



West Valley Demonstration Project

**Draft Waste-Incidental-to-Reprocessing Evaluation
for the Concentrator Feed Makeup Tank
and the Melter Feed Hold Tank**



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Prepared by the
U.S. Department of Energy
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**DRAFT FOR U.S. NUCLEAR REGULATORY COMMISSION
CONSULTATION REVIEW AND PUBLIC REVIEW**

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NOTATION

Acronyms and Abbreviations

ALARA	as low as reasonably achievable
Am	americium
CFMT	concentrator feed makeup tank
CFR	Code of Federal Regulations
CHBWV	CH2M Hill B&W West Valley
Cm	curium
Co	cobalt
Cs	cesium
DOE	Department of Energy
FR	Federal Register
HLW	high-level waste
I	iodine
IP	industrial package
K	potassium
LLW	low-level waste
MFHT	melter feed hold tank
Mn	manganese
Ni	nickel
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]
SA	supplement analysis
Sr	strontium
Tc	technetium
THOREX	thorium uranium extraction [process]
U	uranium
WCS	Waste Control Specialists
WIR	waste incidental to reprocessing
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 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
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
nCi	nanocurie, 10 ⁻⁹ curie
pCi	picocurie, 10 ⁻¹² curie
R	Roentgen
rem	Roentgen equivalent man
s	second
Sv	sievert

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 draft 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 is evaluating whether the concentrator feed makeup tank and the melter feed hold tank from the West Valley Demonstration Project in New York meet the waste-incident-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 concentrator feed makeup tank and the melter feed hold tank are vessels used in the vitrification process to prepare and temporarily store, respectively, slurry consisting of pre-treated high-level waste and glass formers that was supplied to the vitrification melter.
- The Department is responsible for disposal of low-level radioactive waste produced during 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 concentrator feed makeup tank and the melter feed hold tank have been characterized for radioactivity, determined to have radionuclide concentrations that do not exceed limits for Class C low-level radioactive waste, and packaged for shipment to an offsite low-level waste disposal facility.
- To dispose of these vessels offsite as low-level radioactive waste, the Department must first demonstrate that they meet criteria of Department of Energy Manual 435.1-1, *Radioactive Waste Management Manual*, which it has accomplished by the evaluations described in this draft evaluation.
- This draft evaluation is being 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 will make its final determination of whether the concentrator feed makeup tank and the melter feed hold tank are or are not high-level waste after consideration of the Commission's comments and any state and public comments on this draft evaluation.

1.1 Purpose

The purpose of this draft waste-incidental-to-reprocessing evaluation is to evaluate whether the concentrator feed makeup tank and the melter feed hold tank – which contained pretreated high-level waste (HLW) at the West Valley Demonstration Project (WVDP) site in Western New York – meet the waste-incidental-to-reprocessing criteria, are not HLW, and may be managed as low-level waste (LLW) pursuant to Department of Energy (DOE Manual 435.1-1, *Radioactive Waste Management Manual*).

These vessels were used in DOE's process to vitrify liquid HLW, which 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 the two vessels offsite to meet its obligations under the *West Valley Demonstration Project Act of 1980*, which is described below. Note that the concentrator feed makeup tank and the melter feed hold tank are frequently referred to as *the vessels* or *the subject vessels* in the discussions that follow.

Waste Incidental to Reprocessing Requirements and this Draft Evaluation

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 process is intended for those waste streams for which it can be easily determined up front (that is, without detailed analysis) that they do not pose the long-term hazards associated with HLW (DOE Guide 435.1-1). The concentrator feed makeup tank and the melter feed hold tank do not fall into this category.

In the *West Valley Demonstration Project Waste Management Environmental Statement Supplement Analysis* (DOE 2006), DOE stated:

"At this point [in 2006], DOE intends to prepare draft waste-incidental-to-reprocessing (WIR) determinations in accordance with DOE Order 435.1 for the components of the Vitrification Facility included in this SA, as those components [including the subject vessels] have been in direct proximity to HLW in the vitrification process and require a WIR determination to be classified as LLW or another waste type. DOE intends to issue the draft WIR determination for publication in the Federal Register for a 45-day comment period. In the same timeframe, DOE will forward the draft WIR determination to the Nuclear Regulatory Commission for their review in accordance with their responsibilities under the West Valley Demonstration Project Act. At such time as their review is completed, DOE may issue a final WIR determination."

Consistent with this statement, DOE policy, and guidance in DOE Guide 435.1-1, *Implementation Guide for Use with DOE Manual 435.1-1*, this draft evaluation is being issued for NRC review and public comment.

1.2 Scope

This draft waste-incident-to-reprocessing evaluation applies only to the WVDP concentrator feed makeup tank and the melter feed hold tank and to no other equipment. These vessels were previously decontaminated, characterized, and prepared for shipment as discussed further in Section 2 below.

DOE considers this draft evaluation to be a predecisional draft because DOE is consulting with the U.S. Nuclear Regulatory Commission (NRC) and making the draft evaluation available for state and public review before making a final determination. Making this draft evaluation available to the public at the time it is being provided to NRC for review provides stakeholders an opportunity to review it and submit comments to the Department before the final determination is made.

After consideration of NRC, state, and public comments related to this draft evaluation, DOE will issue its final determination on whether the concentrator feed makeup tank and the melter feed hold tank are or are not HLW and may be managed and disposed of as LLW.

1.3 Technical Basis for the Draft Evaluation

This draft 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 involves evaluating whether the concentrator feed makeup tank and the melter feed hold tank are 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; and
- (2) Will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, Performance Objectives; and
- (3) Are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of 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 draft evaluation focuses on the criteria of DOE Manual 435.1-1, Section II.B(2)(a) summarized above, which are discussed in Section 3 of this draft 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*, that Act does not apply to the West Valley site.¹

¹ DOE considered the Section 3116(a)(1) criteria for perspective and information in this waste-incident-to-reprocessing evaluation as explained in Appendix C.

Waste-incident-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-incident-to-reprocessing] criteria are applicable; for example, DOE may decide to apply DOE Order 435.1.”

This evaluation was prepared in accordance with the WVDP procedure for waste-incident-to-reprocessing determinations (WVES 2011a), which is based on requirements in DOE Manual 435.-1 and the associated guidance in DOE Guide 435.1-1, by a consultant experienced in DOE radioactive waste management and waste-incident-to-reprocessing evaluations under the direction and oversight of a manager and a senior engineer from West Valley Environmental Services with similar experience. The quality assurance process followed was consistent with the DOE requirements at 10 CFR Part 830, Subpart A, *Quality Assurance Requirements*, and DOE Order 414.1D, *Quality Assurance*.

Data used in vessel waste package characterization were validated as discussed in Section 2. Calculations performed in support of this draft evaluation were formally documented and peer reviewed. This draft evaluation also underwent detailed reviews by West Valley Environmental Services, CH2M Hill B&W West Valley (CHBWV)², and the DOE Office of Environmental Management for technical adequacy, completeness, correctness, and compliance with applicable requirements. These reviews were formally documented and all review comments incorporated or otherwise resolved.

1.4 Background

The following general information is provided to put the draft evaluation into context. Section 2 provides more detailed background information on reprocessing of spent nuclear fuel, HLW waste pretreatment, HLW vitrification, the vessel design, residual radioactivity in the vessels, 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 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.

² CHBWV replaced West Valley Environmental Services as DOE’s site contractor in September 2011 after a 60-day transition period.

