated canisters, this would have left approximately 218 curies of total activity in the Melter cavity. The total activity in the Melter then would have been approximately 500 Ci counting the estimated activity in the plugged discharge port and in fissures in the refractory.

Note that the activity in the 4,303 Ci prediction for the total activity in the molten glass pool before use of the evacuated canisters compares to approximately 5,640 Ci based on the radionuclide concentrations measured in the glass shard samples from the evacuated canisters multiplied by the measured mass of glass in the Melter cavity. If the latter estimate were to be used to calibrate the model results, then total activity left in the Melter after processing the hypothetical additional flush solutions would be approximately 565 Ci.

Benefits of Processing the Additional Flush Solutions

Consideration was given to potential benefits from processing of additional flush solutions in the Vitrification Melter. However, such benefits would have been limited for the following reasons:

- The gamma radiation levels on the outside of the waste package in its present condition are low and compliance with radiological control program requirements in handing of the waste package at the WVDP and the LLW disposal facility will ensure the protection of individuals during operations related to disposal as discussed in Section 5.2.4.
- Worker radiation doses would not have been significantly reduced. The external dose rates with a reduced amount of residual radioactivity (and thinner shielding) would have been approximately the same. In any case, the highest dose rate three meters from the waste package, which is located three meters from the bottom, is produced by radioactivity in the plugged primary discharge port; this dose rate would not have been affected by a reduction in the radioactivity inside the Melter cavity from processing of additional flush solutions
- The processing of solutions from the flushes actually performed left the Melter in a condition suitable for disposal as LLW as discussed in Section 5.3.
- The potential impacts to the general population from disposal of the Melter waste package at the LLW disposal facility without further decontamination will be negligible as discussed in Section 5.2.2, so a further reduction in residual radioactivity would not have been beneficial from the standpoint of potential doses to members of the public.
- The potential impacts to an inadvertent intruder from disposal of the Melter waste package at the LLW disposal facility without further decontamination will be negligible as discussed in Section 5.2.3, so a further reduction in residual radioactivity would not have been beneficial from the standpoint of an inadvertent human intruder.

However, a single monetary benefit would have been realized had the hypothetical additional flush solutions been processed in the Vitrification Melter: The shielded container could have been designed and constructed of lighter weight steel, which would have reduced costs associated with materials, fabrication, and transportation by approximately $100,000.[36]

Costs of Processing the Additional Flush Solutions

A total of approximately $1 million in additional costs (in 2002 dollars) would have been involved. One additional flushing and processing cycle would have taken about two weeks to complete at a cost of approximately $1 million, based on vitrification system operating costs that were running $25 million to $30 million per year[37]. (Kurasch 2011)

Another factor in considering costs of processing of additional flush solutions was the limited Vitrification Melter service life. Analysis indicated a 35 percent probability of Melter failure occurring

[36] This estimate is for a one-time savings. It is a conservative, order-of-magnitude estimate in 2002 dollars for savings in raw material, handling, and transportation costs (Kurasch 2011).

[37] This estimate is conservative because it does not take other costs in account, such as the cost of manufacture of the additional canister, its transportation to the site, the monetary costs of additional worker radiation exposure, and indirect life-cycle costs. Interim onsite storage of one additional canister would have cost approximately $35,000 per year based on actual and predicted annual costs of interim storage, including surveillance and maintenance. The estimate does not include costs of ultimate disposition. (Kurasch 2011)

within six months of additional operation (Perdue 2004). A failure would have, for all practical purposes, stranded radionuclides within the Melter since neither processing of flush solutions nor use of the evacuated canister system would not have been feasible with the residual glass in the Melter cavity in solid form.[38]

Conclusions

The foregoing discussions show that processing of additional flush solutions in the Vitrification Melter to produce a single additional canister would have produced negligible, if any, benefits in terms of improved worker or public health and safety and would have resulted in approximately $1 million in additional monetary costs.

4.3.2 Evaluation of Use of a Third Evacuated Canister

Consideration was given to whether use of a third evacuated canister could have been economically practical.

Potential Benefits

A third evacuated canister might have been able to remove a small amount of additional molten glass, thereby removing additional key radionuclides.

Costs of Using a Third Evacuated Canister

Two evacuated canisters were available when the Melter cavity was emptied. The evacuated canister system was designed with the expectation that the first evacuated canister would be filled to the 80 percent level with molten glass extracted from the cavity. A second evacuated canister was manufactured and made available as a contingency. (Petkus 2002)

As it turned out, the first evacuated canister was able to remove molten glass sufficient to fill it only to the 57 percent level, which left more molten glass in the cavity than planned. The contingency canister was then deployed and removed additional molten glass from the cavity sufficient to fill this canister to the 42 percent level. Combined, the two evacuated canisters removed 23 percent more glass from the Melter cavity than a single canister filled to the 80 percent level.

Because only two evacuated canisters were available, with the second available for contingency purposes, the hypothetical third canister would have had to have been manufactured and transported to the site. If it is assumed that this could have been accomplished in two weeks – an improbably fast time considering procurement, manufacturing, and transportation – the additional cost, just for keeping the Vitrification Melter in operation during this period, would have been approximately $1 million based on vitrification system operating costs. (Kurasch 2011)

Inherent in this simple analysis is the assumption that the Vitrification Melter could have maintained the glass pool in a molten state while awaiting the hypothetical third evacuated canister. This would not have been possible because the molten glass level after use of the second evacuated canister was below the electrodes mounted in the side of the Melter cavity. This condition made joule heating impossible and the glass in the cavity would have continued to cool to

[38] In the interest of conservatism, no attempt was made to quantify other costs associated with processing of additional flush solutions, such as the monetary value of additional worker radiation dose that would have been necessary to continue Vitrification Melter operations.

the point where it solidified in the bottom of the cavity before the third canister could, as a practical matter, reasonably be expected to have been made available.

Conclusion

Even if the use of a third canister had been technically practical, it would not have been economically practical because of the high costs.

4.3.3 Evaluation of Vitrification Melter Dismantlement

Vitrification Melter dismantlement was evaluated in comparison to the costs associated with intact removal of the Melter, which was considered to be the base case in the analysis. Specifically, the cost of intact removal and shipment of the Vitrification Melter for disposal was compared to the costs for dismantlement with respect to the sum of the following:

- Process labor to remotely dismantle and size-reduce the Vitrification Melter;
- Process labor required to put either the intact Vitrification Melter or its dismantled pieces into disposal-ready containers, working remotely, and either ship the Melter for disposal or place it in interim storage,
- Transportation costs to interim off-site storage or final disposal, and
- Disposal costs.

The following table summarizes those costs and illustrates cost savings from shipping the Vitrification Melter intact versus dismantling the Vitrification Melter.

Table 4-6. Vitrification Melter Dismantlement Cost Analysis[1]

Cost Category	Intact Melter Option ($K)	Dismantled Melter Option ($K)	Cost Savings (Intact minus Dismantle) ($K)
Process Labor	354	3,368	-3,032
Transportation	3,258	1,692	1,566
Disposal	29	32	-3
TOTAL	3,641	5,110	-1,468

NOTE: (1) From Perdue 2004, Table 13.

The process labor costs for Vitrification Melter dismantlement are much larger than for leaving the Melter intact. It should be noted that this cost comparison assumed that the dismantled Vitrification Melter pieces could be disposed of as LLW. However, as noted above, it is likely that some Vitrification Melter components such as residual glass may be considered orphan waste and, as such, have no identified pathway for disposal. This cost comparison does not consider the implications of continued indefinite storage of orphan waste, another factor that adds conservatism to the results since the potential costs of evaluation of orphan waste, its onsite storage, and its ultimate disposal are not taken into account.

The analysis described above was completed in 2004. DOE considered whether improvements in technologies since that time would change the assessment results, with the Vitrification Melter now located in its shipping container and the low-density cellular concrete not yet poured into the Melter or the container. The conclusions were: (1) the Vitrification Melter would still have to be dismantled remotely in a shielded cell because of its radiation levels, which, considering radioactive decay, are about 90 percent of the levels in 2004, and (2) no improved cutting or dismantlement technologies developed since 2002 would make a significant difference in this process.

The estimates in Table 4-6 for Vitrification Melter dismantlement remain valid today with one exception: There would be additional costs associated with the Vitrification Melter dismantlement work to decontaminate the shielded cell where the work was done. Shielded cells that might be used for this purpose have been decontaminated since 2002. The work would result in spreading radioactive contamination in the work area, which would have to be cleaned up with the resulting additional radioactive waste placed in containers and disposed of offsite.

4.3.4 Summary and Conclusions

The technical practicality assessment showed that processing vitrification system decontamination solutions in the Vitrification Melter, removing molten residual glass using the Evacuated Canister System, and dismantling the Vitrification Melter were the technically practical options for removing key radionuclides from the Melter to the extent practical. The economic practicality analysis demonstrated that processing two batches of vitrification system decontamination solutions was economically practical, but that processing of additional decontamination solutions in the Vitrification Melter would not have been not economically practical because of high cost and negligible benefit. Dismantlement was considered economically impractical due to the high cost associated with dismantlement and uncertainties associated with management of the orphan waste that might be generated.

In summary, both processing of vitrification system decontamination solutions and the Evacuated Canister System were successfully deployed and removed key radionuclides to the maximum extent that is technically and economically practical.

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

Section Purpose

The purpose of this section is to evaluate whether the Vitrification Melter waste package will be managed to meet safety requirements comparable to the performance objectives of 10 CFR 61, Subpart C for disposal of low-level radioactive waste.

Section Contents

This section addresses whether the Vitrification Melter waste package will meet safety requirements comparable to the performance objectives of 10 CFR 61, Subpart C for disposal of low-level radioactive waste and explains how the Vitrification Melter waste package will meet criteria for disposal as low-level radioactive waste.

Key Points

- Management of the Vitrification Melter waste package will meet safety requirements comparable to the performance objectives of 10 CFR 61, Subpart C.
- The Vitrification Melter waste package will meet the waste acceptance criteria for the Area 5 Radioactive Waste Management Site at the Nevada National Security Site.
- The performance objectives in the Texas Administrative Code applicable to the commercial WCS low-level waste disposal facility for Federal waste mirror the performance objectives in 10 CFR 61, Subpart C and the facility must be operated to provide reasonable assurance that those performance objectives will be met; consequently, disposal of the Vitrification Melter waste package at the WSC Federal waste disposal facility will meet safety requirements comparable to the performance objectives of 10 CFR 61, Subpart C.

5.1 Introduction

The second criterion of Section II.B.2(a) of DOE Manual 435.1-1 is evaluated in this section. This criterion reads as follows:

"[The waste] will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, *Performance Objectives*."

Section 2 describes the design of the Vitrification Melter and its operational history. Section 2 also explains how the Vitrification Melter was characterized for residual radioactivity, with Table 2-2 providing the total activity estimates in the final waste form. As noted previously, the Vitrification Melter has been placed in a custom made shipping container and the void spaces in the Vitrification Melter and the shipping container will be filled with low-density cellular concrete in preparation for transport to an offsite disposal facility.

This section addresses the second criterion in the following subsections:

Section 5.2 begins by summarizing key DOE safety requirements related to disposal of LLW.

Section 5.2.1 describes DOE's general safety requirement and compares it with the similar general safety requirements promulgated by NRC and the State of Texas;

Section 5.2.2 provides the following information regarding requirements for the protection of the general population from releases of radioactivity:

- A description of the DOE requirements,
- A comparison between these requirements and the similar requirements promulgated by NRC and the State of Texas,
- A summary of the results of the most recent performance assessment for Nevada National Security Site Area 5 facility related to protection of the general population,
- An estimate of the impact on the general population of disposal of the Vitrification Melter in the Nevada National Security Site Area 5 facility, and
- A summary of the results of a performance assessment of the WCS Federal Facility Waste Disposal Facility related to protection of the general population.

Section 5.2.3 provides similar information related to protection of individuals from inadvertent intrusion into the closed LLW disposal facilities.

Section 5.2.4 discusses protection of individuals during operations at the WVDP, at the Nevada National Security Site, and at the WCS LLW disposal facility.

Section 5.2.5 compares DOE requirements for stability of the disposal site after closure with the requirements of the NRC and the State of Texas and briefly discusses preliminary closure plans for the Nevada National Security Site Area 5 facility and the WCS facility.

Section 5.3 begins with a discussion of DOE waste acceptance criteria.

Section 5.3.1 discusses waste acceptance criteria for the Nevada National Security Site.

Section 5.3.2 summarizes how it is determined that a waste package meets the Nevada National Security Site waste acceptance criteria.

Section 5.3.3 demonstrates that the Vitrification Melter Waste package meets the Nevada National Security Site waste acceptance criteria.

Section 5.4 discusses waste acceptance criteria for the WCS Federal Facility Waste Disposal Facility and how it would be established that the Vitrification Melter Waste package meets these criteria if the waste package were to be shipped to that facility for disposal.

5.2 DOE Safety Requirements

DOE has established requirements for management of radioactive waste to ensure protection of workers, the public, and the environment, and complies with applicable Federal, State, and local laws and regulations. DOE has also established specific requirements for its radioactive waste disposal facilities, including the Area 5 Radioactive Waste Management Site at the Nevada National Security Site. These requirements include:

(1) Performance objectives set forth in Chapter IV of DOE Manual 435.1-1, which include maximum dose limits;

(2) DOE regulations at 10 CFR Part 835 and DOE Order 458.1, *Radiation Protection of the Public and the Environment*,[39] cross referenced in Chapters I and IV of DOE Manual 435.1-1;

(3) Waste acceptance requirements, which, among other things, establish limits on radionuclides that may be disposed of based on a performance assessment of the facility;

(4) A performance assessment of the disposal facility, with updates, to provide reasonable expectation that DOE's performance objectives will not be exceeded;

(5) A composite analysis that considers other radioactivity sources in the area as well as the disposal facility;

(6) A performance assessment and composite analysis maintenance plan;

(7) A preliminary closure plan; and

(8) A monitoring plan (DOE Manual 435.1-1).

For wastes to be disposed of at DOE facilities, DOE establishes waste acceptance criteria, based upon an independently reviewed and accepted LLW performance assessment, which also includes provisions for maintenance and updating. Acceptability of the LLW performance assessment is verified against the performance objectives of Section IV.P of DOE Manual 435.1-1, as well as other requirements in DOE Manual 435.1-1, through an independent review process. This review serves as the basis for DOE to issue a Disposal Authorization Statement, which

[39] DOE Order 458.1 has cancelled and superseded DOE Order 5400.5 of the same name, which is cross referenced in DOE Manual 435.1-1.

specifies any additional conditions that the site may need to impose to ensure that the performance objectives of DOE Manual 435.1-1, IV.P are met.

Figure 5-1 illustrates the general process used to provide reasonable expectation that disposal site performance objectives are achieved, which is in addition to the use of formal waste acceptance requirements.

The following subsections address the specific DOE performance objectives, and relevant DOE regulations and Orders, for DOE LLW disposal sites. These performance objectives, regulations, and Orders are set forth or cross referenced in DOE Manual 435.1-1, and provide safety requirements comparable to the NRC performance objectives of 10 CFR 61, Subpart C, as discussed further in Appendix C. As explained in Appendix C, the performance objectives in the State of Texas regulations mirror the NRC performance objectives at 10 CFR 61, Subpart C, i.e., they are essentially identical except for the use of difference section numbers.

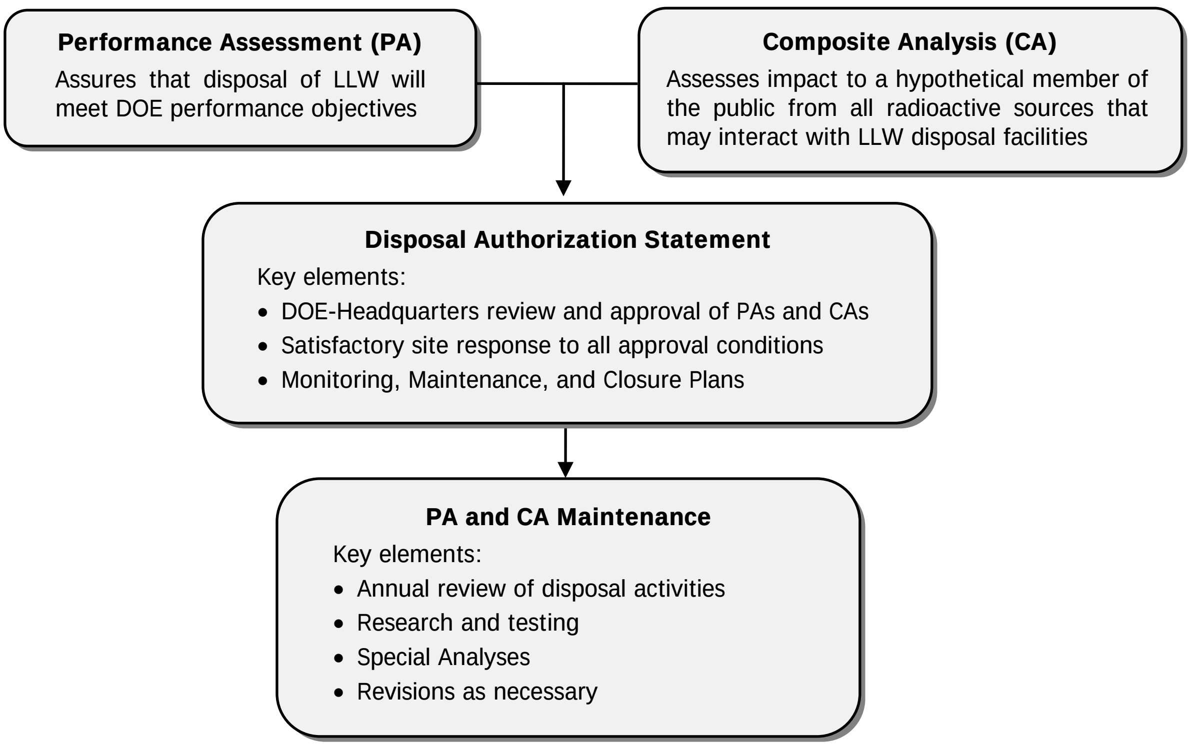


Figure 5-1. General Process Used to Ensure Performance Objectives Are Achieved

5.2.1 General Safety Requirement

The general requirement in DOE Manual 435.1-1, Section IV.P(1), is expressed as follows:

> "Low-level waste disposal facilities shall be sited, designed, operated, maintained, and closed so that a reasonable expectation exists that the following performance objectives will be met for waste disposed of after September 26, 1988."