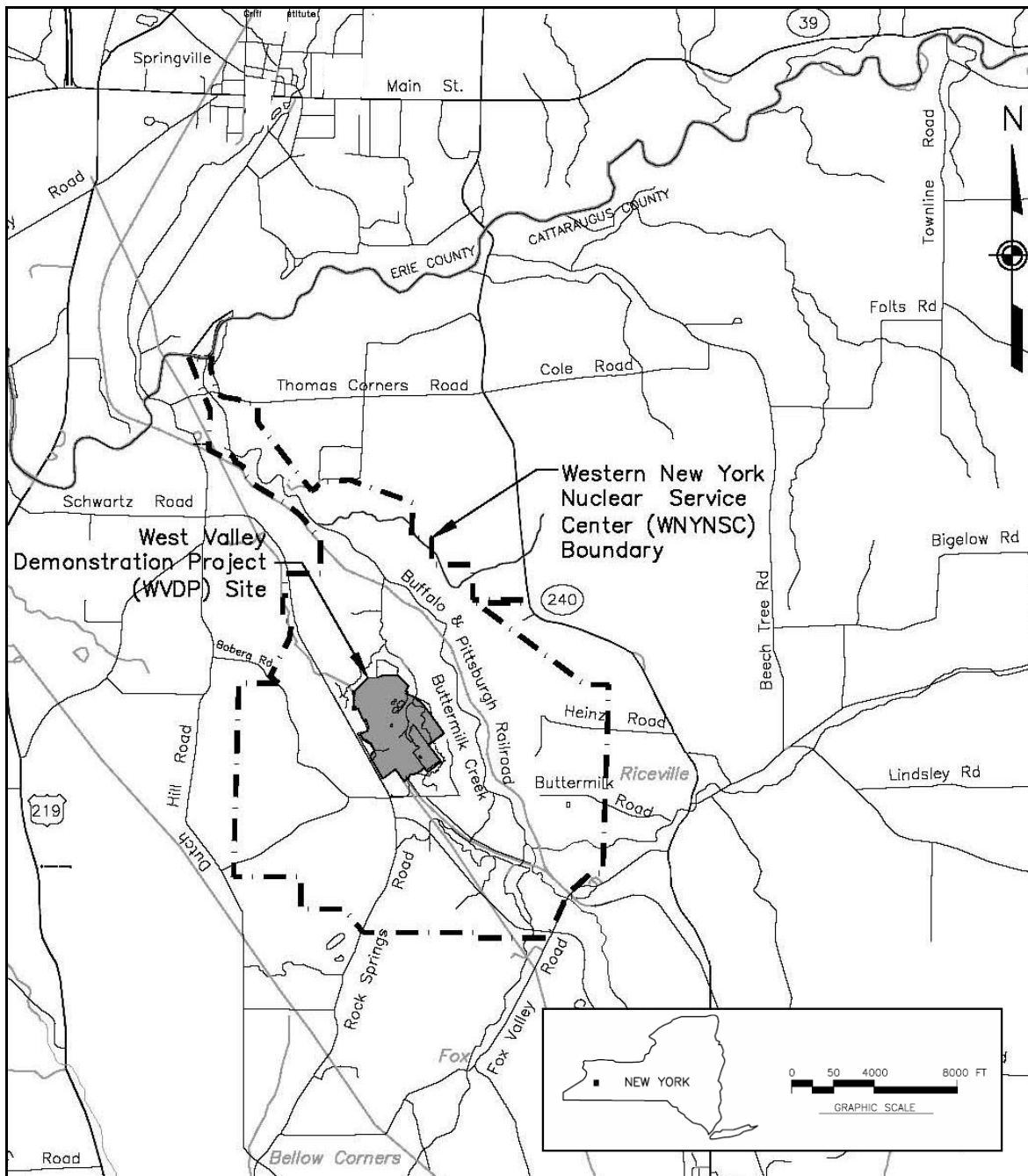


Figure 1-1. The Western New York Nuclear Service Center and the WVDP

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 and the State-Licensed Disposal Area.

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.

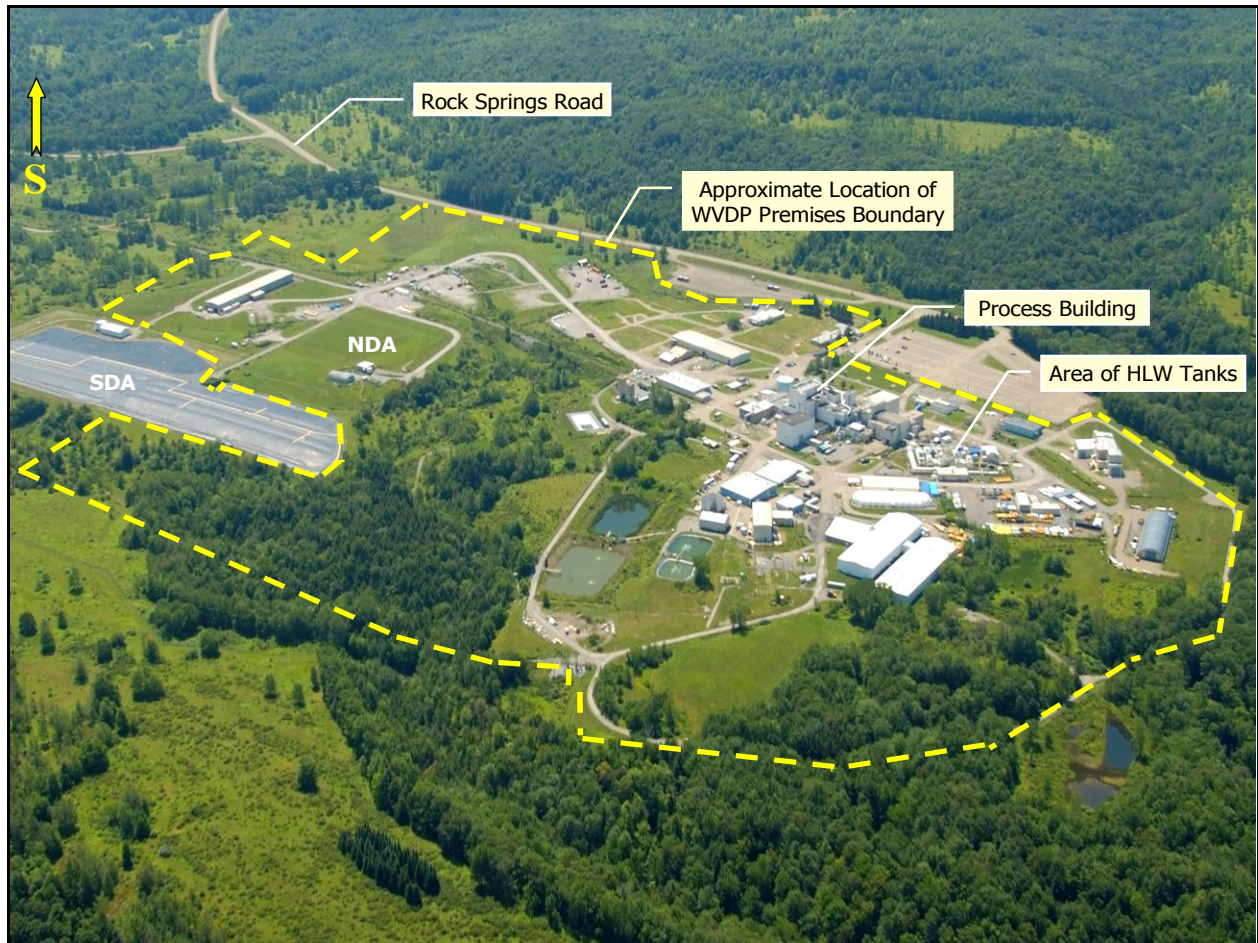


Figure 1-2. The WVDP Area of the Center in 2006 (WVDP photo)

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 uranium (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. (Rykken 1986)

In 1972, Nuclear Fuel Services shut down the reprocessing plant 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 30 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 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).³

1.4.4 Characterization of the Two Vessels

The concentrator feed makeup tank and the melter feed hold tank have been characterized for radioactivity based on measured gamma radiation levels and sample analytical data and found to

³ 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.

not exceed NRC limits for Class C LLW under 10 CFR 61.55 (WMG 2011)⁴. Section 2 of this draft evaluation provides more detail on the characterization process.

1.4.5 Incorporation into a Solid Physical Form

The concentrator feed makeup tank and the melter feed hold tank are both in a solid physical form. Void spaces in both vessels and their waste containers have been filled with grout consisting of low-density cellular concrete to stabilize the vessels within the shipping containers during transport and to encapsulate surface contamination. This grout does not increase the waste disposal volume (which is based on the exterior of the entire disposal package) and was not considered in the classification of the waste.

1.4.6 Potential Waste Disposal Facilities

DOE plans to ship the concentrator feed makeup tank and the melter feed hold tank waste packages 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.⁵

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 is presently inactive. 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 vessels at the Nevada National Security Site, they 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 Texas⁶ for near-surface disposal of Class A, B, and C LLW from Texas Compact⁷ 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 waste⁸. 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. If DOE decides to dispose of the concentrator feed makeup tank and the melter feed hold

⁴ Owing to the concentrations of Cs-137, both vessel waste packages are Class C waste.

⁵ The two vessels constitute a unique waste stream. While it would be physically possible to ship one vessel to one disposal site and the other vessel to the other disposal site, such an arrangement would be costly and inefficient due to factors such as the extensive waste acceptance processes and transportation logistics.

⁶ 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. In April 2012, the Texas Commission on Environmental Quality issued an authorization to accept and begin waste disposal under the license. A request for a contested hearing was granted by the 261st District Civil Court of Travis County Texas in May 2012.

⁷ 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.

⁸ The license does not authorize disposal of greater-than-Class C LLW.

tank at the WCS facility, the waste packages would be disposed of as LLW in the Federal Facility Waste Disposal Facility.⁹

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 vessels 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 concentrator feed makeup tank and the melter feed hold tank. The analysis noted that these vessels 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 draft evaluation, the requirements for disposal of LLW such as the vessel waste packages 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 draft evaluation. Any DOE decision on the facility to which the vessel waste packages would be sent would be made after the final waste-incident-to-reprocessing evaluation and determination, following consideration of NRC and public comments on this draft evaluation, and after DOE confers with appropriate State officials in the states where the waste packages 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 DOE's Office of Environmental Management website (www.em.doe.gov).

⁹ 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, 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 vessels 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.

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 information on the characterization, classification, and packaging for the concentrator feed makeup tank and the melter feed hold tank and concluded that all applicable regulatory requirements had been met (NRC 2004).

1.5 Organization of this Draft Evaluation

Information in the remainder of this draft 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 concentrator feed makeup tank and the melter feed hold tank, including the characterization that has been performed.

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

Section 4 describes how key radionuclides have been removed from the concentrator feed makeup tank and the melter feed hold tank 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 how 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 concentrator feed makeup tank and the melter feed hold tank are less than Class C concentration limits, and that these vessels will be managed in accordance with Chapter IV of DOE Manual 435.1-1.

Section 7 describes the opportunity for NRC and public comments.

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

Section 9 identifies the references cited in the draft evaluation.

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

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

Appendix C 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 vessels, radiological characterization of the vessels, 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 concentrator feed makeup tank is a 6,000-gallon capacity Hastelloy-22 vessel used from 1996 through 2002 to prepare the high-activity waste and glass formers mixture for vitrification.
- The melter feed hold tank is a 5,000-gallon capacity stainless steel vessel used from 1996 through 2002 to supply feed material from the concentrator feed makeup tank to the vitrification melter.
- In 2002, both vessels were flushed, emptied, and shut down.
- The two vessels have been characterized for residual radioactivity using measured dose rates and radionuclide scaling factors based on sample analytical data.
- Based on the characterization results, the radioactivity concentrations in the two vessels are below Class C concentrations limits and the two vessels are Class C low-level waste.
- Each vessel has been loaded in a custom-built steel shipping container in preparation for offsite disposal and the vessel and container filled with grout for stabilization purposes.
- The Department plans to ship the packaged vessels to an offsite waste disposal facility. For purposes of this draft evaluation, this facility is 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 subject vessels 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 draft 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 concentrator feed makeup tank, 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 provides similar information for the melter feed hold tank.
- Section 2.7 describes DOE plans for disposing of the subject vessels.

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 draft evaluation. Section 4 addresses removal of key radionuclides in the subject vessels.

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-Reactor¹⁰ 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.)

¹⁰ N-Reactor was a dual-purpose reactor that produced electricity as well as plutonium.

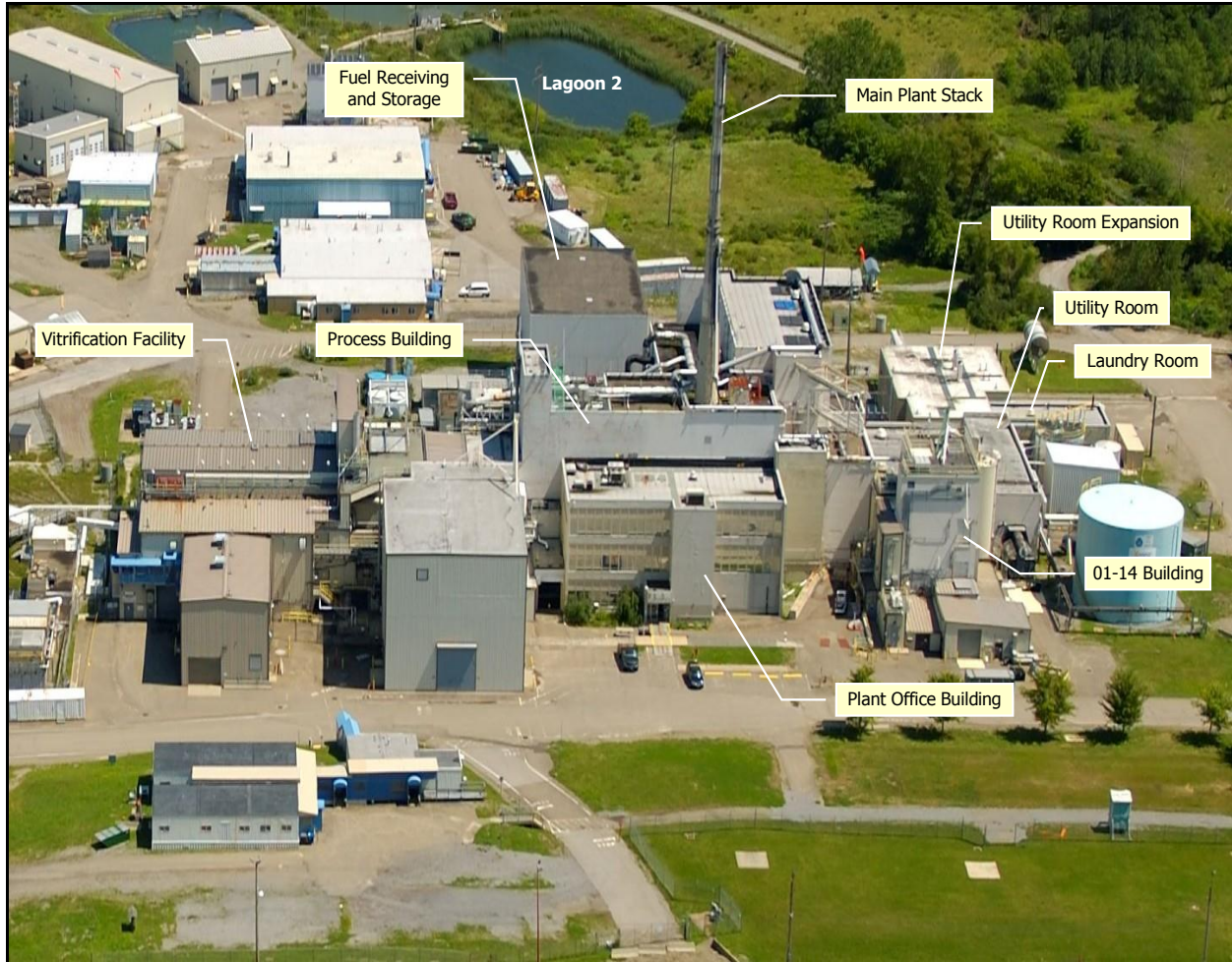


Figure 2-1. The Process Building, the Vitrification Facility, and Ancillary Facilities in 2006 (WVDP photo)

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.