As discussed further in Appendix C, the general requirement in NRC's performance objectives for licensed LLW disposal facilities at 10 CFR 61.40 sets forth a nearly identical, comparable requirement:

"Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44."

The general safety requirement of the State of Texas echoes the NRC general safety requirement as shown in Appendix C.

The four relevant DOE performance objectives are addressed in Subsections 5.2.2 through 5.2.5.

5.2.2 Protection of the General Population from Releases of Radioactivity

DOE requirements in DOE Manual 435.1-1, Section IV.P(1), read as follows:

"(a) Dose to representative members of the public shall not exceed 25 millirem (0.25 mSv) in a year total effective dose equivalent from all exposure pathways, excluding the dose from radon and its progeny in air.

(b) Dose to representative members of the public via the air pathway shall not exceed 10 millirem (0.10 mSv) in a year total effective dose equivalent, excluding the dose from radon and its progeny.

(c) Release of radon shall be less than an average flux of 20 pCi/m^2/s (0.74 Bq/m^2/s) at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/L (0.0185 Bq/L) of air may be applied at the boundary of the facility."

As discussed further in Appendix D, DOE's dose limits are comparable to those in the NRC performance objectives at 10 CFR 61.41, although DOE uses more current radiation protection methodology[40]. The NRC performance objective at 10 CFR 61.41 provides:

"Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable."

The requirement of the State of Texas for protection of the general population is virtually identical to the NRC requirement as shown in Appendix C.

Assessment of Area 5 Performance

The report of the most recent annual review of performance assessments and composite analyses for the Area 3 and Area 5 waste disposal facilities at the Nevada National Security Site was issued in 2011 (NST 2011). This report addresses matters such as new or revised waste streams, monitoring results, research and development, the inventory estimates at planned

[40] NRC recommends in NUREG-1573, *A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities*, *Recommendations of NRC's Performance Assessment Working Group* (NRC 2000), use of the more current radiation protection methodology, making NRC standards comparable to those of DOE in this regard.

closure, updated performance assessment results, and updated composite analysis results. It also identifies special analyses that were performed in the previous year[41].

As explained in the report of the annual review (NST 2011), the updated Area 5 performance assessment provides reasonable expectation that DOE's performance objectives will be achieved. This report summarizes the results of probabilistic analyses for Area 5 for the following scenarios:

- All pathways dose for members of the public, with the predicted peak annual dose estimated to occur at 1,000 years after planned closure (i.e., in 3,028, the end of the compliance period) for the resident farmer scenario;
- The air pathway dose for members of the public, with the predicted peak annual dose to a resident farmer at 1,000 years after facility closure; and
- The Radon 222 flux density at the surface of the disposal units, which is predicted to reach a peak 1,000 years after facility closure.

This report shows that the predicted potential doses to representative members of the public to be much less than the performance objective dose limits.

Estimated Impact of the Vitrification Melter

Even though the Vitrification Melter waste package meets the waste acceptance criteria for the Nevada National Security Site Area 5 Radioactive Waste Management Site, a dose assessment was performed to quantify the potential dose impact of disposal of this equipment in Area 5 for perspective and additional information (DOE 2010b)[42]. The impact was assessed by evaluating the Area 5 shallow land burial inventory disposed of through Fiscal Year 2009 with and without the

[41] A LLW disposal facility performance assessment involves detailed analyses of potential radiation doses to those who may be affected in future years to ensure that the closed facility will meet its performance objectives. These performance objectives include dose limits for a member of the public and for a hypothetical person who, unaware of the buried radioactivity, might drill a well into the buried waste, referred to as the post-drilling scenario, or establish a farm on the site, known as the intruder-agriculture scenario. A LLW disposal facility performance assessment makes use of two basic models.

A conceptual model describes all of the relevant properties of the disposal site. Area 5 is scheduled for closure in 2028. The estimated radionuclide inventory at closure is made up of two components: the known activity in the buried waste and the projected activity in waste to be disposed of in the future, which is based primarily on the waste acceptance criteria and the types and amounts of radioactivity in the waste already disposed on in the facility. The closure date and the estimated radionuclide inventory at closure are two examples of the many elements which make up the conceptual model

A mathematical model is used with the conceptual model to calculate potential doses under different scenarios. The Nevada National Security Site uses the GoldSim mathematical model, a widely-used software package that simulates the future behavior of the closed disposal site in a probabilistic manner, providing a range of results with different probabilities. The Nevada National Security Site typically expresses key performance assessment results as mean values and 95th percentile values.

Special analyses and composite analyses use similar methodologies, with the focus on the waste stream of interest and all relevant radioactivity sources at the site, respectively. Special analyses are performed for waste streams with a sum of fractions greater than one or where preliminary screening indicates that disposal of a new waste stream has a potential to alter performance assessment assumptions or conceptual models. A composite analysis is required for all DOE sites that manage radioactive waste; these analyses are updated annually.

[42] This dose assessment was performed using the same methodology as a special analysis even though no special analysis was required for the Vitrification Melter. A special analysis was not required for the Vitrification Melter waste package even though the waste stream sum of fractions exceeded one because the screening evaluation indicated that the available capacity sum of fractions (the fraction of the available capacity that the entire waste stream represents) was less than the criterion that would trigger a special analysis.

Vitrification Melter Inventory.[43] Table 5-1 shows the results for members of the public. The limiting pathways/scenarios are those among the various combinations of pathways and scenarios analyzed that are estimated to produce the highest annual all-pathways dose to representative members of the public.

Table 5-1. Estimated Maximum Dose Impacts to a Representative Member of the Public Associated With Vitrification Melter Disposal (mrem/y)[(1)]

Limiting Pathway/Scenario (Probability)	Limit	Estimated Maximum Annual Dose		
		Area 5 (Without Melter)	Area 5 (With Melter)	Difference Due to Melter
Member of Public: All Pathways/Resident Farmer (Mean)	25	1.4	1.4	negligible
Member of Public: All Pathways/ Resident Farmer (95th Percentile)	25	4.5	4.5	negligible

NOTES: (1) From DOE 2010b.

The information in Table 5-1 shows that disposal of the Vitrification Melter in the Area 5 facility is estimated to have no significant impact on facility performance insofar as members of the public are concerned.

The Waste Profile for the Vitrification Melter, which describes the characteristics of the waste as required by the waste acceptance criteria for the Nevada National Security Site Area 5 Radioactive Waste Management Site, was submitted by the WVDP in May 2010 (WVES 2010b). The Waste Profile was reviewed by the Nevada National Security Site Waste Acceptance Review Panel, a group of waste management specialists who review new and revised waste streams planned for disposal in the Area 5 Radioactive Waste Management Site. The panel's review resulted in formal acceptance of the Vitrification Melter waste package for disposal, conditional upon it meeting waste-incidental-to-reprocessing criteria in accordance with DOE Manual 435.1 (that is, this evaluation resulting in a determination that it can be managed as LLW) (DOE 2010a). Section 5.3 below discusses the waste acceptance process in more detail.

Assessment of WCS Federal Facility Waste Disposal Facility Performance

Unlike the Nevada National Security Site Area 5 LLW disposal facility, which has been in operation for decades, the WCS Federal Facility Waste Disposal Facility is not yet in operation. However, WCS included a performance assessment with its license application (WCS 2007) and completed an updated performance assessment in September 2011 (WCS 2011). The updated performance assessment of the Federal Facility Waste Disposal Facility considered a total inventory at closure of 26 million cubic feet of waste with 5.6 million curies of radioactivity[44]. It included the following estimated dose for the post-institutional control period: a maximum annual dose of

[43] These analyses were performed without accounting for the projected future inventory, as noted above, the normal practice used for special analyses. As such, they produce slightly different results than the performance assessments reported in the 2010 annual review (NST 2011) because the performance assessments reported in the annual review account for projected future inventory.

[44] The radioactive material license provides for a total LLW volume of 26 million cubic feet with 5.6 million curies of total activity; the limits for Class A containerized, Class B, and Class C waste are 8,100,000 cubic feet and 5,500,000 curies of total activity (TCEQ 2011). As shown in Table 5-4 below, the Vitrification Melter waste package would constitute small fractions of these limits.

0.0064 millirem per year to an adjacent resident, a small fraction of the 25 millirem per year limit[45]. The updated performance assessment (WCS 2011) was developed in compliance with a license condition that requires WCS to prepare an updated performance assessment prior to accepting waste for disposal and annually thereafter to demonstrate that performance objectives will be met (TCEQ 2011).[46]

5.2.3 Protection of Individuals from Inadvertent Intrusion

DOE requirements of DOE Manual 435.1-1, Section IV.P(2)(h), for protection of individuals from inadvertent intrusion read as follows:

> "For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 millirem (1 mSv) in a year and 500 millirem (5 mSv) total effective dose equivalent excluding radon in air."

As discussed in Appendix C, NRC in 10 CFR 61.42 sets forth the following requirements:

> "Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

As discussed further in Appendix C, DOE's dose limits for the hypothetical human intruder are more stringent than the dose limit used for NRC's performance objective at 10 CFR 61.42. Typically, NRC applies a whole-body dose limit of 500 mrem per year to assess compliance with the requirement at 10 CFR 51.42 (NRC 2007) , whereas DOE imposes a 100 mrem per year and 500 mrem per year total effective dose equivalent (excluding radon in air) for chronic and acute inadvertent human intruder exposures, respectively.

The State of Texas requirement for protection of individuals from inadvertent intrusion tracks the NRC requirement as shown in Appendix C.[47]

Assessment of Area 5 Performance

The report of the 2010 annual review of performance assessments and composite analyses for the Area 3 and Area 5 radioactive waste management sites at the Nevada National Security Site (NST 2011) demonstrates that there is a reasonable expectation that the Area 5 Radioactive Waste Management Site will meet the DOE intruder dose criteria. The scenarios evaluated as described in this report were as follows:

---

[45] This estimate is for the Federal Facility Waste Disposal Facility Canister Disposal Unit, where the Vitrification Melter waste package would be disposed of.

[46] Because the WCS facility is not yet in operation, this waste-incidental-to-reprocessing evaluation does not quantitatively estimate the potential additional dose impact from disposal of the Vitrification Melter in the WCS facility in the same manner as was done for the Nevada National Security Site Area 5 Radioactive Waste Management Site (i.e., by evaluating the inventory to be disposed of, with and without the Vitrification Melter.)

[47] Texas imposed on WCS a 25 mrem per year limit for intruder doses (WCS 2011). This matter and the comparability of DOE, NRC, and State of Texas regulatory provisions for imposing additional requirements on LLW disposal is discussed further in Appendix C.

- The drilling worker intruder scenario, with the predicted peak annual acute dose 1,000 years after facility closure, the end of the compliance period; and
- The home construction intruder scenario, with the predicted peak annual acute dose 1,000 years after facility closure.

Chronic intruder scenarios are no longer reported in the Annual Summary Report for the Area 5 Radioactive Waste Management Site because chronic intrusion would be unlikely due to a change in the institutional control policy made in 2008. The planned land-use restrictions will prohibit public access to groundwater for 1,000 years within the compliance boundary negotiated with the State of Nevada, which is to include the Area 5 Radioactive Waste Management Site.[48] (NST 2011)

Estimated Impact of the Vitrification Melter

Table 5-2 shows the estimated impacts of disposal of the Vitrification Melter in the Nevada National Security Site Area 5 Radioactive Waste Management Site to a hypothetical human intruder.

Table 5-2. Estimated Acute Dose Impacts to an Inadvertent Intruder Associated With Vitrification Melter Disposal (mrem/y)[(1)]

Limiting Pathway/Scenario (Probability)	Annual Limit (Acute Exposure)	Estimated Maximum		
		Area 5 (Without Melter)	Area 5 (With Melter)	Difference Due to Melter
Inadvertent Intruder: Construction (mean)[(2)]	500	280	280	negligible
Inadvertent Intruder: Construction (95th Percentile)	500	600	600	negligible

NOTES: (1) From DOE 2010b. The limiting pathways/scenarios are those estimated to produce the highest dose. In each case, the peak annual doses were estimated to occur 1,000 years after facility closure.
(2) The mean estimate is used for DOE Manual 435.1-1 compliance purposes[49]. The 95th percentile estimate provides additional information.

The information in Table 5-2 shows that disposal of the Vitrification Melter in the Area 5 facility is estimated to have a negligible impact on facility performance insofar as inadvertent intruders are concerned.

Assessment of WCS Federal Facility Waste Disposal Facility Performance

The updated WCS performance assessment (WCS 2011) provides the following estimated doses to inadvertent intruders:

- A maximum acute dose of 1.4 millirem per year to the intruder driller, and
- A maximum chronic dose of 0.62 millirem per year to the intruder resident farmer.

These estimated doses are well below the 25 millirem annual limit[50].

[48] Chronic intruder doses continue to be calculated by the performance assessment model but are no longer reported in the Annual Summary Report for reasons specified in that report (DOE 2010b). This practice is consistent with Section IV.P(2) of DOE Manual 435.1-1, which provides for considering the likelihood of inadvertent intruder scenarios in interpreting the results of the analyses if adequate justification is provided.

[49]The use of the mean also is consistent with NRC guidance in NUREG-1573, *A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal*, *Recommendations of NRC's Performance Assessment Working Group* (NRC 2000).

5.2.4 Protection of Individuals During Operations

The DOE requirements in DOE Manual 435.1-1, Section I.E(13), for protection of individual during operations read as follows:

> "Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR Part 835, *Occupational Radiation Protection,* and DOE [Order] 5400.5 [now DOE Order 458.1], *Radiation Protection of the Public and the Environment*."

> As discussed in Appendix C, NRC in 10 CFR 61.43 provides similar, comparable requirements:

> "Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in 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 State of Texas requirements track the NRC requirements, as discussed in Appendix D.

Comparability of DOE, NRC, and State of Texas Requirements

DOE's requirements and dose limits for protection of individuals during operations in 10 CFR Part 835 and DOE Order 5400.5 (now DOE Order 458.1), cross referenced in Section I.E(13) of DOE Manual 435.1-1, are comparable to the relevant NRC standards for radiation protection in 10 CFR Part 20, as cross referenced in the NRC performance objective at 10 CFR 61.43. For example, both DOE and NRC limit occupational dose to a total effective dose equivalent of 5 rem per year and doses to the public from operations to 0.1 rem per year, as further discussed in Appendices C and D. As explained in Appendix D, the State of Texas requirements and dose limits for protection of individuals during operations mirror the NRC requirements and dose limits.

DOE's regulatory and contract requirements for DOE facilities and activities ensure compliance with DOE's regulations at 10 CFR Part 835 and relevant DOE Orders that establish dose limits for the public and the workers during operations. Appendix D provides additional details concerning the DOE requirements for protecting individuals during operations.

In addition, DOE's regulation at 10 CFR 835.101(c) requires that each radiation protection program include formal plans and measures for applying the ALARA (as low as is reasonably achievable) approach to occupational exposures.

Protection of Individuals During Operations at the WVDP and the Nevada National Security Site

The DOE requirements apply to the workers at the WVDP who will be involved with grouting and preparing the Vitrification Melter and its waste package for disposal, as well as to the public at the site. The DOE performance requirements also apply to the workers at the Nevada National Security Site who would handle disposal of the Vitrification Melter waste package and to the public at that site.

Both the WVDP and the Nevada National Security Site maintain radiation protection programs based on the requirements of 10 CFR 835. These programs also comply with various DOE

[50] These estimates are for the Federal Facility Waste Disposal Facility Canister Disposal Unit where the Vitrification Melter waste package would be disposed of.

directives (including DOE Order 5400.5 [now DOE Order 458.1], other Orders, policies, guides, and manuals), and supplemental technical standards.

The WVDP radiological protection program and these measures are described in the WVDP Radiological Controls Manual (WVES 2010a). The Nevada National Security Site radiological protection program and ALARA measures are described in the Nevada National Security Site Radiological Control Manual (NST 2010)

Gamma radiation levels on the sides of the Vitrification Melter waste package range from approximately 2 to 6 millirem per hour (WVNSCO 2004); levels will be lower after the waste package is grouted. Workers involved with handling of the waste package have received doses below the WVDP administrative control level of 500 mrem per year, which is 10 percent of the annual DOE occupational dose limit of 5,000 mrem per year in 10 CFR 835, Subpart C. The radiation doses to workers to be involved with preparation of the Vitrification Melter waste package for shipment will be minimized by compliance with the WVDP radiological control program and the associated ALARA processes.

Compliance with the radiological control program requirements and the ALARA processes will provide reasonable expectation that WVDP worker doses will be well below the 500 mrem per year limit, especially considering the low radiation levels on the outside of the Vitrification Melter waste package and the short duration of the work to prepare the waste package for shipment. Furthermore, the work associated with preparing the waste package for shipment is similar in nature to other WVDP waste management work for which worker doses have been maintained ALARA and well below the 500 mrem annual limit.

Compliance with the WVDP radiological control program requirements and the associated ALARA processes will also ensure that potential exposures to the public from onsite work related to preparing the Vitrification Melter for shipment are well below the applicable limit[51]. This work will be performed within a radiologically controlled area within the WVDP security fence. Past WVDP experience with similar waste management work indicates that potential doses to the public will be very low. In 2010, for example, a year in which similar waste management work was performed by the WVDP, the estimated dose to a maximally exposed offsite individual from WVDP airborne radioactivity emissions was 0.0017 mrem (CHBWV 2011). The airborne pathway is the only pathway of interest for potential exposure to a member of the public from onsite work to prepare the Vitrification Melter for shipment. Such factors provide reasonable expectation that doses to the public from preparing the waste package for shipment will continue to be far below the applicable limit.

Doses to workers at the Nevada National Security Site who would be involved with handling the Vitrification Melter waste package to dispose of it in the Area 5 Radioactive Waste Management Site would be minimized by compliance with that site's radiological control program and the associated ALARA processes. Compliance with the radiological control program requirements, following ALARA processes, the low radiation levels on the waste package, and the short duration of the work to place it in the disposal facility provides reasonable expectation that worker doses will be ALARA.

Potential exposures to members of the public associated with onsite handling of the Vitrification Melter at the Nevada National Security Site are also expected to be very low. Operations to dispose

[51] The applicable limit is 10 mrem per year for exposure to a member of the public from air emissions as specified in the U.S. Environmental Protection Agency requirement in 40 CFR 61.92, with which DOE complies.

of the shielded Vitrification Melter waste package would be of short duration, would take place in a radiologically controlled area with no routine public access, and would take place at the isolated government-controlled Nevada National Security Site.

Protection of Individuals During Operations at the WCS Facility

If DOE were to transport the Vitrification Melter waste package to the WCS facility for disposal, individuals would be protected during operations in a manner similar to the Nevada National Security Site. As noted above, the applicable State of Texas dose standards mirror those of NRC. WCS is required to comply with the requirements of Texas Administrative Code, Title 30, Part 1, Chapter 336, Subchapter D, *Standards for Protection Against Radiation*, which provide for a comprehensive program to protect individuals and the public during waste disposal site operations.

5.2.5 Stability of the Disposal Site After Closure

The DOE requirements in DOE Manual 435.1-1, Sections IV.Q(1)(a) and (b) and IV.Q(2)(c), for stability of the disposal site after closure are expressed as follows:

> <u>"Disposal Site Stability (DOE Manual 435.1, Section IV.Q(1)(a) and (b))</u>. A preliminary closure plan shall be developed and submitted to Headquarters for review with the performance assessment and composite analysis. The closure plan shall be updated following issuance of the disposal authorization statement to incorporate conditions specified in the disposal authorization statement. Closure plans shall:
>
> (a) Be updated as required during the operational life of the facility.
>
> (b) Include a description of how the disposal facility will be closed to achieve long-term stability and minimize the need for active maintenance following closure and to ensure compliance with the requirements of DOE 5400.5 [now DOE Order 458.1], *Radiation Protection of the Public and the Environment*."
>
> <u>"Disposal Facility Closure (DOE Manual 435.1, Section IV.Q(2)(c))</u>. Institutional control measures shall be integrated into land use and stewardship plans and programs, and shall continue until the facility can be released pursuant to DOE Order 5400.5 [now DOE Order 458.1], *Radiation Protection of the Public and the Environment*."

As discussed in Appendix C, NRC requirements in 10 CFR 61.44 set forth similar, comparable requirements:

> "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 explained in Appendix C, the State of Texas requirements for the stability of the disposal site after closure mirror the NRC requirements.

DOE has developed a preliminary closure plan for the Area 5 Radioactive Waste Management Site in accordance with the DOE requirements. This plan entails use of 2.5-meter (8.2-foot) thick closure cover, consistent with assumptions used in the performance assessment (NST 2011), although the closure cover thickness over the top of the buried Vitrification Melter waste package would more than 10 feet (three meters)[52]. The plan will ensure that the applicable requirements of DOE Order 458.1 will be met following closure of the Area 5 Radioactive Waste Management Site, which is currently planned for 2028. The applicable requirements of DOE Order 458.1 include the public dose limit of 100 mrem per year total effective dose. The airborne emissions limit of 10 mrem per year effective dose equivalent at 40 CFR 61.92 also applies to emissions of radionuclides from DOE facilities to the ambient air.

The WCS license application (WCS 2007) – in Volume 2, Section 6, *Closure* – describes features of the planned closure system for that facility to meet the State requirements for stability of the disposal site after closure. These features include a depth of disposal significantly greater than five meters (16.4 feet) for all waste.

5.3 The Vitrification Melter Meets Disposal Site Waste Acceptance Criteria

As noted previously, the WVDP has been accepted as an approved waste generator by the Nevada National Security Site and has shipped LLW there for disposal on numerous occasions.

To protect workers, the public, and the environment, DOE establishes waste acceptance criteria for its LLW disposal facilities, which, among other things, provide limits on the radionuclides that may be disposed of at the facility, based on a performance assessment for the facility. As discussed in Section 5.2, the performance assessment (and updates) for each LLW disposal facility provides reasonable expectation that DOE's performance objectives in Chapter IV of DOE manual 435.1-1 will not be exceeded. Accordingly, disposal of the Vitrification Melter in compliance with the waste acceptance criteria for the Nevada National Security Site Radioactive Waste Management Site will provide reasonable expectation that disposal will not exceed the DOE performance objectives in Chapter IV of DOE Manual 435.1-1.

To help establish the relationship between the waste acceptance criteria and performance assessments of the waste disposal sites, this subsection provides a summary of disposal site waste acceptance criteria and explains how the Vitrification Melter meets these criteria. It also addresses meeting the WCS waste acceptance criteria.

5.3.1 Nevada National Security Site Waste Acceptance Criteria

For its LLW disposal facilities, DOE provides formal waste acceptance criteria that comprise the technical and administrative requirements that a waste must meet in order for it to be accepted at the disposal facility (DOE Manual 435.1-1, Attachment 2). The Nevada National Security Site

[52] NRC at 10 CFR 61.52(a)(2) provides that "Wastes designated as Class C pursuant to § 61.55, must be disposed of so that the top of the waste is a minimum of 5 meters below the top surface of the cover or must be disposed of with intruder barriers that are designed to protect against an inadvertent intrusion for at least 500 years." This requirement, which does not apply to DOE LLW waste disposal facilities, is among the NRC rules at 10 CFR 61 that were developed for nationwide application at a wide variety of facilities with geological and environmental settings that are largely unknown until a specific facility is proposed.

DOE uses a different approach because its established disposal sites have well known geological and environmental characteristics. This approach involves setting basic performance objectives, which are comparable to those of NRC as demonstrated in Appendix C, and demonstrating with a site-specific performance assessment that the total disposal system (site, design features, waste form, radionuclide content operating practices, and closure plans) provides reasonable expectation that the performance objectives will be met. (DOE Guide 435.1-1, Appendix A, *Basis for Regulation of Low-Level Waste*)

Table 5-1 and Table 5-2 show that disposal of the Vitrification Melter waste package at the Nevada National Security Site Area 5 facility would have a negligible impact on disposal site performance.

provides specific radionuclide waste acceptance criteria for LLW (DOE 2011a) that are expressed primarily in terms of waste package activity limitations based on plutonium 239 equivalent grams (PE-g).

This quantity relates the amount of a particular radionuclide to plutonium 239. Appendix B to the criteria document (DOE 2011a) contains a table of PE-g radionuclide conversion factors. These conversion factors relate amounts of an individual radionuclide to plutonium 239. For example, the conversion factor for cesium 137 is 2.72E-14 PE-g/Bq or approximately 1.0E-03 PE-g/Ci.

The Nevada National Security Site waste package limit for a single Department of Transportation Type A drum is 300 PE-g total. The limit for a strong-tight container such as an intermodal shipping container is also 300 PE-g total. An additional limitation of 2,000 PE-g per individual shipment also applies, except for Type B shipping containers in cases where the containers themselves are to be disposed of. (DOE 2011a)

Action levels for individual radionuclides are also provided in the Nevada National Security Site waste acceptance criteria to identify radionuclides that must be reported on two key documents to be submitted by the waste generator: the Waste Profile and the Package Storage and Disposal Request. These action levels are used to identify waste streams that may require special consideration with regard to meeting the waste acceptance requirements.[53,54]

The criteria require radionuclides known or reasonably expected to be present in a waste stream to be reported in the Package Storage and Disposal Request and the Waste Profile as follows:

(1) When the activity concentration in the final waste form exceeds one percent of a specified reporting action level specified in Table E-1 of the Waste Acceptance Criteria (DOE 2011a),

(2) Any alpha-emitting transuranic radionuclide with half-life over 20 years that exceeds 10 pCi/g, or

(3) Any radionuclide whose concentration exceeds one percent of the total activity concentration.

[53] Radionuclides whose concentrations exceed one percent of the action level are required to be specifically reported on the Package Storage and Disposal Request and the Waste Profile and require rigorous characterization (DOE 2011a).

[54] Other requirements address transuranic activity (the concentrations of alpha-emitting transuranic radionuclides with half-lives over 20 years which must not exceed 100 nCi/g) and the amounts of fissile material present. The DOE Manual 435.1-1 definition of transuranic waste includes alpha-emitting transuranic radionuclides with half-lives greater than 20 years. In the 10 CFR 61.55 definition of low-level waste, NRC includes a limit for alpha-emitting transuranic radionuclides with half-lives greater than five years. In practice, Cm-244 (with its 18.1 year half-life) is the only radionuclide covered by 10 CFR 61.55 that is not addressed as a transuranic radionuclide by DOE Manual 435.1-1. The 10 CFR 61.55 requirements also include specific limits for Pu-241 and Cm-242, because these two radionuclides decay to alpha-emitting transuranic isotopes with half-lives greater than five years (i.e., Am-241 and Pu-238, respectively). The definition of transuranic waste in DOE Manual 435.1-1 is based on EPA's regulations at 40 CFR Part 191 and the Waste Isolation Pilot Plant Land Withdrawal Act, which identify transuranic radionuclides based on concentrations and half-lives greater than 20 years. Consequently, these three radionuclides (Pu-241, Cm-242, and Cm-244) are not transuranic radionuclides under concentration limits of Table E-1 of the Nevada National Security Site Waste Acceptance Criteria (DOE 2011a). Table E-1 provides radionuclide action levels for waste characterization and reporting purposes.

The Nevada National Security Site Waste Acceptance Criteria were developed to ensure protection of public health and the environment both during ongoing operations of the waste disposal sites and after these sites are closed. The acceptability of radionuclide concentration limits specified to these ends is verified by DOE in a comprehensive performance assessment program, as noted previously.

5.3.2 The Vitrification Melter Waste Package

Determining whether a waste package will meet the Nevada National Security Site radionuclide limits involves: (1) identifying the activity of each reportable radionuclide present, (2) converting this activity to PE-g, (3) summing all the individual PE-g values, and (4) comparing this total to the Nevada National Security Site waste acceptance criteria individual package limit in PE-g. DOE has compared characteristics of the Vitrification Melter waste package to the Nevada National Security Site waste acceptance criteria and determined that it will meet these criteria. For example, the Vitrification Melter contained an estimated 78.5 PE-g as of October 1, 2004 based on the radionuclide estimates in Table 2-2, well below the individual package limit of 300 PE-g (WSMS 2008)[55].

5.3.3 The Vitrification Melter Meets the Nevada National Security Site Waste Acceptance Criteria

As noted previously, the WVDP has been accepted as an approved waste generator by the Nevada National Security Site and has shipped LLW there for disposal on numerous occasions.

Because of the established relationship between the waste acceptance criteria and performance assessments of the waste disposal sites, satisfying the waste acceptance criteria indicates compliance with the disposal site performance assessment and, hence, with the DOE performance objectives.

For its LLW disposal facilities, DOE provides formal waste acceptance criteria that comprise the technical and administrative requirements that a waste must meet in order for it to be accepted at the disposal facility (DOE Manual 435.1-1, Attachment 2). These criteria for the Nevada National Security Site are contained in its Waste Acceptance Criteria document (DOE 2011a).

As described in Section 2, DOE has packaged the Vitrification Melter in a robust Type IP-2 container. Prior to shipment for disposal, internal void spaces will be filled with low-density cellular concrete. The 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 the container void spaces.

A Waste Profile Sheet for disposal of the Vitrification Melter waste package was submitted to the Nevada National Security Site for formal review and approved as specified in the waste acceptance criteria, as noted previously. Table 5-3 compares the radionuclide concentrations in the Vitrification Melter with the radionuclide action levels for waste characterization and reporting provided in the Nevada National Security Site Waste Acceptance Criteria (DOE 2011a).

[55] The waste profile shows a higher estimate of 97.7 PE-g, which was developed using the "high activity" radionuclide concentration estimates, which provide the upper bound of the estimated inventory.

Radionuclides with concentrations exceeding one percent of the action level are highlighted in the table.[56]

Table 5-3. Vitrification Melter Radionuclide Concentrations in Bq/m^3(1)

Nuclide	Action Level(2)	Melter(3)	Nuclide	Action Level(2)	Melter(3)
H-3	6.2E+11	1.38E+04	U-232	4.3E+10	8.80E+07
C-14	5.4E+15	3.97E+07	U-233	8.2E+10	3.59E+07
K-40	9.4E+10	1.43E+08	U-234	1.3E+10	1.71e+07
Mn-54	NA	1.49E+08	U-235	1.1E+10	6.56E+05
Fe-55	NA	6.69E+06	U-236	2.8E+11	1.97E+06
Ni-59	1.7E+14	3.64E+05	U-238	3.5E+11	3.93E+06
Co-60	1.6E+12	1.46E+08	Np-237	3.4E+10	1.08E+07
Ni-63	3.2E+14	1.77E+09	Pu-238	1.8E+12	1.19E+09
Sr-90	4.3E+11	4.32E+11	Pu-239	5.1E+11	2.78E+08
Zr-95	NA	2.89E+09	Pu-240	5.2E+11	2.11E+08
Tc-99	3.2E+09	1.94E+07	Pu-241	5.8E+12	5.46E+09
I-129	3.4E+09	2.77E+05	Pu-242	3.7E+11	3.97E+04
Cs-137	2.5E+11	7.52E+12	Am-241	1.7E+11	5.25E+09
Pm-147	NA	4.95E+07	Am-243	5.8E+10	6.30E+07
Eu-154	1.7E+12	2.12E+09	Cm-242	NA	1.28E+08
Th-228	4.3E+13	7.13E+07	Cm-243	8.3E+11	2.93E+07
Th-229	2.8E+10	2.8E+02(4)	Cm-244	3.4E+12	7.67E+08
Th-230	6.0E+07	6.36E+05	Cm-245	4.6E+10	1.92E+07
Th-232	8.1E+09	6.99E+05	Cm-246	9.2E+10	3.12E+06

NOTES: (1) To convert Bq/m^3 to pCi/L multiply by 0.027.
(2) From the Table E-1 of the Waste Acceptance Criteria (DOE 2011a).
(3) From the Vitrification Melter waste profile technical basis document (WVES 2010b). The waste profile provides a shorter list of radionuclides focused on those that exceed one percent of the action level.
(4) Th-229 concentration estimated using Th-229 to Th-230 ratio from high-level waste sludge solids characterization (Rykken 1986, Table 22).

Table 5-4 shows that the estimated concentration of Cs-137 in the Vitrification Melter exceeds the action level for this radionuclide. However, this would have a negligible impact on performance of the Nevada National Security Site Area 5 Radioactive Waste Disposal Site as shown by the results of the dose impact analyses provided in Tables 5-1 and 5-2 (DOE 2010b).

5.4 Meeting WCS Waste Acceptance Criteria

The WCS waste acceptance criteria document (WCS 2008) addresses matters such as operations and regulatory parameters, pre-shipment requirements, documentation, and

[56] As noted previously, the waste acceptance criteria document (DOE 2011a) requires that activity concentrations of the radionuclides in the final waste form exceeding one percent of the action level in Table E-1 of that document receive rigorous waste characterization and be reported on the package storage and disposal request and the waste profile. The Vitrification Melter was rigorously characterized by the WVDP.

transportation. It provides various forms including a waste profile sheet. Unlike the Nevada National Security Site waste acceptance criteria, the WCS waste acceptance criteria document does not provide numerical radionuclide concentration action levels. However, the separate WCS Waste Acceptance Plan (WCS 2009) provides additional information related to the waste acceptance process, including waste form requirements and a description of the generator and waste approval processes.

The WCS license (TCEQ 2011) contains additional requirements related to waste disposal, including total waste volume limitations and total activity limitations for certain radionuclides. Table 5-4 shows representative requirements compared to the related parameters for the Vitrification Melter waste package.

TABLE 5-4. Key WCS Federal Facility Waste Disposal Facility License Requirements

Requirement (Section)(1)	License Limit(1)	Vitrification Melter	Melter % of Limit
Total waste volume, ft^3 (§7.B)	8,100,000	2331(2)	0.029
Total activity, curies (§7.B)	5,500,000	4570(3)	0.083
Total C-14, curies (§5.D)	180	0.0212(3)	0.012
Total Tc-99, curies (§5.D)	35	0.0111(3)	0.032
Total I-129, curies (§5.D)	0.15	0.00564(2)	3.8

NOTES: (1) From the WCS license (TCEQ 2011) with the associated section numbers and limits. The §7.B limits are for Class A containerized, Class B, and Class C LLW, collectively.
(2) From the Vitrification Melter Waste Profile Sheet for the Nevada National Security Site (WVES 2010b).
(3) From Table 2-2.

Table 5-4 shows that the volume of the Vitrification Melter waste package is a small fraction of the WCS federal facility waste disposal facility capacity limit and that the total activity and the activity of the license-limited radionuclides in the waste package are small fractions of the WCS limits.

If DOE were to elect to dispose of the Vitrification Melter at the WCS federal facility waste disposal facility, DOE would confirm that the waste package meets the waste acceptance criteria for that facility prior to shipment. DOE would follow the WCS process and submit all of the necessary supporting information, such as the Waste Profile Form[57].

[57] Because the WCS facility is licensed to accept Class C LLW, DOE would expect that the Vitrification Melter waste package would be approved for disposal. However, because the Vitrification Melter is not among the planned waste streams identified in the documents supporting the WCS license application, a license amendment may be necessary to obtain approval from the regulator for disposal of the Vitrification Melter waste package.

6.0 The Waste Does Not Exceed Class C Concentration Limits and Will Be Managed in Accordance With DOE Requirements as LLW

Section Purpose

The purpose of this section is to demonstrate that the Vitrification Melter waste package will be in a solid physical form, will not exceed Class C concentration limits, and will be managed in accordance with DOE requirements as low-level radioactive waste as applicable.

Section Contents

This section provides information showing that the grouted Vitrification Melter waste package will be in a solid physical form, will not exceed the concentration limits for Class C low-level waste in 10 CFR 61.55, and will be managed and disposed of as low-level waste in accordance with DOE requirements.

Key Points

- The grouted Vitrification Melter waste package will be in a solid physical form.
- The radioactivity in the Vitrification Melter waste package will not exceed Class C concentration limits.
- The Vitrification Melter waste package will be managed and disposed of at an offsite low-level radioactive waste disposal facility in accordance with applicable requirements for low-level waste.

The third and final criterion of DOE Manual 435.1-1, Section II.B(2)(a) to be demonstrated is:

> "[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 DOE Manual 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 low-level waste as set out in 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize."