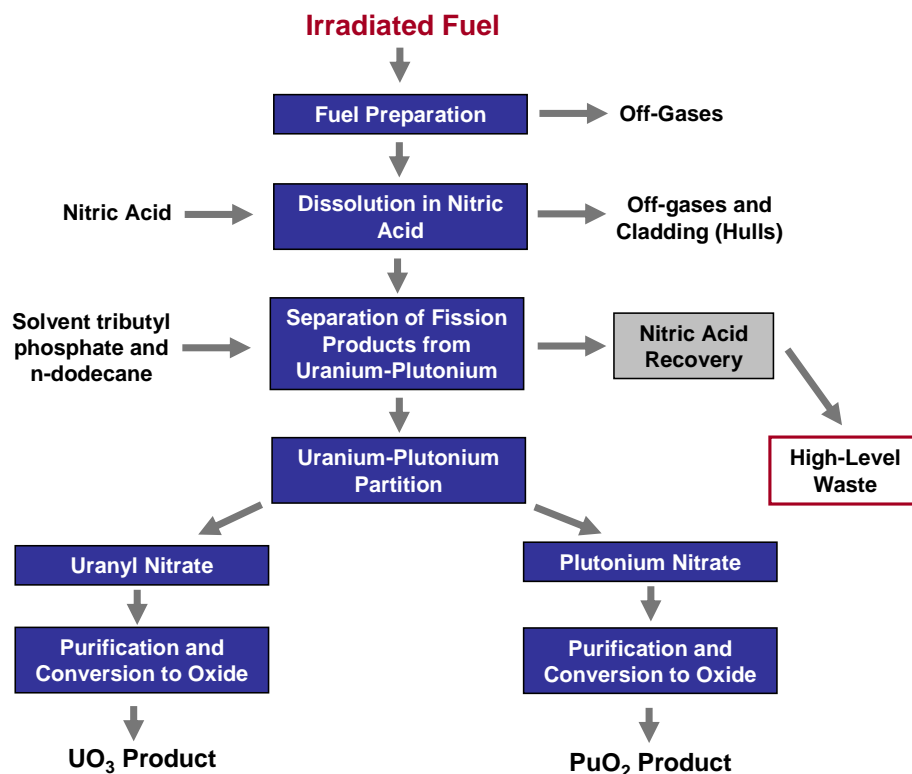


Figure 2-2. Spent Fuel Reprocessing Diagram (PUREX Process)

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 Eisenstatt 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

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) (Continued)

Radionuclide	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
Nb-93m	(1)	2.3E+02	(1)	2.3E+02
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

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) (Continued)

Radionuclide	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
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
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 Premises¹¹. 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 WVDP¹².

2.3.2 Waste Treatment Preparations

To manage the HLW, DOE selected onsite processing using a salt/sludge¹³ 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 exchange¹⁴ columns containing zeolite;
- Adapting existing tanks within the Process Building for use with the Integrated Radwaste Treatment System;

¹¹ 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.

¹² In October 2007, West Valley Environmental Services LLC (WVES) superseded WVNSCO as DOE's site contractor. In September 2011, CH2M Hill-B&W West Valley LLC became the site contractor for Phase 1 decommissioning and facility disposition activities.

¹³ Salt in this context means the liquid portion of the stored waste, i.e., the supernatant.

¹⁴ 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.

- 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 subject vessels.

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.

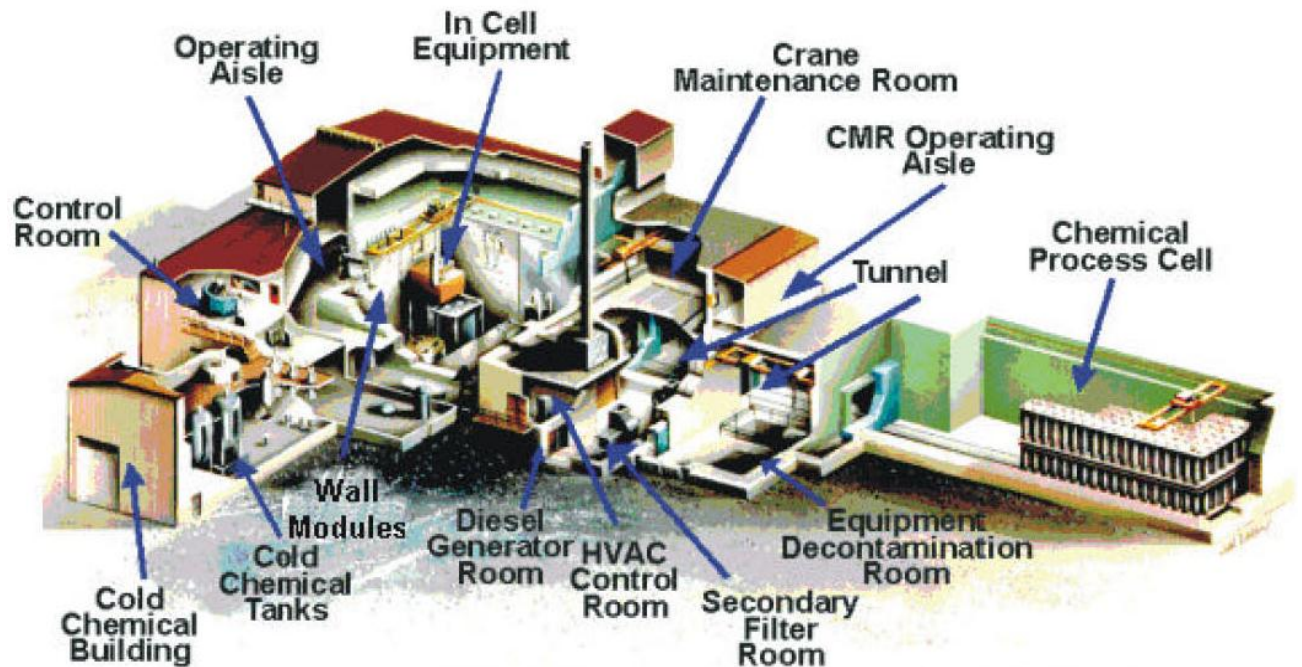


Figure 2-3. Vitrification Facility General Arrangement. (The Equipment Decontamination Room and the Chemical Process Cell, where the HLW canisters were temporarily stored, were part of the original Process Building.)

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 subject vessels in the vitrification process.

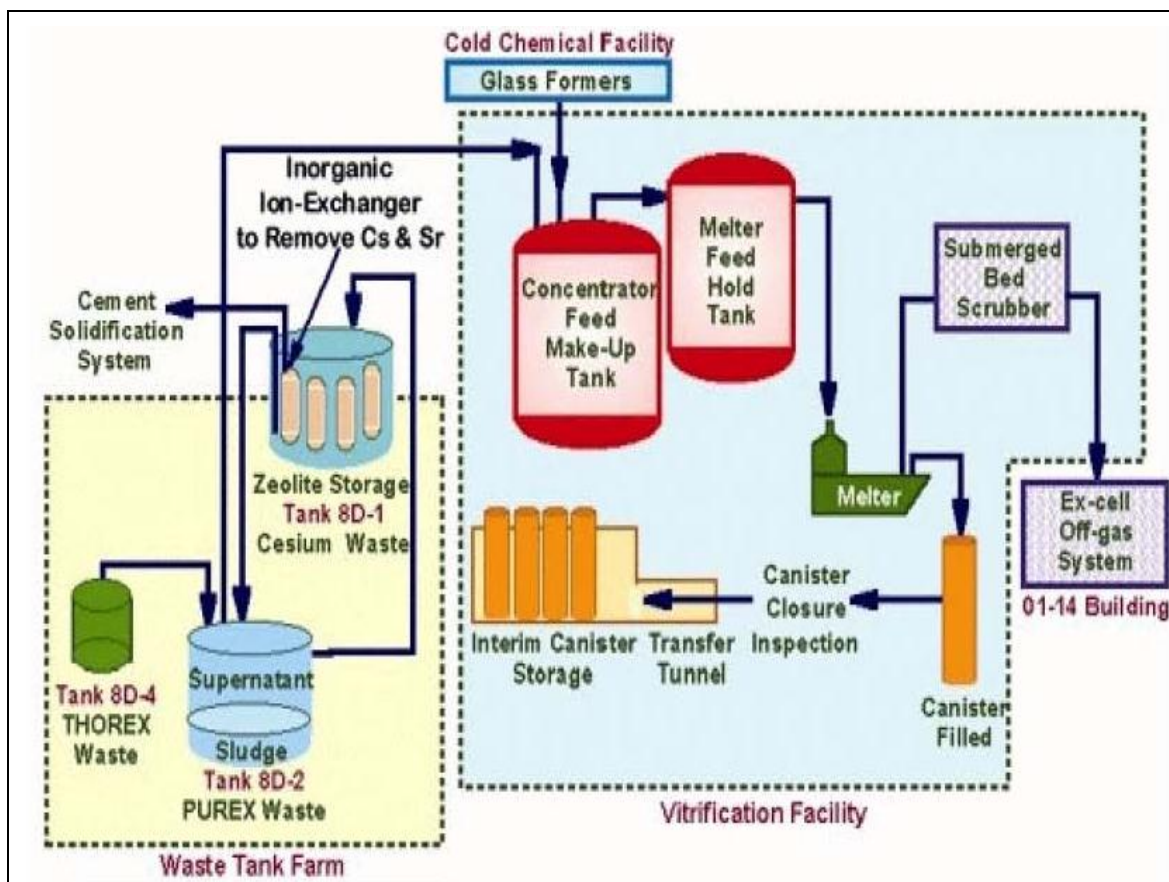


Figure 2-4. Vitrification Process Flow Diagram (for information, not to scale)

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 preparations for Phase 1 of the decommissioning. DOE plans for waste shipment are summarized in Section 2.7 below.

2.5 Concentration Feed Makeup Tank Description, Operation, and Characterization

Figure 2-5 shows the concentrator feed makeup tank in position before the start of the vitrification program.

2.5.1 Description

The concentrator feed makeup tank is a cylindrical vessel approximately 13.5 feet long and 10 feet in diameter with a nominal capacity of approximately 6,000 gallons. Constructed of Hastelloy C-22, it contains an agitation system used to stir its contents and four baffles. The lower part of the vessel exterior is covered with heat transfer coils formed of half-sections of 3.5 inch stainless steel pipe covered with fiberglass insulation and stainless steel sheet. The vessel was supported by a lower skirt attached to a base plate.



Figure 2-5. The Concentrator Feed Makeup Tank (left) and the Melter Feed Hold Tank Installed in the Vitrification Cell (WVDP photo)

2.5.2 Operational History

During use, the concentrator feed makeup tank received pretreated liquid HLW. This waste was combined with the heel remaining from the previous batch and with vitrification process recycle streams. The vessel contents were then sampled and analyzed, and chemicals were added to achieve the required waste form composition. Excess water was removed by evaporation. After verification that the mixture met specifications, the feed slurry batch was transferred to the melter feed hold tank.

Following completion of vitrification in 2002, the concentrator feed makeup tank was extensively flushed as described in Section 4. Preparations for removal of the vessel from the Vitrification Cell included removal of external hardware such as the agitation system motor.

In 2004, the concentrator feed makeup tank was loaded into its shipping container inside the Process Building Equipment Decontamination Room, which adjoins the Vitrification Facility. Figure 2-6 shows the shipping container.



Figure 2-6. The Concentrator Feed Makeup Tank Shipping Container (WVDP photo)

The vessel and the container were then filled with cement grout and relocated to an onsite rail storage area. The loaded shipping container, which is steel shielded and meets Department of Transportation Industrial Package 2 (IP-2) requirements (WMG 2004d), is approximately 19 feet long, 13 feet wide, and 14 feet high and weighs approximately 322,000 pounds. The vessel itself, before grout was added, weighed approximately 18,206 pounds. (CHBWV 2011a)

2.5.3 Characterization

Details of the waste package characterization appear in the waste profile prepared for disposal at the Nevada National Security Site (CHBWV 2011a) and the associated characterization report (WMG 2011). The characterization process made use of sample analytical data and the average measured dose rate of collimated readings taken with a shielded radiation probe one foot from the sides of the installed vessel (1.62 R/h).

A QAD¹⁵ geometry model was used to calculate a dose-to-curie conversion factor for cesium 137, the amount of cesium 137 estimated from the measured dose rate, and the amounts of other radionuclides estimated using radionuclide scaling factors based on sample analytical data. The RADMAN^{TM13} and Megashield^{TM13} computer codes were used in the calculations.

Table 2-2 shows the estimated residual radioactivity in the concentrator feed makeup tank, which totaled 96.5 Ci as of October 1, 2004.

Table 2-2. Concentrator Feed Makeup Tank Total Activity Estimate⁽¹⁾

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
C-14	NA	U-238	6.39E-06
K-40	NA	Np-237	6.66E-05
Mn-54	NA	Pu-238	5.19E-03
Co-60	1.85E-03	Pu-239	1.40E-03
Sr-90	1.12E+00	Pu-240	1.07E-03
Zr-95	NA	Pu-241	1.54E-02
Tc-99	4.17E-03	Pu-242	NA
Cs-137	9.53E+01	Am-241	2.48E-02
Eu-154	5.17E-02	Am-242m	NA

¹⁵ The QAD, RADMANTM and MegashieldTM software are computer codes commonly used in evaluation of radioactive waste packages and associated shielding.

Table 2-2. Concentrator Feed Makeup Tank Total Activity Estimate⁽¹⁾ (continued)

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Th-228	NA	Am-243	2.55E-04
Th-230	NA	Cm-242	9.78E-05
Th-232	2.53E-06	Cm-243	1.55E-04
U-232	1.41E-04	Cm-244	4.02E-03
U-233	5.84E-05	Cm-245	NA
U-234	2.79E-05	Cm-246	NA
U-235	NA		

LEGEND: NA = not available.

NOTES: (1) From WMG 2011 as of October 1, 2004.

The Nevada National Security Site Waste Acceptance Criteria (DOE 2012) includes limits on the amount of plutonium 239 equivalent grams that can be disposed in an individual package. Calculations included in the waste profile show that the plutonium 239 equivalent gram estimate in the concentrator feed makeup tank disposal package falls well below the individual package limit of 300 (CHBWV 2011a)¹⁶.

2.5.4 Characterization Quality Assurance

The characterization performed was consistent with the WVDP characterization process described in the *Characterization Management Plan for the Facility Characterization Project* (Michalczak 2004). This plan provides data quality objectives and describes how data were validated.

The data used in the characterization were supplied by WVNSCO, the site contractor. The dose rate data used were collected for this purpose in February 2004 by WVNSCO using a calibrated instrument – a Geiger-Mueller detector with a Ludlum model 2241 scaler/ratemeter, with the detector shielded to provide collimated measurements (WVNSCO 2004c). The analytical data came from analysis of five samples as shown in the characterization report (WMG 2011). These samples were collected during the 2001 – 2003 period and analyzed by the onsite Analytical and Process Chemistry Laboratory, which was subject to periodic assessments to ensure that analytical requirements were being satisfied (Michalczak 2004). The WMG characterization report was also independently reviewed by the site contractor and changes from this review were incorporated (WMG 2011).

2.5.5 Consideration of Data Variability and Results Uncertainty

The characterization made use of the average value of 10 dose rate measurements to calculate the amount of Cs-137 present in the vessel, and geometric means of sample analytical data to calculate scaling factors used to estimate the amounts of the other radionuclides present. To account for uncertainties in the radionuclide activity estimates, the Nevada National Security Site waste profile radiological technical basis document (CHBWV 2011a) identifies high and low activity ranges that are plus 20 percent and minus 20 percent, respectively, of the final waste form activity concentrations, which are based on the estimates in the characterization report (WMG 2011) and shown in Table 2-2.

¹⁶ The waste profile (CHBWV 2011a) indicates that the maximum plutonium 239 equivalent gram value for the waste stream is 13. The WCS LLW disposal facility does not have plutonium 239 equivalent gram limits.

Consideration of Data Variability

The dose rates measured on the melter feed hold tank varied with location, ranging from 1.17 R/h to 2.25 R/h (WVNSCO 2004c). The average value of 1.62 R/h was used in the calculation, as noted previously. Some variation in dose rates in different locations on the outside of the vessel would be expected due to varying amounts of contamination in the vessel interior and the 1.17 R/h to 2.25 R/h range is consistent with this expectation. The use of an average dose rate value in the calculation is considered to be appropriate since it accounts for the observed variations.

Consideration of the variability in the sample analytical data is more complex because of the multiple data sets used in the calculation and the multiple analyses of the five samples. As discussed later in this draft evaluation, a total of 66 batches of HLW slurry numbered 10 through 75 were prepared for vitrification in the concentrator feed makeup tank, along with two batches of lower-activity vitrification system decontamination solutions identified as batches 76 and 77.

Analytical data used in characterization came from two samples of batch 72 taken at different times, one sample from batch 74, one sample from batch 75, and one sample of residual liquid collected from the vessel after completion of vitrification (WVG 2011)¹⁷. Each of the batch samples was analyzed nine times for most radionuclides of interest (WVNSCO 2002j) and the average values used in the calculations. The July 2003 sample was analyzed three times as shown in the characterization report (WVG 2011). The geometric means of the measured radionuclide concentrations in these samples were used to produce the scaling factors that were used in the characterization (WVG 2011).