As explained previously, the Vitrification Melter has been packaged in a shielded IP-2 shipping container and voids in the Vitrification Melter and the space between the Vitrification Melter and the inside of the container will be filled with low-density cellular concrete prior to shipment for disposal. Hence, the Vitrification Melter waste package will be in a solid physical form. (The Melter itself is already in a solid physical form, as noted previously.)

Because the Vitrification Melter contains a mixture of radionuclides, the total concentration is determined by the sum of the fractions rule, as specified in NRC's regulations at 10 CFR 61.55(a)(7) (§336.362(a)(7) of the Texas Administrative Code parallels the NRC's regulations). Additionally, because the radionuclide mixture contains some long-lived radionuclides that are listed on Table 1 of 10 CFR 61.55 (reproduced in Table 4-1 of this waste-incidental-to-reprocessing evaluation), and some short-lived radionuclides that are listed on Table 2 of 10 CFR 61.55

(reproduced in Table 4-2 of this waste-incidental-to-reprocessing evaluation), waste classification would be determined as specified in 10 CFR 61.55(a)(5), which states:

"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 that determined by the concentration of nuclides 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."

Radiological characterization of the Vitrification Melter before packaging was as described in Section 2.5.3. Using the results of that characterization, Table 6-1 below shows that the Melter does not exceed Class C concentration limits.

Table 6-1. Vitrification Melter Waste Concentration Results(1)

Isotope	Estimated Activity (Ci)	Class C Limit (Ci/m^3)	Class C Limit (nCi/g)	Melter Concentration (Ci/m3)	Melter Concentration (nCi/g)	Table 1(2) Fraction	Table 2(2) Fraction
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	NA		1.00E+02		4.67E-04	4.67E-02	
Am-241	3.00E+00		1.00E+02		6.24E+01	6.24E-01	
Am-242m	NA		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	NA		1.00E+02		2.25E-01	2.25E-03	

Table 6-1. Vitrification Melter Waste Concentration Results(1) (continued)

Isotope	Estimated Activity (Ci)	Class C Limit (Ci/m^3)	Class C Limit (nCi/g)	Melter Concentration (Ci/m3)	Melter Concentration (nCi/g)	Table 1(2) Fraction	Table 2(2) Fraction
Cm-246	NA		1.00E+02		3.66E-02	3.66E-04	
					Sums of fractions(3)	9.51E-01	4.59E-02

LEGEND: NA = not available. (The waste profile document (WVES 2010b) identifies concentrations only for these radionuclides.)

NOTES: (1) From WSMS 2008, with the calculations based on the Vitrification Melter weight and size, except for Pu-242, Cm-245, and Cm-246, for which concentrations are from the waste profile document (WVES 2010b), and Am-242m, which was estimated using the ratio of Am-242m to Am-241 in Rykken 1986. 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 106,000 pounds and the volume used was 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 375 cubic feet. The resulting sums for fractions were 9.52E-01 for Table 1 and 9.19E-02 for Table 2 (WSMS 2008). Note that void spaces inside the Vitrification Melter comprise approximately 378 cubic feet, so the sensitivity analysis shows that taking void spaces into account would have had a negligible impact on the waste classification.

This table shows that the sum of fractions for both the 10 CFR 61.55 Table 1 (long-lived) and Table 2 (short-lived) radionuclides is below 1.0, and thus the Vitrification Melter does not exceed concentration limits for Class C LLW. The calculations were performed using the weight and size of the Vitrification Melter itself; neither the grout nor the shipping container was considered[58].

Note that the sum-of-fractions estimates in Table 6-1 are inherently conservative because the radionuclide inventories were based on a date of September 30, 2004. Radioactive decay since that time has reduced the inventories of the shorter-lived radionuclides such as Sr-90, Cs-137, Pu-238, and Cm-244 thereby reducing the Table 1 and Table 2 sums of fractions values to approximately 8.7E-01 and 3.5E-02 respectively.

For comparison purposes, DOE calculated the waste classification sums of fractions using the alternate estimate for residual radioactivity in the Vitrification Melter that was described in Section 2.5.3. The resulting sums of fractions were 0.470 for Table 1 and 0.0201 for Table 2, showing that the results in Table 6-1 above are conservative (DOE 2011c).

As discussed previously, this waste may be transported to the Nevada National Security Site Area 5 Radioactive Waste Management Site for disposal. At the Nevada National Security Site, the Vitrification Melter waste package would be disposed of as LLW and managed in accordance with DOE requirements for LLW disposal in Chapter IV of DOE Manual 435.1-1. The required Waste Profile has been developed by DOE in accordance with the Nevada National Security Site Waste Acceptance Criteria (WVES 2010b). This Waste Profile has been formally approved by the Nevada

[58] In accordance with NRC concentration averaging guidance (NRC 1995a), it would have been acceptable to take into account the mass of the grout used to encapsulate the Vitrification Melter within the waste package, which is required for stabilization purposes. This approach would have resulted in smaller sums of fractions. In its Technical Evaluation Report (NRC 2011b), NRC also noted that "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) [site-specific (intruder) analysis considerations]. 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."

National Security Site, as noted previously (DOE 2010a), in case a final decision is made to send the Vitrification Melter waste package to that facility for disposal.

As noted previously, DOE may elect to send the Vitrification Melter to the commercially-operated WCS Federal Facility Waste Disposal Facility in Texas. As demonstrated earlier in this evaluation, the Vitrification Melter waste package will be in a solid physical form (with voids in the Vitrification Melter and the space between the Vitrification Melter and the inside of the container filled with low-density cellular concrete), and will not exceed Class C concentration limits in 10 CFR 61.55. In this regard, the Texas Administrative Code has similar requirements concerning waste stability and as little free standing liquid as possible.[59] In addition, the State of Texas Class C concentration limits mirror the concentration limits in 10 CFR 61.55; consequently, disposal of the Vitrification Melter waste package in the WCS Federal Facility Waste Disposal Facility would not exceed Class C LLW concentration limits set forth in the Texas Administrative Code.

DOE Order 435.1, *Radioactive Waste Management*, provides that requirements in the Order that duplicate or conflict with requirements of an applicable Agreement State do not apply to facilities and activities licensed by the Agreement State. Therefore, the provisions in Chapter IV of DOE Manual 435.1-1 concerning matters such as monitoring, waste acceptance criteria, performance assessments, composite analysis, disposal facility operations, disposal authorizations, institutional control, and disposal facility closure do not apply to the WCS facility; instead, these matters are governed by the State of Texas requirements and license conditions.

Accordingly, as demonstrated above, disposal of the Vitrification Melter waste package at the WCS Federal Facility Waste Disposal Facility would meet the third criterion of DOE Manual 435.1-1, Section II.B.2(a).

[59] Texas Administrative Code, Title 30, Part 1, §336.362, Appendix E.

7.0 CONSULTATION WITH NRC AND OPPORTUNITY FOR PUBLIC COMMENT

As explained previously, DOE consulted with NRC concerning this evaluation and made this evaluation available in draft form for public review and comment, including the States of Nevada and Texas where the Vitrification Melter waste package might be disposed of as LLW.

DOE considered the NRC's request for additional information (NRC 2011a) and technical evaluation report (NRC 2011b) resulting from the consultation and made changes to the draft evaluation as described in the response to the request for additional information (DOE 2011c). Comments from the public were also considered.

8.0 CONCLUSIONS

Based on information provided in the preceding sections of this evaluation, DOE concludes that the Vitrification Melter is not HLW based on the criteria of DOE Manual 435.1-1 and may be managed as LLW.

9.0 REFERENCES

Federal Statutes

West Valley Demonstration Project Act, Public Law 96-368 (S. 2443), of October 1, 1980.

Low-Level Radioactive Waste Policy Act of 1985.

Waste Isolation Pilot Plant Land Withdrawal Act of 1992 as amended in 1996, Public Law 102-579, Public Law 104-201, 110 Stat. 2422, 1996.

Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, Public Law 108-375, 118 Stat. 1811. 108th United States Congress, October 28, 2004.

Code of Federal Regulations and Federal Register Notices

10 CFR Part 20, *Standards for Protection Against Radiation*.

10 CFR 61.55, *Waste Classification*.

10 CFR 61, Subpart C, *Licensing Requirements for Land Disposal of Radioactive Waste, Performance Objectives*.

10 CFR Part 835, *Occupational Radiation Protection*.

40 CFR Part 61, *National Emission Standards for Hazardous Air Pollutants.*

40 CFR Part 191, *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes.*

47 FR 40705, *Long-Term Management of Liquid High-Level Radioactive Wastes Stored at the Western New York Nuclear Service Center, West Valley; Record of Decision*. U.S. Department of Energy, Washington, D.C., September 15, 1982.

65 FR 10061, *Record of Decision for the Department of Energy's Waste Management Program: Treatment and Disposal of Low-Level Waste and Mixed Low-Level Waste; Amendment of Decision for the Nevada Test Site*. U.S. Department of Energy, West Valley, New York, February 25, 2000.

70 FR 35073, *West Valley Demonstration Project Waste Management Environmental Impact Statement, Record of Decision*. U.S. Department of Energy, West Valley, New York, June 16, 2005.

75 FR 20582, *Record of Decision: Final Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center*. U.S. Department of Energy, West Valley, New York, April 20, 2010.

DOE Orders, Policies, Manuals, and Standards

DOE Order 435.1, *Radioactive Waste Management*, Change 1. U.S. Department of Energy, Washington, D.C., August 28, 2001.

DOE Order 458.1, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., June 6, 2011.

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., January 7, 1993.

DOE Manual 435.1-1, *Radioactive Waste Management Manual*, Change 1. U.S. Department of Energy, Washington, D.C., June 19, 2001.

DOE Guide 435.1-1, *Implementation Guide For Use With DOE M 435.1-1*. U.S. Department of Energy, Washington, D.C., July 9, 1999.

State Regulations

Texas Administrative Code, Title 30, Part 1, Chapter 336, *Radioactive Substance Rules* (http://info.sos.state.tx.us/pls/pub/readtac$ext.ViewTAC?tac_view=4&ti=30&pt=1&ch=336)

Other References

ASME 2004, *The Decommissioning Handbook*, Toboas, A., A, Moghissi, and T. LaGuardia, Ed. American Society of Mechanical Engineers and the American Nuclear Society, New York, New York, 2004.

Brooks 1993, *Slurry Fed Ceramic Melter Disassembly Report*, Brooks, R. West Valley Nuclear Services, West Valley, New York, April 23, 1993.

Cambridge 2004, *Cambridge Dictionary of American English*, Cambridge University Press, 2004 (http://dictionary.cambridge.org/define.asp?key=practical*2+0&dict=A).

CHBWV 2011, *West Valley Demonstration Project Annual Site Environmental Report for Calendar Year 2010*. CH2MHill-B&W West Valley, LLC, West Valley, New York, September 2011.

CLI-05-05, 2005, *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services), slip op.* at 11, January 18, 2005.

Cole, et al. 1995, *Comparison of Selected DOE and Non-DOE Requirements, Standards, and Practices for Low-Level Radioactive Waste Disposal*, DOE/LLW-225, Revision 0. Cole, L., D. Kudera, and W. Newberry, Idaho National Engineering Laboratory, Lockheed Idaho Technologies Company, Idaho Fall, Idaho, December 1995.

DesCamp and McMahon 1996, *Vitrification Facility at the West Valley Demonstration Project*, Topical Report DOE/NE/44139-77. DesCamp, V.A. and C.L. McMahon, West Valley Nuclear Services Company, July 1996.

DOE 1994, *Decommissioning Handbook*, DOE/EM-0142P. U. S. Department of Energy, Washington, D.C., March 1994.

DOE 1997, *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200-F. U.S. Department of Energy, Washington, D.C., May 1997.

DOE 1999, *Plutonium Recovery from Spent Fuel Reprocessing by Nuclear Fuel Services at West Valley, New York from 1966 to 1972, Revised*. U. S. Department of Energy, Washington, D.C., November 1999.

DOE 2003, *Final West Valley Demonstration Project Waste Management Environmental Impact Statement*, DOE/EIS-0337. DOE-West Valley Office, West Valley, New York, December 2003.

DOE 2006, *West Valley Demonstration Project Waste Management Environmental Impact Statement, Supplement Analysis*, DOE/EIS-0337-SA-01, Revised Final. U.S. Department of Energy, West Valley, New York, June 7, 2006.

DOE 2010a, Letter from DOE-Nevada Site Office to DOE-West Valley on Approval to Ship West Valley Environmental Services (WVES) Low-Level Radioactive Waste [the Vitrification Melter Waste Package] to the Nevada Test Site, dated July 6, 2010.

DOE 2010b, Letter from DOE-Nevada Site Office to DOE-West Valley on Special Performance Assessment (PA) Results Summary of Impacts of the Disposal of the West Valley Glass Melter in Area 5 Radioactive Waste Management Site (RWMS), WMP:6655:JC, dated July 29, 2010.

DOE 2011a, *Nevada Test Site Waste Acceptance Criteria*, DOE/NV-325-Rev. 8-01. U.S. Department of Energy, National Nuclear Security Administration, Nevada Site Office, Waste Management Project, Las Vegas, Nevada, January 2011.

DOE 2011b, *West Valley Demonstration Project, Draft Waste Incidental to Reprocessing Evaluation for the Vitrification Melter.* U.S. Department of Energy, West Valley New York, March 2011.

DOE 2011c, *Department of Energy Responses to the Nuclear Regulatory Commission's Request for Additional Information on the Draft Waste Incidental to Reprocessing Evaluation for the Vitrification Melter*. U.S. Department of Energy, Washington, D.C., June 16, 2011.

DOE 2011d, *Results Summary for the Area 5 Radioactive Waste Management Site Performance Assessment Model Version 4.110*, DOE-NV/25946—1267, dated July 20, 2011.

DOE and NRC 1981, *West Valley Demonstration Project Memorandum of Understanding Between the U.S. Department of Energy and the U.S. Nuclear Regulatory Commission*. September 23, 1981.

DOE and NYSERDA 1981, *Cooperative Agreement between the United States Department of Energy and New York State Energy Research and Development Authority on the Western New York Nuclear Service Center at West Valley, New York.* Signed November 3, 1980, amended September 18, 1981.

DOE and NYSERDA 2010, *Final Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center*, DOE/EIS-0226-D. U.S. Department of Energy and the New York State Energy Research and Development Authority, West Valley, New York, January 2010.

Eisenstatt 1986, *Description of the West Valley Demonstration Project Reference High-Level Waste Form and Canister*, DOE/NE/44139-26. Eisenstatt, L. R., U.S. Department of Energy, West Valley, New York, 1986.

Fowler 1930, *A Dictionary of Modern English Usage*. Fowler, H., 1930.

HEW 1970, *Radiological Health Handbook*, Revised Edition. U.S. Department of Health, Education, and Welfare, Public Health Service, Washington, D.C., January 1970.

Kurasch 2011, *Calculation of the Results of Processing Additional Flush Solutions in the WVDP Vitrification Melter Based on Actual Vitrification Records Data*. Kurasch, D., West Valley Environmental Services, West Valley New York, June 2011.

Kelly and Meess 1997, *THOREX Processing and Zeolite Transfer for High-Level Waste Stream Processing Blending*, Topical Report DOE/NE/44139-82. Kelly, S. and D.C. Meess, West Valley Nuclear Services Company, West Valley, New York, July 1997.

Lachapelle 2003, *Vitrification Facility (VF) Radioisotope Inventory Report*, RIR-403-010, Revision 1. Lachapelle, E., West Valley Nuclear Services Company, West Valley, New York, June 17, 2003.

NRC 1982, *Final Environmental Impact Statement on 10 CFR 61 "Licensing Requirements for Land Disposal of Radioactive Waste*," NUREG-0945. U.S. Nuclear Regulatory Commission, Washington, D.C., November 1982.

NRC 1995a, *Final Branch Technical Position on Concentration Averaging and Encapsulation*. U.S. Nuclear Regulatory Commission, Washington, D.C., January 17, 1995.

NRC 1995b, *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission*, NUREG/BR-0058, Revision 2. U.S. Nuclear Regulatory Commission, Washington, D.C., November 1995.

NRC 1997, *Regulatory Analysis Technical Evaluation Handbook*, NUREG/BR-0184. U.S. Nuclear Regulatory Commission, Washington, D.C., January 1997.

NRC 2000, *A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities*, *Recommendations of NRC's Performance Assessment Working Group*, NUREG-1573. U.S. Nuclear Regulatory Commission, Washington, D.C., October 2000.

NRC 2002, *Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement*. U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 67, No. 5003, February 1, 2002.

NRC 2003, *U.S. Nuclear Regulatory Commission Implementation Plan for the Final Policy Statement on Decommissioning Criteria for the West Valley Demonstration Project at the West Valley Site*. U.S. Nuclear Regulatory Commission, Division of Waste Management, Office of Nuclear Materials Safety and Safeguards, Washington, D.C., May 2003.

NRC 2004, Letter from R.R. Bellamy (NRC) to T.J. Jackson (DOE-West Valley), U.S. Nuclear Regulatory Commission Monitoring Visit 2004-02. U.S. Nuclear Regulatory Commission Region, 1 King of Prussia, Pennsylvania, December 9, 2004.

NRC 2005, *U.S. Nuclear Regulatory Commission Technical Evaluation Report for the U.S. Department of Energy Savannah River Site Draft Section 3116 Waste Determination for Salt Waste Disposal*. U.S. Nuclear Regulatory Commission, Washington, D.C., December 2005.

NRC 2007, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations*, NUREG-1854. Draft Final Report, U.S. Nuclear Regulatory Commission, Washington, D.C., August 2007.

NRC 2011a, *Request for Additional Information on the Draft Waste Incidental to Reprocessing Evaluation for the Vitrification Melter*. Letter from L.W. Camper, U.S. Nuclear Regulatory Commission to F. Marcinowski. U.S. Department of Energy, Washington, D.C., May 10, 2011.

NRC 2011b, *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, Washington, D.C., September 30, 2011.

NST 2010, *Nevada Test Site Radiological Control Manual*, DOE/NV/25946—801, Revision 1. National Security Technologies LLC, Las Vegas, Nevada, February 2010.