The radionuclide concentrations and distributions in the different samples are somewhat different, as can be seen from the table in the report, a condition attributed mainly to differences among the various waste batches¹⁸. However, it is appropriate to consider different waste batches because residual contamination inside the vessel likely built up over time as vitrification proceeded. The use of the geometric means of the concentrations from the different samples is considered to be reasonable for estimating the scaling factors because this approach takes into account the differing radionuclide distributions in the samples.

Consideration of Data Uncertainty

Uncertainty in a typical field dose rate measurement made using a calibrated instrument depends on various factors. However, the use of the average value of the nine measurements taken in various locations on the concentrator feed makeup tank (WVNSCO 2004c) minimizes the uncertainty in the Cs-137 activity estimate.

Regarding the analytical data used to develop scaling factors, the spreadsheet showing analytical data from samples of the various concentrator feed makeup tank batches (WVNSCO 2002j) identifies the uncertainty in the individual measurements. Table 2-3 shows the estimated uncertainty for Cs-137 and Am-241, the two radionuclides most important in classification of the

¹⁷ This information appears in the report (WVG 2011) in a table labeled "Analysis of Multiple Sample Data Sets (SCAL)" marked CFMT.

¹⁸ This table also appears in Attachment A to the Nevada National Security Site waste profile (CHBWV 2011a). It shows measured radionuclide concentrations from analysis of seven samples. The radionuclide concentration data shows some variation in radionuclide distributions. For example, the Sr-90 to Cs-137 ratios in the seven sets of analytical data vary from 1.53E-03 to 7.13E-02. As noted in the text, geometric means of the measured radionuclide concentrations in these samples were used to produce the scaling factors.

concentration feed makeup tank waste package¹⁹, for sets of nine measurements of one sample. For Cs-137, sample 00-1534, the first batch 72 sample is used. For Am-241, sample 01-2498, the batch 75 sample, is used.²⁰

Table 2-3. Measured Radionuclide Concentrations and Associated Uncertainty⁽¹⁾

Cs-137, Sample 00-1534	Analysis	Concentration ($\mu\text{Ci/g}$)	Uncertainty ($\mu\text{Ci/g}$)	Percent Uncertainty
	1	4.70E+02	7.20E+01	15%
	2	4.98E+02	7.63E+01	15%
	3	4.85E+02	3.00E-01	<1%
	4	5.32E+02	8.15E+01	15%
	5	4.68E+02	2.90E+00	1%
	6	4.81E+02	2.90E+00	1%
	7	4.48E+02	2.90E+00	1%
	8	4.81E+02	2.90E+00	1%
	9	5.00E+02	3.10E+00	1%
Am-241, Sample 01-2498	Analysis	Concentration ($\mu\text{Ci/g}$)	Uncertainty ($\mu\text{Ci/g}$)	Percent Uncertainty
	1	3.88E+00	2.33E-01	6%
	2	3.87E+00	2.34E-01	6%
	3	3.72E+00	2.24E-01	6%
	4	4.17E+00	2.51E-01	6%
	5	3.66E+00	2.21E-01	6%
	6	4.04E+00	2.43E-01	6%
	7	3.87E+00	2.33E-01	6%
	8	3.75E+00	2.26E-01	6%
	9	3.80E+00	2.29E-01	6%

NOTE: (1) From WVNSC 2002j.

In these representative examples, the maximum amount of uncertainty in the individual measurements is 15 percent for Cs-137 and six percent for Am-241. The use of the average of multiple measurements minimizes the uncertainty associated with the analytical data used in calculating the scaling factors related to Cs-137.

Conclusions

Given the approach used in the characterization, and the uncertainty in the analytical data, DOE concludes that uncertainty associated with radionuclide estimates is likely bounded by the ± 20 percent concentration range estimates included in the Nevada National Security Site waste profile (CHBWV 2011a).

¹⁹The Class C calculations on page 14 of the waste characterization report (WVG 2011) show Am-241 to dominate the Table 1 sum of fractions and Cs-137 to dominate the Table 2 sum of fractions.

²⁰ These samples were considered representative in that they were from two different batches and the radionuclides were selected because they are important to waste classification as discussed in the previous footnote.

2.5.6 NRC Independent Assessment

In October and November 2004, NRC representatives made monitoring visits to evaluate activities associated with packaging and eventual offsite disposal of the three vitrification process components: the vitrification melter, the concentrator feed makeup tank, and the melter feed hold tank. The NRC representatives determined that the three components were appropriately characterized, packaged, and prepared for offsite disposal in accordance with regulatory requirements. 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). The NRC conclusion independently confirmed the validity of the characterization process used by DOE for the concentrator feed makeup tank.

2.6 The Melter Feed Hold Tank Description, Operation, and Characterization

2.6.1 Description

The melter feed hold tank is similar to the concentrator feed makeup tank in many respects, although slightly smaller as can be seen on Figure 2-5. It is a cylindrical vessel approximately 10 feet long and 10 feet in diameter with a nominal capacity of approximately 5,000 gallons. Constructed of stainless steel, it contains four baffles and an agitation system used to stir its contents. The vessel exterior is partially covered by a cooling jacket. The agitation system was used to maintain homogeneity of the slurry. The vessel was supported by four trunnions.

2.6.2 Operational History

During use, the melter feed hold tank held and mixed HLW slurry feed for delivery to the melter. Following completion of vitrification in 2002, the vessel was extensively flushed as described in Section 4. In 2004, the melter feed hold tank was loaded into its shipping container inside the Process Building Equipment Decontamination Room. The vessel and the container were then filled with cement grout and relocated to an onsite rail storage area.

The loaded melter feed hold tank IP-2 shipping container (WMG 2004e), which is a shorter version of the concentrator feed makeup tank IP-2 container, is approximately 15 feet long, 13 feet wide, and 14 feet high and weighs approximately 272,000 pounds. The vessel itself, before grout was added, weighed approximately 23,486 pounds. (CHBWV 2011a)

2.6.3 Characterization

As with the concentrator feed makeup tank, details of the waste package characterization appear in the waste profile prepared for disposal at the Nevada National Security Site (WVES 2011a) and the associated characterization package (WMG 2011). The characterization process, like that for the concentrator feed makeup tank, made use of sample analytical data and the average measured dose rate of collimated readings taken with a shielded radiation probe one foot from the sides of the installed vessel (1.64 R/h).

The radioactivity in the melter feed hold tank package was estimated using the same process as with the concentrator feed makeup tank, using a QAD geometry model and the RADMAN™ and Megashield™ computer codes. Table 2-4 shows the estimated residual radioactivity in the melter feed hold tank, which totaled 103 Ci as of October 1, 2004.

Table 2-4. Melter Feed Hold Tank Total Activity Estimate⁽¹⁾

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
C-14	3.98E-04	U-236	2.12E-05
K-40	1.54E-03	U-238	4.23E-05
Mn-54	1.67E-03	Np-237	7.26E-05
Co-60	1.58E-03	Pu-238	9.19E-03
Ni-63	1.89E-02	Pu-239	2.28E-03
Sr-90	5.34E+00	Pu-240	1.74E-03
Zr-95	3.72E-02	Pu-241	5.88E-02
Tc-99	8.34E-04	Pu-242	NA
Cs-137	9.71E+01	Am-241	4.33E-02
Eu-154	3.18E-02	Am-242m	NA
Th-228	7.79E-04	Am-243	3.93E-04
Th-230	6.84E-06	Cm-242	3.42E-04
Th-232	7.53E-06	Cm-243	2.84E-04
U-232	9.40E-04	Cm-244	7.36E-03
U-233	3.86E-04	Cm-245	NA
U-234	1.84E-04	Cm-246	NA
U-235	7.07E-06		

LEGEND: NA = not available.

NOTES: (1) From WMG 2011 as of October 1, 2004.

As with the concentrator feed makeup tank, calculations included in the waste profile show that the plutonium 239 equivalent gram estimate in the melter feed hold tank disposal package falls well below the individual package limit of 300 (CHBWV 2011a).