NST 2011, *2010 Annual Summary Report for the Area 3 and Area 5 Radioactive Waste Management Sites at the Nevada Test Site Nye County, Nevada: Review of the Performance Assessments and Composite Analyses*, DOE/NV/25946—6893. National Security Technologies, LLC, North Las Vegas, Nevada, March 2011.

Petkus 2002, *System Description, Primary Process System 63I*, WVNS-SD-63I. Petkus. L., West Valley Nuclear Services Company, West Valley, New York, August 22, 2002.

Petkus, et al. 2003, "A Complete History of the High-Level Waste Plant at the West Valley Demonstration Project." Petkus, L.L, J. Paul, P. Valenti, H. Houston, and J. May, Waste Management 03 Conference, Tucson, Arizona, February 23-27, 2003.

Perdue 2004, *Economic Practicality Analysis for the West Valley Demonstration Project's Vitrification Facility Melter*. Perdue, R.K., Westinghouse Electric Company Science and Technology Department, Pittsburgh, Pennsylvania, November 8, 2004.

Random House 1997, *Random House Unabridged Dictionary,* Random House, Inc. (http://www.infoplease .com/dictionary.html, Web page visited March 15, 2005).

RHI 2011, RHI Monofrax website, www.rhi-ag.com/internet/localchannels/monofrax/en/default.html.

Rykken 1986, *High-Level Waste Characterization at West Valley*, DOE/NE/44139-14. Rykken, L.E., West Valley Nuclear Services Co., West Valley, New York, June 2, 1986.

TCEQ 2011, *Radioactive Material License R04100 Issued to Waste Control Specials, LLC*, Amendment 10. Texas Commission on Environmental Quality, Austin, Texas, December 8, 2011.

Vance, et al. 1997, *Design of Equipment Used for High-Level Waste Vitrification at the West Valley Demonstration Project*, Topical Report DOE/NE/44139-73. Vance, R.F, B.A. Brill, D.E. Carl, H.S. Dhingra, R.A. Meigs, and M.G. Studd, West Valley Nuclear Services Company, West Valley, New York, June 1997.

VCT 2001, *Vitrification Completion Team Results and Commitments*. Memorandum from E.A Lowes (WVDP Deputy Director) to A.C. Williams (OH/WVDP Director), DW:2001:0313. U.S. Department of Energy, West Valley, New York, April 25, 2001.

WCS 2007, *Application for License to Authorize Near-Surface Land Disposal for Low-Level Radioactive Waste, Section 8: Performance assessment*, Revision 12a. Waste Control Specialists LLC, Rosebud, Texas, March 16, 2007. (http://www.urs-slc.com/wcs/)

WCS 2008, *Waste Acceptance Criteria*. Waste Control Specialists LLC, Rosebud, Texas, June 18, 2008.

WCS 2009, *Waste Acceptance Plan for Low-Level Radioactive Waste Disposal Facilities, Waste Control Specialists, LLC, Andrews, Texas Facility*, Revision 1. Waste Control Specialists LLC, Rosebud, Texas, September 3, 2009.

WCS 2011, *Updated Performance Assessment for the Low-Level Waste Facility, Radioactive Material License No. R04100, CN600616890/RN101702439*. Waste Control Specialists LLC, Rosebud, Texas, October 17, 2011.

Webster's 1984, Webster's II New Riverside University Dictionary. Riverside Publishing Co., 1984.

Wilhite 2001, *Comparison of LLW Disposal Performance Objectives, 10 CFR 61 and DOE 435.1*, WSRC-RP-2001-00341. Wilhite, E.L., Westinghouse Savannah River Company, Aiken, South Carolina, 2001.

WMG 2004a, *West Valley Nuclear Services Company (WVNSCO) Melter Characterization Results*, Report 4005-RE-024, Revision 3. WMG, Inc. Peekskill, New York, May 2004.

WMG 2004b, *WVDP Package Description Report*, Report 4005-RE-030, Revision 2. WMG, Inc. Peekskill, New York, May 2004.

WSMS 2008, *Vitrification Melter Waste Classification*, WSMS-WV-08-CAL002. Washington Safety Management Solutions, West Valley New York, March 2008.

WVES 2010a, *WVDP Radiological Controls Manual*, WVDP-010, Revision 30. Mortensen, K.P., West Valley Environmental Services LLC, West Valley, New York, April 27, 2010.

WVES 2010b, *Waste Profile Sheet for WVDP Vitrification Melter*, WVDP-000000021, Revision 1 (and associated documentation). West Valley Environmental Services LLC, West Valley, New York, May 24, 2010.

WVES 2011, *Data package with Microsoft Excel spreadsheet of HLW canister dose rates and analytical data from samples collected from the Concentrator Feed Makeup Tank for Batches 10 – 77.* West Valley Environmental Services, West Valley, New York, April 26, 2011.

WVNSCO 2001, *Waste Incidental to Reprocessing (WIR) Evaluation for Vitrification Facility Expended Materials*, Revision 1. West Valley Nuclear Services Company, West Valley, New York, October 25, 2001.

WVNSCO 2002a, *HLW Processing Systems Flushing Operations Run Plan*, WVNS-IRP-005, Revision 4. West Valley Nuclear Services Company, West Valley, New York, March 29, 2002.

WVNSCO 2002b, *Report on Deployment of Miscellaneous Tanks and Piping Cleaning Equipment and Methodology*, Tank Focus Area Milestone B.1.1 for Technical Task Plan (TTP) Oh0-0-WT-22. West Valley Nuclear Services Company, West Valley, New York, August 30, 2002.

WVNSCO 2002c, *Letter from J. Paul of WVNSCO to A. Williams of DOE-West Valley on completion of Fiscal Year 2002 contract milestone on Melter Feed Hold Tank Flushing*, WV 48, WD:2002:0400. West Valley Nuclear Services Company, West Valley, New York, July 30, 2002.

WVNSCO 2002d, *Letter from J. Paul of WVNSCO to A. Williams of DOE-West Valley on completion of Fiscal Year 2002 contract milestone on completion of HLW vitrification operations*s WV 48, WD:2002:0484. West Valley Nuclear Services Company, West Valley, New York, September 13, 2002.

WVNSCO 2004, *Radiation and Contamination Survey Report 127046*. West Valley Nuclear Services Company, West Valley, New York, November 15, 2004.

APPENDIX A
Drawings for the Vitrification Melter and Its Shipping Container

Appendix Purpose

The purpose of this appendix is to provide copies of the drawings of the Vitrification Melter and its shipping container so this information will be readily available.

Appendix Content

Five of these drawings are included.

Key Points

The drawings show the configuration and key dimensions of the Vitrification Melter and its waste container.

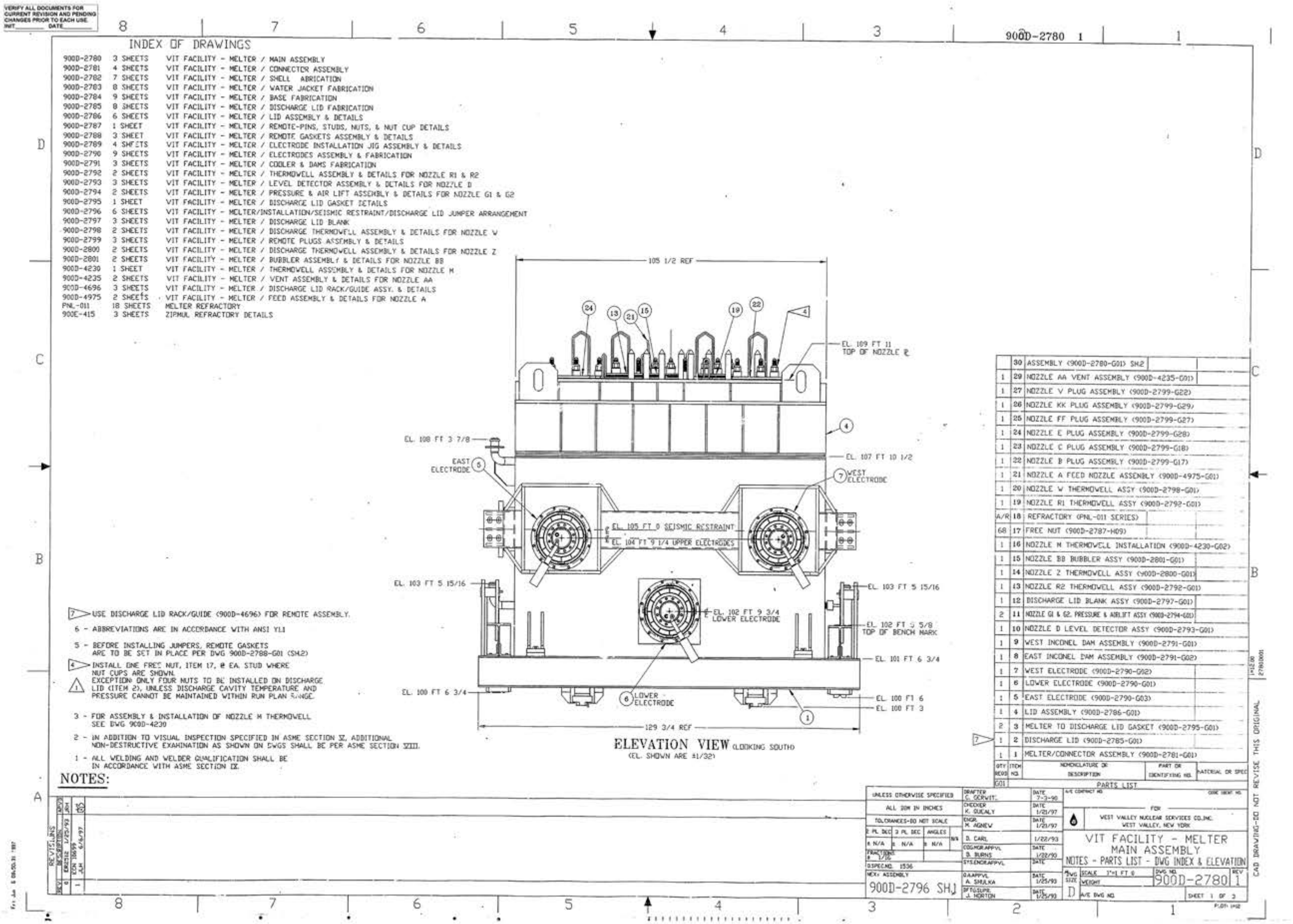


Figure A-1. Vitrification Melter Main Assembly Drawing

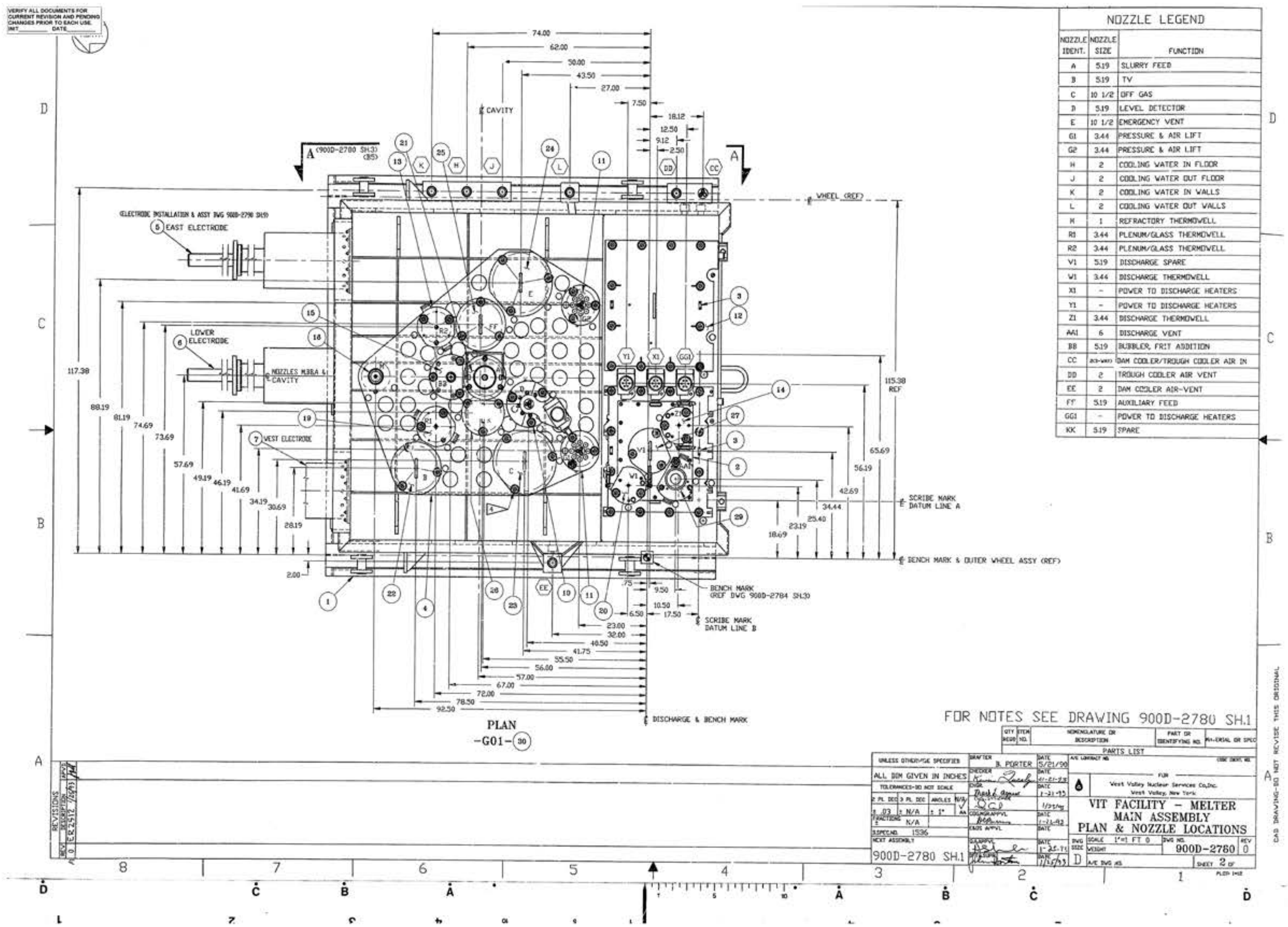


Figure A-2. Vitrification Melter Plan and Nozzle Location Drawing

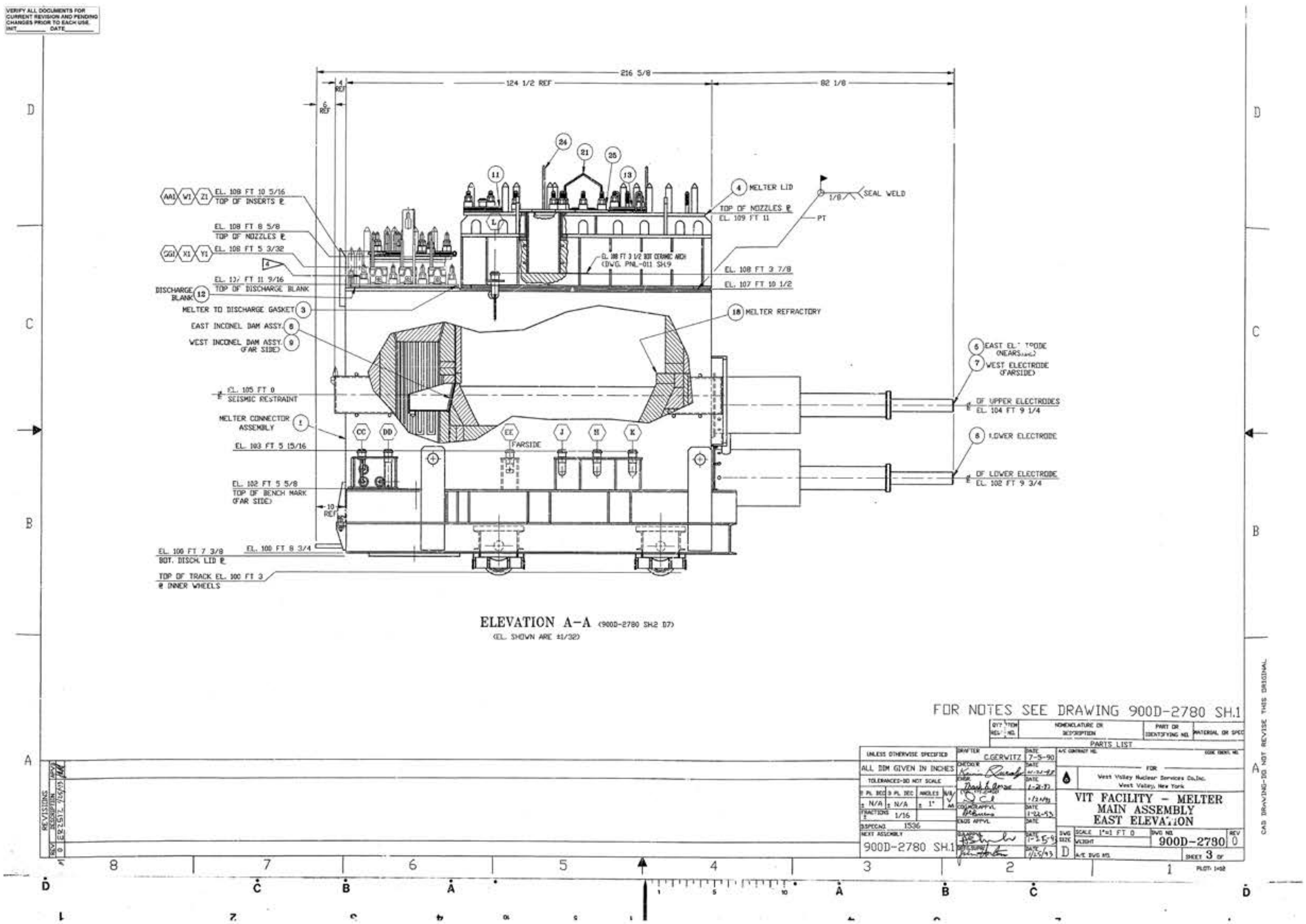


Figure A-3. Vitrification Melter Elevation Drawing

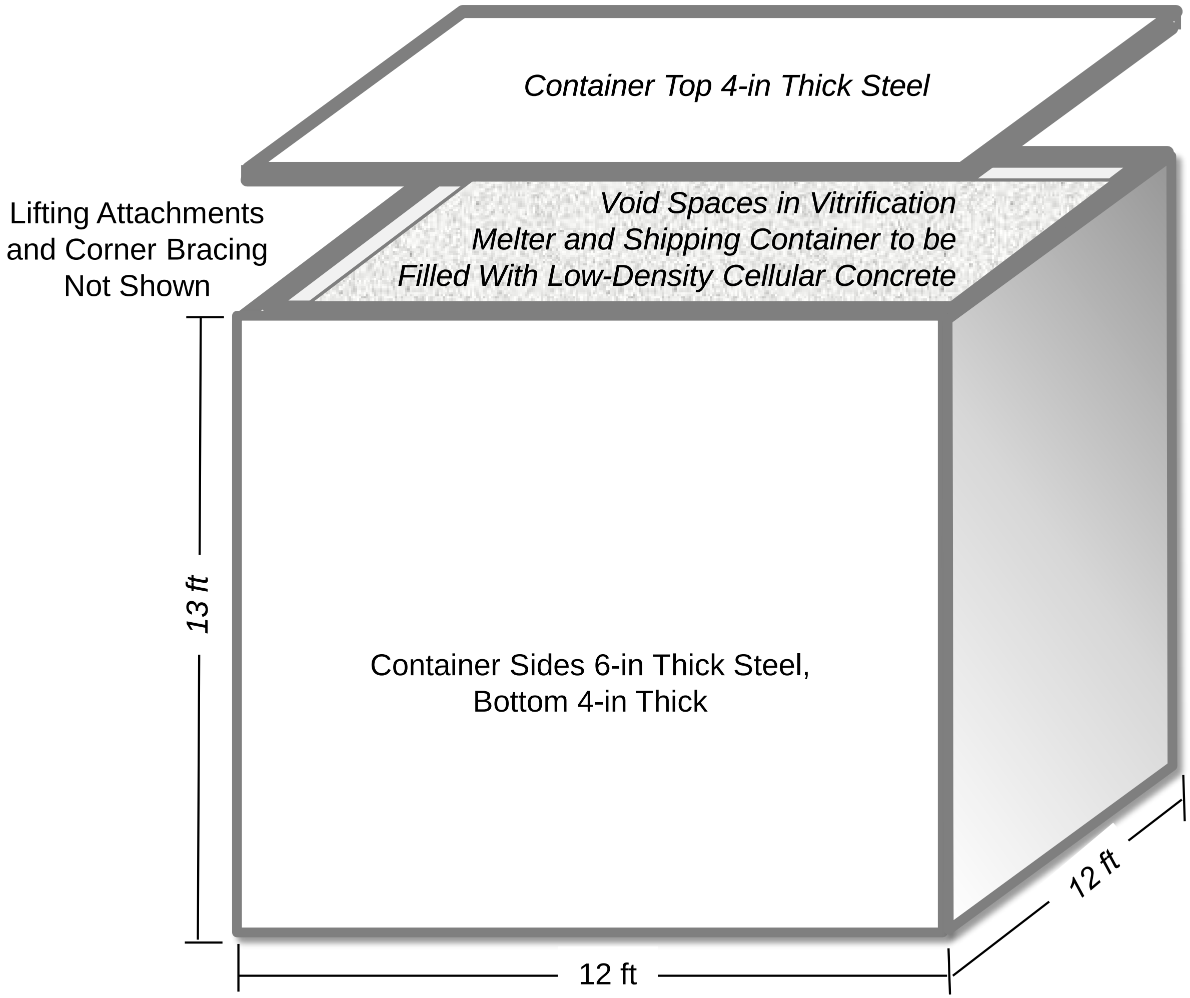


Figure A-4. Vitrification Melter Shipping Container

APPENDIX B
Management Controls to Assure Quality in this Evaluation

Appendix Purpose

The purpose of this appendix is to describe management control systems used to ensure quality in the process of evaluating whether the Vitrification Melter is incidental to reprocessing so it can be managed as low-level waste.

Appendix Content

This appendix identifies applicable Department of Energy and contractor requirements, describes the contractor quality assurance program, explains how quality was assured in data evaluation and engineering calculations, describes qualifications of the waste evaluation team, and outlines how documents and records related to the evaluation are controlled.

Key Points

- The process used was consistent with Department of Energy quality standards in 10 CFR 830, Subpart A, *Quality Assurance Requirements*, and DOE Order 414.1C (now DOE Order 414.1D), *Quality Assurance*.
- The process used was also consistent with contractor quality standards included in the Washington Safety Management Solutions Quality Assurance Plan and related implementing procedures.
- Engineering calculations prepared in support of this evaluation were independently reviewed and formally approved.
- This evaluation was prepared by a contractor team made up of waste management professionals experienced in waste-incidental-to-reprocessing determinations, under the direction and oversight of the Department's West Valley Project Office.
- This evaluation received several reviews.

1.0 Introduction

The purpose of this appendix is to describe management controls used to ensure that this evaluation was prepared with quality commensurate with its importance in relation to safe management of the Vitrification Melter to protect worker and public health and safety and the environment.

The management control systems used in this evaluation are formalized in a series of related standards and written procedures. They are designed to ensure that primary project objectives are met and that an optimum margin of safety for protection of personnel, the public, and the environment is achieved. Implementation of these systems by DOE and contractor managers and engineers provides assurance that information and engineering analyses which form the basis for this evaluation are accurate and supportable, and that the proper conclusions were reached.

This appendix addresses the following matters: (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.

2.0 Requirements and Procedures

The process followed in preparation of this evaluation was consistent with applicable requirements and guidance promulgated by DOE and the two contractors involved – West Valley Environmental Services (WVES) and Washington Safety Management Solutions (WSMS)[60] – including the following:

DOE

- 10 CFR 830, Subpart A, *Quality Assurance Requirements*
- DOE Order 414.1C (now DOE Order 414.1D), *Quality Assurance*
- DOE Order 435.1, *Radioactive Waste Management*
- DOE Manual 435.1-1, *Radioactive Waste Management Manual*
- DOE Guide 435.1-1, *Implementation Guide*
- DOE Order 5400.5, *Radiation Protection of the Public and the Environment* [now DOE Order 458.1]

Contractor Manual

- WSMS QA 100, *Washington Safety Management Solutions Quality Assurance Plan* (WSMS 2008)

3.0 Quality Assurance

The quality assurance process followed in this evaluation was based on DOE requirements contained in (1) 10 CFR Part 830, Subpart A which establishes quality requirements for DOE contractors conducting activities including providing items and services that affect, or may affect, nuclear safety of DOE facilities and (2) DOE Order 414.1C (now DOE Order 414.1D), *Quality Assurance* and associated guidance.

The WSMS Quality Assurance Plan (WSMS 2009) establishes WSMS policy related to quality and provides procedures to implement this policy, consistent with applicable DOE requirements in 10 CFR 830 and DOE Order 414.1C (now DOE Order 414.1D). It also establishes responsibilities for achieving and verifying quality in company activities, including engineering evaluations such as the one reflected in this document. American Society of Mechanical Engineers NQA-1, *Quality Assurance Requirements for Nuclear Facilities* (ASME 2000) and other consensus quality standards were used in the development of the WSMS quality assurance program.

4.0 Data Evaluation and Engineering Calculations

This evaluation did not entail collecting data by field measurements or by obtaining samples and analyzing them in a laboratory. Rather, it utilized existing data from prior field measurements

[60] This company is now known as URS Safety Management Solutions.

and sample analyses. All calculations were performed by other contractor staff with two exceptions: the sum-of-fraction calculation (Table 6-1) and the PE-g estimate.

To assure quality, these engineering calculations were prepared, reviewed, and approved in accordance with the following process:

- Calculations were formally documented by compiling all relevant information into a calculation package, which included compilations of the calculations and all related information on which they were based. This calculation package describes input data and their sources, software used, assumptions made, and calculation results. Sufficient detail is included to allow a reviewer to independently duplicate the results.
- This calculation package was reviewed by a technically competent person who had not been involved in its development to independently check the calculations and their accuracy. This check included verifying matters such as (1) use of appropriate data sources, (2) correct utilization of input data, (3) use of appropriate methods, (4) use of valid assumptions, (5) accuracy of results by representative independent calculations, and (6) reasonableness of the results in considering the range of expectations.
- This calculation package was also reviewed and approved by the WSMS Project Manager, who verified that it had received a thorough technical review and that review comments were appropriately resolved.
- This calculation package includes a coversheet signed by the preparer, the reviewer, and the WSMS Project Manager signifying that it was prepared, reviewed, and approved using this basic process.

5.0 Evaluation Preparation and Review

This evaluation was prepared by the WSMS team described below in Section 6.

Evaluation Preparation

The WSMS Project Manager assigned particular team members to assume the lead on individual parts of the evaluation. These parts were prepared in accordance with requirements and guidance described above, factoring in the results of supporting engineering calculations and results, as applicable.

Appropriate precedents were considered. These included waste determinations prepared by the DOE pursuant to the Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, even though that act does not apply to the WVDP (DOE 2006a and DOE 2006b).

As individual parts of the document were completed, they were integrated into a single document to form the complete evaluation, and the evaluation was then edited for consistency.

Evaluation Review

As individual parts of the evaluation were completed, they received one or more peer reviews by members of the WSMS team. Review comments were incorporated or otherwise resolved.

The complete evaluation received additional review by the WSMS team. This detailed review addressed adequacy, completeness, correctness, and compliance with applicable requirements. Review comments were incorporated or resolved.

The completed evaluation was reviewed by the WVES Project Manager and WVES technical specialists, as assigned, to ensure that it met all DOE requirements and was technically accurate and complete. Comments from this review were incorporated before the evaluation was submitted to DOE.

Evaluation Approval

DOE reviewed the evaluation and approved it after written resolution of the DOE comments and incorporation of those comments.

6.0 Personnel Qualification

Members of the WSMS team dedicated to the evaluation and preparation of this evaluation report are experienced regulatory analysts. They brought to this project extensive knowledge and experience in radioactive waste management and related technical standards such as:

- DOE Order 435.1, *Radioactive Waste Management*;
- DOE Manual 435.1-1, *Radioactive Waste Management Manual*;
- DOE Guide 435.1-1, *Implementation Guide for Use With DOE M 435.1-1*;
- 10 CFR Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste*; and
- NUREG-0782, *Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste* (NRC 1981).

Team members included engineers and scientists who are WSMS employees, along with an independent consultant. All are knowledgeable of the Vitrification Melter and have previous experience with waste-incidental-to-reprocessing determinations.

7.0 Oversight and Independent Review

A key element in implementation of the quality standards and procedures outlined above is that individuals are responsible for their own work. WSMS management also provided direction and oversight to ensure that these standards and procedures were closely followed, and regularly assessed compliance with quality requirements. In addition, WVES management and DOE personnel in the WVDP Field Office provided oversight and review to this end.

8.0 Document and Record Control

Documents and records associated with this evaluation are controlled in accordance with the WVDP Records Management System (WVES 2008). This system is used to ensure that records important to safety and quality are generated, reviewed, approved, collected, and maintained so they accurately reflect completed work and facility conditions, and comply with applicable statutory or contractual requirements.

The WVDP Records Management System incorporates the requirements of DOE Order 200.1, *Information Management Program* (now DOE Order 200.1A, *Information Technology Management*), and DOE Order 414.1C (now DOE Order 414.1D), *Quality Assurance*. Schedules for records retention and disposition comply with the General Records Schedule of the National Archives and Records Administration and other approved records schedules.

The Records Management procedures include instructions for retention, protection, preservation, changes, traceability, accountability, and retrievability of records. They also provide controls to ensure records are legible, accurate, complete, retrievable, and validated by authorized personnel. Records are stored and maintained to minimize the risk of damage, larceny, vandalism, or deterioration. Active records are not sent to records holding facilities, but are stored in a facility where the records may be readily accessed.

9.0 References

Federal Statutes

Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, Public Law 108-375, 118 Stat. 1811. 108th United States Congress, October 28, 2004.

Code of Federal Regulations and Federal Register Notices

10 CFR 830, Subpart A, *Quality Assurance Requirements.*

DOE Orders, Policies, Manuals, and Standards

DOE Order 200.1, *Information Management Program*. U. S. Department of Energy, Washington, D.C., September 30, 1996.

DOE Order 200.1A, *Information Technology Management*. U. S. Department of Energy, Washington, D.C., December 23, 2008.

DOE Order 414.1C, *Quality Assurance*. U. S. Department of Energy, Washington, D.C., June 17, 2005.

DOE Order 414.1D, *Quality Assurance*. U. S. Department of Energy, Washington, D.C., April 25, 2011.

DOE Order 435.1, *Radioactive Waste Management*, Change 1. U. S. Department of Energy, Washington, D.C., August 28, 2001.

DOE Order 458.1, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., June 6, 2011.

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., January 7, 1993.

DOE Manual 435.1-1, *Radioactive Waste Management Manual*, Change 1. U. S. Department of Energy, Washington, D.C., June 19, 2001.

DOE Guide 435.1-1, *Implementation Guide for Use with DOE M-435.1-1*. U.S. Department of Energy, Washington, D.C., July 1999.

Other References

ASME 2000, *Quality Assurance Requirements For Nuclear Facilities*, NQA-1. American Society of Mechanical Engineers, New York, 2000.

DOE 2006a, *Basis for Section 3116 Determination, Salt Waste Disposal, Savannah River Site*, DOE-WD-2005-001. U.S. Department of Energy – Savannah River, Aiken, South Carolina, January 2006.

DOE 2006b, *Basis for Section 3116 Determination for the Idaho Technology and Engineering Center Tank Farm Facility*, DOE/NE-ID-11226, Revision 0. U.S. Department of Energy – Idaho National Laboratory, Idaho Falls, Idaho, November 2006.

NRC 1981, *Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste*, NUREG-0782. U.S. Nuclear Regulatory Commission, Washington, D.C., 1981.

WSMS 2009, *Washington Safety Management Solutions LLC Quality Assurance Plan*, WSMS QA 100, Revision 1. Washington Safety Management Solutions, Aiken, South Carolina, August 10, 2009.

WVES 2010, *WVNSCO Manual for Records Management and Storage*. WVDP-262, Revision 14. West Valley Environmental Services LLC, West Valley, New York, June 15, 2010.

APPENDIX C
Comparability of DOE, NRC and Texas Requirements for LLW Disposal

Appendix Purpose

The purpose of this appendix is to show that Department of Energy, Nuclear Regulatory Commission, and State of Texas requirements for disposal of low-level waste are comparable.

Appendix Content

This appendix identifies applicable Department of Energy performance objectives and the similar Nuclear Regulatory Commission and State of Texas performance objectives and discusses their comparability.

Key Points

- Requirements for low-level waste disposal are embodied in sets of performance objectives for the waste disposal facility.
- The Department of Energy performance objectives are described in DOE Manual 435.1-1, *Radioactive Waste Management Manual*.
- The Nuclear Regulatory Commission performance objectives are described in Subpart C, *Performance Objectives*, of 10 CFR Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste*.
- The performance objectives in the Texas Administrative Code that apply to the WCS low-level waste disposal facility – which are included in the Title 30, Part 1, Chapter 336, Subchapter H, Rule §336.723-727 – mirror the Nuclear Regulatory Commission performance objectives.
- Department of Energy, Nuclear Regulatory Commission, and State of Texas performance objectives for low-level waste disposal are comparable.
- The Department of Energy, the Nuclear Regulatory Commission, and the State of Texas all have provisions for imposing additional requirements for low-level waste disposal and the State of Texas has imposed additional requirements for the WCS low-level waste disposal facility.

1.0 Introduction

This appendix identifies performance objectives for disposal of LLW by the DOE, the NRC, and the State of Texas. It then compares these performance objectives. As noted previously, the performance objectives in the State of Texas regulations mirror the NRC performance objectives at 10 CFR 61, Part C, i.e., they are essentially identical except for the use of difference section numbers.

Information in this appendix is based in part on previous detailed comparison studies of DOE and NRC performance objectives for LLW disposal (Cole, et al. 1995 and Wilhite 2001).

2.0 Applicable Performance Objectives

DOE Manual 435.1-1, *Radioactive Waste Management Manual*, describes DOE requirements for disposal of LLW. The comparable NRC requirements appear in Subpart C of 10 CFR Part 61, which lists one general requirement and four performance objectives, which are reproduced below.

Section 61.40, General Requirement

"Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44."

Section 61.41, Protection of the General Population from Releases of Radioactivity

"Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirem to the whole body, 75 millirem to the thyroid, and 25 millirem to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable."

Section 61.42, Protection of Individuals from Inadvertent Intrusion

"Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

Section 61.43, Protection of Individuals During Operations

"Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in 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."

Section 61.44, Stability of the Disposal Site After Closure

"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."

The State of Texas requirements for LLW disposal at Texas Administrative Code, Title 30, Part 1, Chapter 336, Subchapter H, Rule §336.723-777 are based on the NRC requirements at Subpart C of 10 CFR Part 61 and are exactly the same except for minor wording differences identified below.

3.0 Comparability of the General Requirements

3.1 DOE

The general requirement in DOE Manual 435.1-1, Section IV.P(1), is expressed as follows:

"Low-level waste disposal facilities shall be sited, designed, operated, maintained, and closed so that a reasonable expectation exists that the following performance objectives will be met for waste disposed of after September 26, 1988."

3.2 NRC

The NRC regulations in 10 CFR 61.40 provide in relevant part:

"Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44."

3.3 State of Texas

The State of Texas regulations (Rule §336.723) mirror the NRC regulations in 10 CFR 61.40.

3.4 Discussion

The statement of NRC requirements in 10 CFR 61.40 is nearly identical to that of the DOE general requirement. The DOE requirement adds the concept of maintenance, which is implicit in the NRC requirement. The DOE requirement does not mention control after closure, but this concept is embodied in the DOE requirements for closure, specifically DOE Manual 435.1, Section IV.Q (2)(c), which requires DOE control until it can be shown that release of the disposal site for unrestricted use will not compromise DOE requirements for radiological protection of the public.

The DOE general requirement for LLW disposal, the NRC general requirement of 10 CFR 61.40, and the State of Texas general requirement are therefore comparable.

4.0 Comparability Regarding Protection of the General Population from Releases of Radioactivity

4.1 DOE

DOE requirements of DOE Manual 435.1-1, Section IV.P(1), read as follows:

"(a) Dose to representative members of the public shall not exceed 25 millirem in a year total effective dose equivalent from all exposure pathways, excluding the dose from radon and its progeny in air.