2.6.4 Characterization Quality Assurance

The characterization of the melter feed hold tank waste package, like the characterization of the concentrator feed makeup tank waste package, was consistent with the WVDP characterization process described in the *Characterization Management Plan for the Facility Characterization Project* (Michalczak 2004).

The data used in the characterization were supplied by WVNSCO. The dose rate data used were collected for this purpose in February 2004 by WVNSCO using the same instrument used for the concentrator feed makeup tank (WVNSCO 2004c). The analytical data came from analysis of six samples (WMG 2011), which were analyzed by the onsite Analytical and Process Chemistry Laboratory. And the information on the melter feed hold tank characterization in the WMG characterization report was also independently reviewed by the site contractor and changes from this review were incorporated (WMG 2011).

2.6.5 Consideration of Data Variability and Results Uncertainty

As with the concentrator feed makeup tank, the characterization made use of the average value of dose rate measurements to calculate the amount of Cs-137 present in the vessel, and geometric means of sample analytical data to calculate scaling factors used to estimate the amounts of the other radionuclides present. To account for uncertainties in the radionuclide activity estimates, the Nevada National Security Site waste profile radiological technical basis document

(CHBWV 2011a) identifies high and low activity ranges that are plus 20 percent and minus 20 percent, respectively, of the final waste form activity concentrations, which are based on the estimates in the characterization report (WMG 2011) and shown in Table 2-4²¹.

A brief discussion of data variability and data uncertainty follows.

Consideration of Data Variability

The dose rates measured on the concentrator feed makeup tank varied with location, ranging from 1.35 R/h to 2.39 R/h (WVNSCO 2004c). An average value of 1.64 R/h was used in the calculation, as noted previously. As with the concentrator feed makeup tank, the variations in measured dose rates are consistent with expectations. The use of an average dose rate value in the calculation is considered to be appropriate since it accounts for the observed variations.

Analytical data used in characterization came from four of the samples used in characterization of the concentrator feed makeup tank – two samples of batch 72 taken at different times, one sample from batch 74, one sample from batch 77 – along with two glass shard samples taken from the two evacuated canisters used to remove molten glass from the vitrification melter (WMG 2011)²². The batch samples were the same ones used in the concentrator feed makeup tank characterization. The geometric means of the measured radionuclide concentrations in these samples were used to produce the scaling factors related to Cs-137 that were used in the characterization (WMG 2011).

The radionuclide concentrations and distributions in the different samples are somewhat different, as can be seen from the table in the report, a condition attributed mainly to differences among the various waste batches²³. However, as with the concentrator makeup tank, it is appropriate to consider different waste batches and the use of the geometric means of the concentrations from the different samples for estimating the scaling factors is considered to be appropriate because this approach takes into account the differing radionuclide distributions.

Consideration of Data Uncertainty

The use of the average value of nine dose rate measurements taken in various locations on the concentrator feed makeup tank (WVNSCO 2004c) minimizes the uncertainty associated with the Cs-137 calculation.

Regarding the analytical data used to develop scaling factors, the uncertainties associated with the batch sample results were relatively small as shown in Table 2-3 as discussed previously. Table 2-5 shows uncertainties in Cs-137 and Am-241 for the three analyses of each of the glass shard samples from the evacuated canisters.

²¹ DOE would expect to use the same uncertainty range in the WCS waste profile if the vessel waste packages were to be sent to that facility for disposal.

²² This information appears in the report (WMG 2011) in a table labeled "Analysis of Multiple Sample Data Sets (SCAL)" marked MFHT. Two evacuated canisters were used to remove molten glass from the melter cavity at the conclusion of the vitrification program as discussed later in this draft evaluation.

²³ This table also appears in Attachment B to the Nevada National Security Site waste profile (CHBWV 2011a). It shows measured radionuclide concentrations from analysis of seven samples. As with the concentrator feed makeup tank, the radionuclide concentration data shows obvious variations in radionuclide distributions.

Table 2-5. Measured Radionuclide Concentrations and Associated Uncertainty⁽¹⁾

	Analysis		Concentration ($\mu\text{Ci/g}$)	Uncertainty ($\mu\text{Ci/g}$)	% Uncertainty
	Sample 04-0073	Cs-137	1	2.19E+03	5.58E+01
2			2.26E+03	6.08E+01	3
3			2.46E+03	6.78E+01	3
Am-241		1	1.49E+00	8.98E-02	6
		2	1.51E+00	9.08E-02	6
		3	1.48E+00	8.90E-02	6
Sample 04-0074	Cs-137	1	2.47E+03	6.66E+01	3
		2	2.50E+03	6.69E+01	3
		3	2.40E+03	6.62E+01	3
	Am-241	1	2.22E+00	1.33E-01	6
		2	1.97E+00	1.18E-01	6
		3	1.27E+00	7.64E-02	6

NOTE: (1) From WVNSCO 2004a (sample 04-0073) and WVNSCO 2004b (sample 00-0074).

As can be seen from these representative examples, the maximum amount of uncertainty in the individual measurements is three percent for Cs-137 and six percent for Am-241. The use of the average of the three measurements minimizes the uncertainty associated with the analytical data used in calculating the scaling factors related to Cs-137.

Conclusions

Given the approach used in the characterization, and the uncertainty in the analytical data, DOE concludes that uncertainty associated with radionuclide estimates is likely bounded by the ± 20 percent concentration range estimates included in the Nevada National Security Site waste profile (CHBWV 2011a).

2.6.6 NRC Independent Assessment

As explained in previously, NRC representatives concluded in an assessment performed in connection with two 2004 monitoring visits that the melter feed hold tank was appropriately characterized, packaged, and prepared for offsite disposal in accordance with regulatory requirements (NRC 2004). The NRC representatives evaluated the characterization and waste profile methodologies, the design and fabrication of this waste package, and verification that the package was 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). The NRC conclusion independently confirmed the validity of the characterization process used by DOE for the melter feed hold tank.

2.7 WVDP Waste Management Plans

This section briefly summarizes DOE plans for managing WVDP LLW, including the concentrator feed makeup tank and the melter feed hold tank.

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-incident-to-reprocessing determination using the evaluation process (this draft 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 concentrator feed makeup tank and the melter feed hold tank. The analysis noted that these vessels 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 (WVES and URS 2010). DOE plans to ship the concentrator feed makeup tank and the melter feed hold tank waste packages to an offsite LLW disposal facility. For the purposes of this draft 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 vessel waste packages will be sent will be made after consideration of NRC and public comments on this predecisional draft evaluation, and after DOE confers with appropriate State officials for the states where the waste package may be disposed.²⁴

Section 6 demonstrates that the concentrator feed makeup tank and the melter feed hold tank waste packages do not exceed concentration limits for Class C LLW.

²⁴ 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 subject vessels in the WCS facility.

3.0 WASTE DETERMINATION CRITERIA

Section Purpose

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

Section Contents

This section provides brief background information on Department of Energy criteria that apply to this draft waste-incidental-to-reprocessing evaluation that have been considered, then describes the Department's criteria that apply to management of the concentrator feed makeup tank and the melter feed hold tank.

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 draft evaluation for the subject vessels is being performed in accordance with DOE Manual 435.1-1. DOE's waste determination criteria are described in Section 3.2.

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 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 draft evaluation as a waste determination.

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

- “(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) [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) [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.”²⁵

As will be demonstrated in the next three sections of this draft evaluation, DOE has evaluated the subject vessels against these criteria, and, for the reasons presented, the draft evaluation shows that the vessel waste packages meet the applicable criteria and can be managed and disposed of as LLW.

²⁵ DOE did not authorize alternative requirements for the subject vessels.

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 concentrator feed makeup tank and the melter feed hold tank) have 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 these vessels 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 subject vessels to the maximum extent that is technically and economically practical.
- The key radionuclides in the concentrator feed makeup tank and the melter feed hold tank are those long-lived and short-lived radionuclides listed in Tables 1 and 2 of the Nuclear Regulatory Commission's regulations at 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 flushing the subject vessels with demineralized water under high pressure was the only method technically practical.
- The two vessels were flushed with demineralized water under high pressure, which proved to be effective in removing key radionuclides.
- Other flush solutions that passed through the vessels also likely removed key radionuclides.
- The economic practicality assessment evaluated additional flushing while the vitrification process was still operational and concluded that this approach would not have been economically practical.
- This assessment demonstrated that further efforts to remove key radionuclides would have increased costs and worker radiation dose without resulting benefits.