(b) Dose to representative members of the public via the air pathway shall not exceed 10 millirem in a year total effective dose equivalent, excluding the dose from radon and its progeny.

(c) Release of radon shall be less than an average flux of 20 pCi/m^2/s at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/L of air may be applied at the boundary of the facility."

4.2 NRC

NRC regulations in 10 CFR 61.41 are expressed as follows:

"Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirem to the whole body, 75 millirem to the thyroid, and 25 millirem to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable."

4.3 State of Texas

The State of Texas regulations (Rule §336.724) mirror the NRC regulations in 10 CFR 61.41 with two minor wording differences. The Texas rule uses the phrase "annual dose above background" instead of "annual dose." In the second sentence, the Texas rule uses the phrase "Effort shall be made" instead of "Reasonable effort should be made."[61]

4.4 Discussion

DOE uses more current radiation protection methodology, consistent with that used in NRC's radiation protection standards in NRC's 10 CFR 20, *Standards for Protection Against Radiation*. Because NRC has not revised 10 CFR 61.41 to reflect the more current methodology in 10 CFR 20, DOE's requirements and those in 10 CFR 20 differ slightly from those in 10 CFR 61.41. However, the resulting allowable doses are comparable, as NRC has acknowledged (NRC 2005). NRC has indicated that it expects DOE to use the newer methodology in 10 CFR 20 and DOE Manual 435.1-1 for the WVDP decommissioning (NRC 2002). Both NRC and DOE use a performance assessment to assess whether the dose limit will be met.

The DOE requirements go beyond this NRC performance objective by specifying an assessment of the impacts of LLW disposal on water resources (i.e., DOE Manual 435.1, Section IV.P (2)(g)). The NRC requirement includes maintaining releases to the environment ALARA. Although this requirement is not included in the DOE performance objective, it is included in the performance assessment requirements (i.e., DOE Manual 435.1-1, Section IV.P (2)(f)).

Because the State of Texas regulations are essentially the same as the NRC regulations, the conclusions about the comparability of the DOE and NRC requirements also apply to the comparability of the State of Texas requirements.

5.0 Comparability Regarding Protection of Individuals from Inadvertent Intrusion

5.1 DOE

DOE requirements of DOE Manual 435.1-1, Section IV.P(2)(h), for protection of individuals from inadvertent intrusion read as follows:

"For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 millirem (1 mSv) in a year and 500 millirem (5 mSv) total effective dose equivalent excluding radon in air."

[61] WCS also uses a performance criterion for radon gas flux emanating from the disposal facility cover of 20 pCi/m2/s based on a provision of 40 CFR Part 192 of 20 pCi/m2/s, although 40 CFR Part 192 does not apply to LLW disposal facilities. This is not a State of Texas requirement, but it is the same as DOE's criterion in DOE Manual 435.1-1, Section IV.P(1) except for DOE's separate limit at the facility boundary.

5.2 NRC

NRC requirements of 10 CFR 61.42 are expressed as follows:

"Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

5.3 State of Texas

The State of Texas regulations (Rule §336.725) mirror the NRC regulations in 10 CFR 61.42. However, Texas has imposed on WCS a limit of 25 mrem per year (0.25 mSv) for inadvertent intruders (WCS 2011).

5.4 Discussion

The DOE LLW disposal requirement that the performance assessment include an assessment of the impacts on a person inadvertently intruding into the disposal facility is more stringent than the NRC requirement. The NRC waste classification system is based on intruder calculations using a 500 millirem per year dose limit (NRC 1982). The DOE requirement uses a 100 millirem per year limit for chronic exposures and a 500 millirem limit for acute exposures.

The State of Texas regulations mirror the NRC regulations. However, as noted above, Texas has imposed an additional requirement for a lower limit of 25 mrem per year dose limit for inadvertent intruders. Therefore the State of Texas requirement is more limiting than the DOE and NRC requirements.[62] The comparability of DOE, NRC, and State of Texas provisions for imposing additional requirements is discussed in Section 8 below.

6.0 Comparability Regarding Protection of Individuals During Operations

6.1 DOE

The DOE requirements in DOE Manual 435.1-1, Section I.E(13), for protection of individual during operations read 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* [now DOE Order 458.1]."

6.2 NRC

The NRC requirements of 10 CFR 61.43 are expressed as follows:

"Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in 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."

[62] Note that Paragraph 4.b(4) of DOE Order 435.1, *Radioactive Waste Management*, requires DOE to comply with applicable Federal, State, and local laws and regulations. Therefore, if the melter waste package were to be transported to the WCS LLW facility for disposal, the facility would have to meet the 25 mrem per year dose limit for inadvertent intruders and the waste package would have to meet associated waste acceptance criteria.

6.3 State of Texas

The State of Texas regulations (Rule §336.726) mirror the NRC regulations in 10 CFR 61.43.

6.4 Discussion

The ALARA concept is an integral part of DOE radiation and environmental protection programs. DOE requirements for occupational radiological protection are addressed in 10 CFR 835, and similar requirements for radiological protection of the public and the environment are addressed in DOE Order 458.1. The NRC 10 CFR 61.43 requirement references 10 CFR 20, *Standards for Protection Against Radiation*, which contains similar radiological protection standards for workers and the public.

Appendix D provides additional information on the comparability of DOE and NRC radiation dose standards that apply to protection of individuals during operations. The State of Texas radiation dose standards mirror the NRC dose standards as explained in Appendix D.

7.0 Comparability Regarding Stability of the Disposal Site After Closure

7.1 DOE

The DOE requirements of DOE Manual 435.1-1, Sections IV.Q(1)(a) and (b) and IV.Q(2)(c), for stability of the disposal site after closure are expressed as follows:

> <u>“Disposal Site Stability (DOE Manual 435.1, Section IV.Q(1)(a) and (b))</u>. A preliminary closure plan shall be developed and submitted to Headquarters for review with the performance assessment and composite analysis. The closure plan shall be updated following issuance of the disposal authorization statement to incorporate conditions specified in the disposal authorization statement. Closure plans shall:
>
> (a) Be updated as required during the operational life of the facility.
>
> (b) Include a description of how the disposal facility will be closed to achieve long-term stability and minimize the need for active maintenance following closure and to ensure compliance with the requirements of DOE 5400.5, *Radiation Protection of the Public and the Environment* [now DOE Order 458.1].”
>
> <u>“Disposal Facility Closure (DOE Manual 435.1, Section IV.Q(2)(c)</u>). Institutional control measures shall be integrated into land use and stewardship plans and programs, and shall continue until the facility can be released pursuant to DOE Order 5400.5, *Radiation Protection of the Public and the Environment* [now DOE Order 458.1].”

7.2 NRC

The NRC requirements of 10 CFR 61.44 state that:

> “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.”

7.3 State of Texas

The State of Texas regulations (Rule §336.727) mirror as the NRC regulations in 10 CFR 61.43.

7.4 Discussion

The DOE LLW disposal requirements address long-term stability of the site by requiring a description of how closure will achieve stability in the closure plan, and by a description of how closure will minimize the need for active maintenance following closure (DOE Manual 435.1, Section IV.Q (1)(b)). Additionally, one of the performance assessment requirements (DOE Manual 435.1, Section IV.P (2)(c)) states: "Performance assessments shall address reasonably foreseeable natural processes that might disrupt barriers against release and transport of radioactive materials." Thus, the performance assessment will include a projection of the long-term stability of the site, considering reasonably foreseeable natural processes such as erosion, degradation of waste packages, etc.

8.0 Comparability Regarding Provisions for Imposing Additional Requirements

8.1 DOE

Section 4.d of DOE Order 435.1, *Radioactive Waste Management*, states that:

"DOE, within its authority, may impose such requirements, in addition to those established in this Order, as it deems appropriate and necessary to protect the public, workers, and the environment, or to minimize threats to property."

8.2 NRC

NRC provisions for imposing additional requirements on the license for a LLW disposal facility are contained in 10 CFR 61.24(h), which states:

"(h) The Commission may incorporate in any license at the time of issuance, or thereafter, by appropriate rule, regulation or order, additional requirements and conditions with respect to the licensee's receipt, possession, and disposal of source, special nuclear or byproduct material as it deems appropriate or necessary in order to:

(1) Promote the common defense and security;

(2) Protect health or to minimize danger to life or property;

(3) Require reports and the keeping of records, and to provide for inspections of activities under the license that may be necessary or appropriate to effectuate the purposes of the Act and regulations thereunder."

8.3 State of Texas

The Texas provisions for imposing additional requirements on the license for a low-level waste disposal facility are contained in Rule §336.716(g), which states:

"(g) The commission may incorporate in any license at the time of issuance, or thereafter, by appropriate rule or order, additional requirements and conditions with respect to the licensee's receipt, possession, and disposal of waste as it deems appropriate or necessary in order to:

(1) protect the health and safety of the public and the environment; and

(2) require reports and recordkeeping and to provide for inspections of activities under the license that may be necessary or appropriate to effectuate the purposes of the TRCA [Texas Radiation Control Act] and rules thereunder."

8.4 Discussion

The DOE requirement is broader in scope than the NRC and State of Texas requirements because the DOE requirement applies to all aspects of radioactive waste management while the NRC and State of Texas requirements apply to licenses for LLW disposal facilities. Otherwise, the requirements are comparable.

9.0 References

Code of Federal Regulations

10 CFR Part 20, *Standards for Protection Against Radiation*.

10 CFR 61, Subpart C, *Licensing Requirements for Land Disposal of Radioactive Waste, Performance Objectives*.

10 CFR Part 835*, Occupational Radiation Protection*.

40 CFR Part 192, *Health and Environmental Standards for Uranium and Thorium Mill Tailings.*

DOE Orders, Policies, and Manuals

DOE Order 435.1, *Radioactive Waste Management*

DOE Order 458.1, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., June 6, 2011.

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., January 7, 1993.

DOE Manual 435.1-1, *Radioactive Waste Management Manual*, Change 1. U. S. Department of Energy, Washington, D.C., June 19, 2001.

State Regulations

Texas Administrative Code, Title 30, Part 1, Chapter 336, *Radioactive Substance Rules* (http://info.sos.state.tx.us/pls/pub/readtac$ext.ViewTAC?tac_view=4&ti=30&pt=1&ch=336)

Other References

Cole, et al. 1995, *Comparison of Selected DOE and Non-DOE Requirements, Standards, and Practices for Low-Level Radioactive Waste Disposal*, DOE/LLW-225, Revision 0. Cole, L., D. Kudera, and W. Newberry, Idaho National Engineering Laboratory, Lockheed Idaho Technologies Company, Idaho Falls, Idaho, December 1995.

NRC 1982, *Final Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste," Summary and Main Report*, NUREG-0945, Volume 1. U.S. Nuclear Regulatory Commission, Washington, D.C., November 1982.

NRC 2002, *Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement*. U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, 67 FR 5003, February 1, 2002.

NRC 2005, *U.S. Nuclear Regulatory Commission Technical Evaluation Report for the U.S. Department of Energy Savannah River Site Draft Section 3116 Waste Determination for Salt Waste Disposal*. U.S. Nuclear Regulatory Commission, Washington, D.C., December 2005.

WCS 2011, *Updated Performance Assessment for the Low-Level Waste Facility, Radioactive Material License No. R04100, CN600616890/RN101702439*. Waste Control Specialists LLC, Rosebud, Texas, October 17, 2011.

Wilhite 2001, *Comparison of LLW Disposal Performance Objectives, 10 CFR 61 and DOE 435.1*, WSRC-RP-2001-00341. Wilhite, E.L., Westinghouse Savannah River Company, Aiken, South Carolina, 2001.

APPENDIX D
Comparability of DOE, NRC, and Texas Dose Standards

Appendix Purpose

The purpose of this appendix is to compare Department of Energy, Nuclear Regulatory Commission, and State of Texas radiation dose standards that apply to individual workers and to members of the public.

Appendix Content

This appendix identifies applicable Department of Energy dose standards and the similar Nuclear Regulatory Commission and State of Texas dose standards and discusses their comparability.

Key Points

- The Department of Energy radiation dose standards appear in 10 CFR Part 835, *Occupational Radiation Protection*, and in DOE Orders.
- The Nuclear Regulatory Commission radiation dose standards appear in 10 CFR Part 20, *Standards for Protection Against Radiation.*
- Department of Energy and Nuclear Regulatory Commission radiation dose standards are comparable.
- The State of Texas dose standards that apply to the WCS low-level waste disposal facility – which are included in the Texas Administrative Code, Title 30, Part 1, Chapter 336, Subchapter D – mirror the Nuclear Regulatory Commission dose standards.

1.0 Introduction

The purpose of this appendix is to compare the DOE, NRC, and State of Texas dose standards that apply to protection of the public and the workers from radiation during operations associated with preparing the Vitrification Melter for shipment at the WVDP and handling of the Melter when it is received at either the Nevada National Security Site or the WCS LLW disposal facility in Texas for disposal, assuming that it will be sent to one of those facilities.

Section 5.2.4 of the body of this evaluation briefly addressed protection of individuals during these operations at the WVDP, the Nevada National Security Site, and the WCS LLW disposal facility. Appendix C also addressed this matter. This appendix provides a more detailed treatment of the dose standards used.

Requirements in NRC's regulations at 10 CFR 61.43 state:

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

This requirement references 10 CFR Part 20, *Standards for Protection Against Radiation*, which contains radiological protection standards for workers and the public. The DOE requirements for occupational radiological protection are provided in 10 CFR Part 835, *Occupational Radiation Protection*, and those for radiological protection of the public and the environment are provided in DOE Order 458.1, *Radiation Protection of the Public and the Environment*. The State of Texas radiation protection standards appear in the Texas Administrative Code, Title 30, Part 1, Chapter 336, Subchapter D.

The NRC standards for radiation protection in 10 CFR Part 20 that are considered in detail in this evaluation are the dose limits for the public and the workers during disposal operations set forth in 10 CFR 20.1101(d), 20.1201(a)(1)(i), 20.120 1(a)(1)(ii), 20.120 1(a)(2)(i), 20.120 1(a)(2)(ii), 20.1201(e), 20.1208(a), 20. 1301(a)(1), 20.1301(a)(2), and 20.1301(b).[63] These NRC dose limits correspond to the DOE dose limits in 10 CFR 835 and relevant DOE orders that establish DOE regulatory and contractual requirements for DOE facilities and activities. As discussed in Section 5.2.4 of this evaluation, operations related to disposal of the Vitrification Melter will meet these dose limits and doses will be maintained ALARA. As explained below, the State of Texas radiation protection standards mirror the NRC radiation protection standards.

2.0 Air Emissions Limit for Individual Member of the Public

2.1 DOE

DOE is subject to and complies with the U.S. Environmental Protection Agency requirement in 40 CFR 61.92.[64]

2.2 NRC

The NRC regulation in 10 CFR 20.1101(d) provides in relevant part:

> "[A] constraint on air emissions of radioactive material to the environment, excluding radon-222 and its daughters, shall be established … such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv)/y from these emissions."

[63] The "standards for radiation protection" in 10 CFR 20 (as cross-referenced in the performance objective in 10 CFR 61.43), which are relevant to this evaluation, are the dose limits for radiation protection of the public and the workers during disposal operations, and not those which address general licensing, administrative, programmatic, or enforcement matters administered by NRC for NRC licensees. Accordingly, this evaluation addresses in detail the radiation dose limits for the public and the workers during disposal operations that are contained in the provisions of 10 CFR 20 referenced above. Although 10 CFR 20.1206(e) contains limits for planned special exposures for adult workers, there will not be any such planned special exposures for work related to the Vitrification Melter. Therefore, this limit is not discussed further in this evaluation. Likewise, 10 CFR 20.1207 specifies occupational dose limits for minors. However, there will not be minors working at the WVDP or the Nevada National Security Site who would receive an occupational dose. Therefore, this limit is not discussed further in this evaluation.

[64] 40 CFR 61.92 provides as follows: "Emissions of radionuclides to the ambient air from DOE facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/y. It is assumed that the individual is an adult living at the site perimeter that is exposed to the maximum yearly radioactive atmospheric release and maximum radiation concentration in food for 365 days per year. For the airborne pathway, the dose is developed by the input of atmospheric release data, vegetation consumption data, milk consumption data, and beef consumption data."

2.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.304 mirrors the NRC regulation.

2.4 Discussion

The DOE, NRC, and State of Texas requirements are comparable.

3.0 Total Effective Dose Equivalent Limit for Adult Workers

3.1 DOE

DOE's regulation in 10 CFR 835.202(a)(1) requires that the occupational dose per year for general employees shall not exceed a total effective dose equivalent of 5 rem[65].

3.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> "(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures...to the following dose limits.
>
> (1) An annual limit, which is the more limiting of –
>
> (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv)."

3.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.305 is mirrors the NRC regulation.

3.4 Discussion

The DOE, NRC, and State of Texas requirements are comparable.

4.0 Any Individual Organ or Tissue Dose Limit for Adult Workers

4.1 DOE

The DOE regulation in 10 CFR 835.202(a)(2) provides in relevant part:

> ". . . the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:
>
> (2) The sum of the deep dose equivalent for external exposures and the committed dose equivalent to any organ or tissue other than the lens of the eye of 50 rems (0.5 sievert)"

[65] The DOE's regulations at 10 CFR 835.202(a)(1) and (a)(2) require that the occupational dose per year for general employees shall not exceed both a total effective dose equivalent of 5 rem and the sum of the deep-dose equivalent for external exposures and the committed dose equivalent to any other organ or tissue other than the lens of the eye of 50 rem. The NRC's regulation specifies that either of these two limits shall be met by NRC licensees, whichever is more limiting. Thus, DOE imposes stricter, separate requirements. The provisions of DOE's requirements at 10 CFR 835.202(a)(1) and (a)(2), which correlate to NRC requirements at 10 CFR 20.1201(a)(1) and (a)(2), are discussed in separate subsections in this evaluation.

4.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

"(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures...to the following dose limits.

(ii) An annual limit, which is the more limiting of – The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv)."

4.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.305 mirrors the NRC regulation.

4.4 Discussion

The DOE, NRC, and State of Texas requirements are comparable.

5.0 Annual Dose Limit to the Lens of the Eye for Adult Workers

5.1 DOE

The DOE regulation in 10 CFR 835.202(a)(3) provides in relevant part:

". . . the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:

(3) A lens of the eye dose equivalent of 15 rems (0.15 sievert)"

5.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

(a) [C]ontrol the occupational dose to individual adults, except for planned special exposure the following dose limits. ...

(2) The annual limits to the lens of the eye, to the skin of the whole body or to the skin of the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv).

5.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.305 mirrors the NRC regulation.

5.4 Discussion

The DOE, NRC and State of Texas requirements are comparable.

6.0 Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers

6.1 DOE

The DOE regulation in 10 CFR 835.202(a)(4) provides in relevant part:

“. . . the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:

(4) A shallow dose equivalent of 50 rems (0.5 sievert) to the skin or any extremity.”

6.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures...to the following dose limits. ...

(2) The annual limits to the lens of the eye, the skin of the whole body, or to the skin of the extremities, which are: ...

(ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

6.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.305 mirrors the NRC regulation.

6.4 Discussion

The DOE, NRC, and State of Texas requirements are comparable.

7.0 Limit on Soluble Uranium Intake

7.1 DOE

Requirements in DOE Order 440.1B for soluble uranium intake are the more restrictive of the concentrations in the American Conference of Governmental Industrial Hygienists threshold limit values (0.2 mg/m^3, which is the same as noted in 10 CFR 20, Appendix B) or the Occupational Safety and Health Administration permissible exposure limit (0.05 mg/m^3). The permissible exposure limit for soluble uranium, which equates to a soluble uranium intake of 2.4 mg/week, is the more restrictive of the two.

7.2 NRC

The NRC regulation in 10 CFR 20.1201(e), concerning occupational dose limits for adults, provides in relevant part: “in addition to the annual dose limits, ... limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity.”

7.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.305 mirrors the NRC regulation.

7.4 Discussion

The DOE requirements are more restrictive.

8.0 Dose Equivalent to an Embryo/Fetus

8.1 DOE

The DOE regulation in 10 CFR 835.206(a) provides in relevant part:

> "The dose equivalent limit for the embryo/fetus from the period of conception to birth, as a result of occupational exposure of a declared pregnant worker, is 0.5 rem (0.005 sievert)."

After declaration of pregnancy, DOE provides the option of a mutually agreeable assignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure during the remainder of the gestation period is unlikely. In addition, personnel dosimetry[66] is provided and used to track exposure carefully.

8.2 NRC

The NRC regulation in 10 CFR 20.1208(a), concerning the dose equivalent to an embryo/fetus, provides in relevant part:

> "ensure that the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv)."

8.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.312 mirrors the NRC regulation.

8.4 Discussion

The DOE, NRC, and State of Texas requirements are comparable.

9.0 Dose Limits for Individual Members of the Public (Total Annual Dose)

9.1 DOE

Provisions in DOE Order 458.1 limit public doses to 0.1 rem per year.

9.2 NRC

The NRC regulation in 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

> (a) "[C]onduct operations so that –
>
> > (1) The total effective dose equivalent to individual members of the public ...does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released..., from voluntary

[66] The term *dosimetry* or *personnel dosimetry* refers to a device carried or worn by an individual working near radiation for measuring the amount of radiation to which he or she is exposed.

participation in medical research programs, and from the …disposal of radioactive material into sanitary sewerage."

9.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.313 mirrors the NRC regulation.

9.4 Discussion

The DOE, NRC, and State of Texas requirements are comparable.

10.0 Dose Limits for Individual Members of the Public (Dose Rate in Unrestricted Areas)

10.1 DOE

DOE's regulation in 10 CFR 835.602 establishes the expectation that the total effective dose equivalent in controlled areas will be less than 0.1 rem per year. In accordance with 10 CFR 835.602, radioactive material areas have been established for accumulations of radioactive material within controlled areas that could result in a radiation dose of 100 millirem per year or greater. Averaged over a work year, this yields a constant average dose rate of 0.00005 rem per hour. In addition, training and dosimetry are required for individual members of the public for entry into controlled areas, as well as signs at each access point to a controlled area.

10.2 NRC

The NRC regulation in 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

(a) "[C]onduct operations so that –

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with § 35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

(b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals."

10.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.313 mirrors the NRC regulation.

10.4 Discussion

The DOE, NRC and State of Texas requirements are comparable.

11.0 Dose Limits for Individual Members of the Public With Access to Controlled Areas[67]

11.1 DOE

The DOE regulation in 10 CFR 835.208 provides:

> "The total effective dose equivalent limit for members of the public exposed to radiation and/or radioactive material during access to a controlled area is 0.1 rem (0.001 sievert) in a year."

DOE requires training for individual members of the public before entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem in a year

11.2 NRC

The NRC regulation in 10 CFR 20.1301(b), concerning dose limits for individual members of the public, provides in relevant part:

> "if ... members of the public [are permitted] to have access to controlled areas, the limits for members of the public [0.1 rem (1 mSv)] continue to apply to those individuals."[68]

11.3 State of Texas

The State of Texas regulation in the Texas Administrative Code, Title 30, Part 1, Rule §336.313 mirrors the NRC regulation.

11.4 Discussion

The DOE, NRC, and State of Texas requirements in this area are comparable.

12.0 As Low As Reasonably Achievable

12.1 DOE

The DOE regulation in 10 CFR 835.2 defines ALARA as "the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations." The DOE regulation in 10 CFR 835.2 also specifies: "ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits as is reasonably achievable."

12.2 NRC

The NRC regulation in 10 CFR 20.1003 defines ALARA in relevant part: "ALARA . . . means making every reasonable effort to maintain exposures to radiation as far below the dose limits . . . as is practical consistent with the purpose for which the . . . activity is undertaken."

[67] DOE defines a controlled area in 10 CFR 835.2 as "any area to which access is managed by or for DOE to protect individuals from exposure to radiation and/or radioactive material." NRC in 10 CFR 20.1003 defines restricted areas as "an area, access to which is limited ... for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials." The two definitions are essentially the same.

[68] NRC in 10 CFR 20.1301(d) allows licensees to request NRC authorization to allow an individual member of the public to receive up to an annual dose limit of 0.5 rem. DOE in 10 CFR 835 is more restrictive for the dose to an individual member of the public with a limit of 0.1 rem maximum annual dose.

12.3 State of Texas

The State of Texas in the Texas Administrative Code, Title 30, Part 1, Rule §336.2 defines ALARA as follows:

> "Making every reasonable effort to maintain exposures to radiation as far below the dose limits in this chapter as is practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of ionizing radiation and licensed radioactive materials in the public interest."

12.4 Discussion

The DOE, NRC, and State of Texas definitions of ALARA are comparable.

13.0 References

Code of Federal Regulations

10 CFR Part 20, *Standards for Protection Against Radiation*.

10 CFR Part 835*, Occupational Radiation Protection*.

40 CFR Part 61, *National Emission Standards for Hazardous Air Pollutants*.

DOE Orders and Policies

DOE Order 440.1B, *Worker Protection Management for DOE Federal and Contractor Employees*. U.S. Department of Energy, Washington, D.C., May 17, 2007.

DOE Order 458.1, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., June 6, 2011.

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., January 7, 1993.

State Regulations

Texas Administrative Code, Title 30, Part 1, Chapter 336, *Radiation Substance Rules*

APPENDIX E
Consideration of the Criteria in Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005

Appendix Purpose

The purpose of this appendix is to discuss the criteria in Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 with respect to this evaluation.

Appendix Content

This appendix describes the subject criteria in relation to the Department's plans for disposal of the Vitrification Melter.

Key Points

- Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 does not apply to the Vitrification Melter.
- However, disposal of the Vitrification Melter at the Nevada National Security Site or the WCS facility as low-level radioactive waste would be consistent with the criteria of Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005.

1.0 Introduction

Sections 4 through 6 of this evaluation demonstrate that the Vitrification Melter waste package meets the criteria of DOE Manual 435.1-1 for determining that the waste is incidental to reprocessing and is not HLW, and will be managed and disposed of as LLW under DOE's regulatory authority as applicable pursuant to the Atomic Energy Act of 1954, as amended. Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 contains similar criteria, and provides that the Secretary of Energy, in consultation with NRC, may determine that waste resulting from reprocessing of spent nuclear fuel at DOE facilities in South Carolina and Idaho, that is to be disposed of within those states, is not HLW where the criteria in section 3116(a)(1)-(3) are met.[69]

[69] The criteria appear in Subsection (a) of Section 3116. Section 3116(a) provides:

> "IN GENERAL—Notwithstanding the provisions of the Nuclear Waste Policy Act of 1982, the requirements of section 202 of the Energy Reorganization Act of 1974, and other laws that define classes of radioactive waste, with respect to material stored at a Department of Energy site at which activities are regulated by a covered State pursuant to approved closure plans or permits issued by the State, the term 'high-level radioactive waste' does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy (in this section referred to as the 'Secretary'), in consultation with the Nuclear Regulatory Commission (in this section referred to as the 'Commission'), determines—
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical; and
>
> (3) (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of title 10, Code of Federal Regulations, and will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
>
> (B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of –
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations;
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for which is conferred on the State outside of this section; and
> (iii) pursuant to plans developed by the Secretary in consultation with the Commission."

Subsection (b) of Section 3116 addresses monitoring by NRC. Subsection (c) addresses inapplicability to certain materials (i.e., materials transported from the covered State). Subsection (d) identifies the covered States (South Carolina and Idaho.) Subsection (e) addresses certain matters concerning construction of section 3116, and provides that the section does not establish any precedent in any State other than South Carolina and Idaho, and does not amend the West Valley Demonstration Act. Subsection (f) provides for judicial review of determinations made pursuant to section 3116 and of any failure by NRC to carry out its monitoring responsibilities.

Although Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 does not apply to the Vitrification Melter,[70] the following discussion addresses the relevant criteria in 3116(a)(1)-(3) for perspective and information, and, because it may be of interest to stakeholders, shows that disposal of the Vitrification Melter waste package as LLW at the Nevada National Security Site or the WCS facility would be consistent with relevant criteria in Section 3116(a)(1)-(3) of the National Defense Authorization Act for Fiscal Year 2005.

2.0 Consideration of Whether the Vitrification Melter Requires Permanent Isolation in a Deep Geologic Repository

The first criterion or clause in Section 3116(a), as set forth in Section 3116(a)(1), provides that the waste "does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste." DOE Manual 435.1-1 does not contain an identical consideration, but similarly provides in relevant part in Chapter II.B.(2)(a) that the waste "will be managed as low-level waste" and meet the criteria in Section II.B.(2)(a).

With respect to the first criterion or clause, as provided in Section 3116(a)(1), the DOE, in consultation with the NRC, has explained:

> "Clause (1), noted above, is a broader criterion for the Secretary, in consultation with the NRC, to consider whether, notwithstanding that waste from reprocessing meets the other two criteria, there are other considerations that, in the Secretary's judgment, require its disposal in a deep geologic repository. Generally, such considerations would be an unusual case because waste that meets the third criterion would be waste that will be disposed of in a manner that meets the 10 CFR 61, Subpart C performance objectives and either falls within one of the classes set out in 10 CFR 61.55 that the NRC has specified are considered "generally acceptable for near-surface disposal" or for which the Secretary has consulted with NRC concerning DOE's disposal plans. As the NRC explained in *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services)* (NRC 2005), the 10 CFR Part 61, Subpart C performance objectives in turn "set forth the ultimate standards and radiation limits for (1) protection of the general population from releases of radioactivity; (2) protection of individuals from inadvertent intrusion; (3) protection of individuals during operations; and (4) stability of the disposal site after closure." It follows that if disposal of a

[70] That Section 3116(a) applies only to waste from reprocessing at DOE facilities in South Carolina and Idaho, which is to be disposed of in those states, is made clear by the language used, which includes the following:

"(c) INAPPLICABILITY TO CERTAIN MATERIALS. – Subsection (a) shall not apply to any material otherwise covered by that subsection that is transported from the covered State.

(d) COVERED STATES.-- For purposes of this section, the following States are covered States:

(1) the State of South Carolina.

(2) the State of Idaho."

(e) CONSTRUCTION. –

***

(2) Nothing in this section establishes any precedent or is binding on the State of Washington, the State of Oregon, or any other State not covered by subsection (d) for the management, storage, treatment, and disposition of radioactive and hazardous materials.

***

(5) Nothing in this section amends the West Valley Demonstration Act (42 U.S.C.2121a note)."

> waste stream in a facility that is not a deep geologic repository will meet these objectives, in the ordinary case that waste stream does not "require disposal in a deep geologic repository" because non-repository disposal will be protective of public health and safety.
>
> It is possible that in rare circumstances a waste stream that meets the third criterion might have some other unique radiological characteristic or may raise unique policy considerations that warrant its disposal in a deep geologic repository. Clause (1) is an acknowledgement by Congress of that possibility. For example, the waste stream could contain material that, while not presenting a health and safety danger if disposed of at near- or intermediate-surface, nevertheless presents non-proliferation risks that the Secretary concludes cannot be adequately guarded against absent deep geologic disposal. Clause (1) gives the Secretary, in consultation with NRC, the authority to consider such factors in determining whether waste that meets the other two criteria needs disposal in a deep geologic repository in light of such considerations." [71]

That is not the case here. As demonstrated in Section 4 of this evaluation, key radionuclides have been removed from the Vitrification Melter to the maximum extent technically and economically practical. Moreover, the Vitrification Melter waste package will be in a solid physical form and will not exceed the concentration limits for Class C LLW in 10 CFR 61.55, as described in Section 6. As explained in Section 5, management and disposal of the Vitrification Melter as LLW at the Nevada National Security Site or the WCS facility also would meet safety requirements comparable to the NRC performance objectives in 10 CFR 61, Subpart C, so as to provide for the protection of human health and safety and the environment. As such, the disposal of the Vitrification Melter waste package as LLW does not present a danger to human health and safety, such that disposal in a deep geologic repository would be warranted. Furthermore, the Vitrification Melter does not present unique radiological characteristics, or raise non-proliferation risks or other unique policy considerations, which, while not manifesting a danger to human health, nevertheless would command deep geologic disposal. Accordingly, the planned disposal of the Vitrification Melter as LLW at the Nevada National Security Site or the WCS facility meets DOE criteria and would be consistent with the first criterion of Section 3116(a).

3.0 Consideration of Removal of Highly Radioactive Radionuclides

The second criterion of Section 3116(a) specifies that the waste "has had highly radioactive radionuclides removed to the maximum extent practical." DOE Manual 435.1-1, Chapter II.B.(2)(a)1, contains a similar provision, which specifies that such wastes "[h]ave been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical."[72]

Section 4, Table 4-3, of this evaluation identifies key radionuclides for the Vitrification Melter. As can be seen in this table, all radionuclides in Tables 1 and 2 of 10 CFR 61.55 were considered. Furthermore, Section 4 of this evaluation describes how key radionuclides in the Vitrification Melter

[71] *Basis for Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (DOE 2006).

[72] In this regard, NRC staff considers key radionuclides and highly-radioactive radionuclides – which are those radionuclides that contribute most significantly to risk to the public, workers, and the environment – to be equivalent for the purpose of evaluating waste determinations (NRC 2007).

have been removed to the maximum extent technically and economically practical, thus satisfying the DOE criterion and evincing consistency with the second criterion of 3116(a).

5.0 Consideration of Radionuclide Concentration Limits and Waste Disposal Performance Objectives

The third criterion in section 3116(a)(3) concerns whether the waste meets the concentration limits for Class C LLW in 10 CFR 61.55 and whether the waste will be disposed of in accordance with the performance objectives at 10 CFR 61, Subpart C.[73] The criteria in DOE Manual 435.1-1, Chapter II (B)(2)(a)2 and (a)3 similarly provide that waste "[w]ill be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C" and "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 10 CFR 61.55", respectively.

Table 6-1 of this evaluation demonstrates that the Vitrification Melter waste package does not exceed the Class C concentration limits in 10 CFR 61.55 (which are mirrored in the Texas Administrative Code, Rule §336.362, Appendix E). In addition, the Vitrification Melter has been packaged in a shielded shipping container which will be filled with low-density cellular concrete, and thus will be in a solid physical form as discussed in Section 6. Section 4 of this evaluation further shows that management and disposal of the waste will meet safety requirements comparable to NRC performance objectives in 10 CFR Part 61, Subpart C. Given these considerations, management and disposal of the Vitrification Melter as planned meets the above-referenced DOE criteria and would be consistent with the third criterion of Section 3116(a).

6.0 Consultation with NRC

Section 3116(a) also provides for consultation with the NRC. As explained previously, DOE has consulted with NRC concerning this evaluation, as well as making this evaluation available for public review and comment. DOE considered NRC comments, as well as comments from the public, before finalizing the evaluation and before making the final determination that the Vitrification Melter is not HLW. Accordingly, such consultation was consistent with the provision for NRC consultation in section 3116 (a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005.

7.0 References

Federal Statutes

Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, Public Law 108-375, 118 Stat. 1811. 108th United States Congress, October 28, 2004.

Code of Federal Regulations

10 CFR 61.55, *Waste Classification*.

10 CFR 61, Subpart C, *Licensing Requirements for Land Disposal of Radioactive Waste, Performance Objectives*.

[73] Although not germane here, section 3116(a)(3) also provides that the waste be disposed of "pursuant to a State-approved closure plan or State issued permit" for activities regulated by South Carolina or Idaho.

DOE Manuals

DOE Manual 435.1-1, *Radioactive Waste Management Manual*, Change 1. U.S. Department of Energy, Washington, D.C., June 19, 2001.

State Regulations

Texas Administrative Code, Title 30, Part 1, Chapter 336, *Radioactive Substance Rules* (http:// info.sos.state.tx.us/pls/pub/readtac$ext.ViewTAC?tac_view=4&ti=30&pt=1&ch=336)

Other References

DOE 2006, *Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center Tank Farm Facility at the Idaho National Laboratory*, U.S. Department of Energy, Washington, D.C., November 9, 2006.

NRC 2005, *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services)*, CLI-05-05. U.S. Nuclear Regulatory Commission, Washington, D.C., January 18, 2005.

NRC 2007, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations*, NUREG-1854. Draft Final Report, U.S. Nuclear Regulatory Commission, Washington, D.C., August 2007.