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

“[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 draft evaluation.

4.1.1 Introduction

The key radionuclides in this draft waste-incident-to-reprocessing 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 435.1-1 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 61.55²⁶;
- 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 subject vessels 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

²⁶ 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 61.55²⁷ 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 435.1-1 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.

²⁷ Title 30 of the Texas Administrative Code has similar requirements (Rule §336.362, Appendix E).

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).

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 draft 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 137²⁸.

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.²⁹

²⁸ 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).

²⁹ 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 two vessels) 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.

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-234³⁰, U-238, and Pu-239 for the Nevada National Security Site Area 5 waste disposal area, based on the following analyses (NST 2012b):

- All-pathway analyses which show that the radionuclide that would contribute most significantly to dose to a member of the public in the controlling scenario (resident farmer) are Tc-99 (79 percent), Pb-210 (13 percent), and U-238 (3 percent), with Pb-210 produced predominately by decay of U-234 in the waste at the time of disposal;
- Intruder analyses which show that the radionuclides that would contribute most significantly to dose to an inadvertent intruder in the chronic agriculture scenario are Tc-99 (70 percent), U-238 (13 percent); and Th-229 (5 percent);
- Intruder analyses which show that the radionuclides that would contribute most significantly to dose to an inadvertent intruder in the acute construction scenario are U-238 (36 percent), Th-229 (25 percent), Pu-239 (8 percent), U-233 (7 percent), and U-234 (7 percent).

The results of the latest performance assessments of the Nevada National Security Site LLW disposal areas are discussed in Section 5.2 of this draft evaluation.

For the commercial WCS LLW disposal facility, the Radioactive Material License (TCEQ 2012) 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 5.2³¹. 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.

4.1.6 Conclusions About Key Radionuclides in the Subject Vessels

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 draft 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.

³⁰U-234 present at the time of disposal is the predominant source of Pb-210 (NST 2012b). 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.

³¹ 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 89, TCEQ 2012).

Table 4-3. Key Radionuclides for this Draft 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	
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 (NST 2012b).
 (3) Alpha emitting transuranic radionuclides with half-life greater than five years (NRC 1982, Table 4.2).

4.2 Removal to the Maximum Extent Technically and Economically Practical

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* (DOE 2006).

The Meaning of the Phrase "Technically and Economically Practical"

Removal to the maximum extent "technically and economically practical" is not removal to the extent "practicable" or theoretically "possible." Nor does the criterion connote removal which may be notionally capable of being done. 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 removal is not sensible or useful in light of the overall benefit to human health and the environment. Such a situation may arise if certain radionuclides are present in such extremely small quantities that they make an insignificant contribution to potential doses to workers, the public, and the hypothetical human intruder.

Because of the close relationship between the two vessels, this section addresses both together. Section 4.2.1 describes the technical practicality assessment of methods that might have been used to remove key radionuclides from the vessels. Section 4.2.2 discusses the method actually used and its effectiveness. Sections 4.2.3 and 4.2.4 discuss two other methods considered that were not technically practical. Much of the information in this section is drawn from three technical documents:

- A 2002 report on flushing of the melter feed hold tank (WVNSCO 2002b),
- A 2002 report describing the equipment and methods used and the results of this decontamination program (WVNSCO 2002c), and
- The *HLW Processing System Flushing Operations Run Plan* (WVNSCO 2002a).

4.2.1 Technical Practicality Assessment

The WVDP considered various proven decontamination technologies that might be suitable for removing key radionuclides from the two vessels, including those discussed in DOE's 1994 *Decommissioning Handbook* (DOE 1994). This handbook discusses 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 were developed with the support of the DOE Office of Science and Technology (now the DOE Office of Science) as part of a continuing program to improve methods used in decontamination and decommissioning work.

DOE's 1994 *Decommissioning Handbook* identifies advantages and disadvantages of decontamination technologies in various applications. For the subject vessels, the technologies proven to be effective in decontaminating pipes and tanks were the ones of primary interest. Table 9.2 of the Handbook showed the following technologies to be highly effective in this application: (1) ultra-high pressure water, (2) grit blasting, (3) flushing with water, and (4) hydroblasting³².

Grit blasting did not show promise in this application owing to potential for the grit material interfering with the vitrification process and resulting in disposal difficulties. The other three methods are generally similar except for the water pressure used. High-pressure spray, a combination of the other three methods, was selected for evaluation and actually used as the primary decontamination method, as discussed below.

The methods evaluated in detail were:

- Flushing vessel internals with water using high-pressure spray,
- Mechanical decontamination using a ball mill, and
- Chemical decontamination.

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 practical.

Prior to decontamination, the insides of the two vessels were expected to be relatively clean in the lower two thirds, with the upper portions coated with residual dried HLW slurry. Dead spaces inside the nozzles of the concentrator feed makeup tank had been observed to be coated with slurry. Dried slurry was also expected on the stiffeners of the head of the melter feed hold tank, which are illustrated in Figure 4-1. (WVNSCO 2002a)

The high-pressure spray method is discussed first, followed by the other alternatives that were evaluated but not used. While the high-pressure spray method was the primary method used to removal key radionuclides from the vessels, it was not the only method that was used as discussed in Section 4.3.

4.2.2 In-Place Decontamination By Flushing With Water Using High-Pressure Spray

The concentrator feed makeup tank and the melter feed hold tank were decontaminated by flushing with demineralized water before they were removed from the Vitrification Facility. This flushing was performed in connection with flushing of other HLW processing systems (WVNSCO 2002a). The process involved use of high-pressure water spray to clean the inner surfaces of the

³²Other decontamination processes described in *The Decommissioning Handbook* (ASME 2004) were considered in a retrospective fashion when this evaluation was prepared. It is unlikely that any of these other processes – which include soft-media blasting and steam vacuum cleaning – would have been more effective than the high-pressure water spray actually used, which was very effective based on the visual inspection results and dose rate reductions that are discussed later in this subsection and shown in Figure 4-2.

two vessels. It was completed in 2002 prior to shutdown of the vitrification melter so the material collected during the flushing could be vitrified for disposal as HLW, which was done to the extent practical.³³

Flushing Equipment Design and Flushing Performed

Substantial development work was performed to determine the optimum process for spraying the vessels interiors with high-pressure water (WVNSCO 2002f). This flushing would focus on the upper third of each vessel.

Special fixtures to position the spray nozzles inside the upper portions of the vessels were developed and tested on a mockup. Figure 4-1 shows spray nozzle positions considered. A separate spray wand for decontamination of the vessel exteriors was designed, built, and tested. Tooling was also developed to clean the vessel nozzles using rotary driven wire brushes. However, these tools were not used because the water spray proved to be effective in nozzle areas, as well as in other parts of the vessels. (WVNSCO 2002f)

Pressurized water at approximately 1,000 pounds per square inch was delivered at 25 to 35 gallons per minute through the Gamajet® rotary spray nozzle. The nozzle was inserted through one or more openings in the vessel heads to ensure complete coverage of internal surfaces. Each complete spray cycle took approximately 24 minutes. (WVNSCO 2002f)

The concentrator feed makeup tank was flushed three times using this process and the melter feed hold tank was also flushed three times using this process. Each such flush involved two complete spray cycles that together took 55-60 minutes to accomplish. (WVNSCO 2002f)

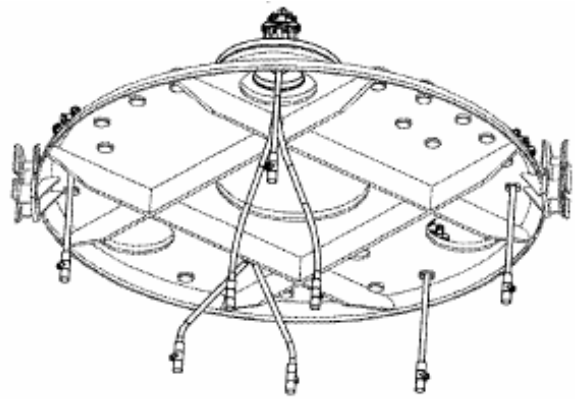


Figure 4-1. Melter Feed Hold Tank Head Showing Potential Spray Nozzle Positions. (Two positions were actually used in the flushing.) (From WVNSCO 2002c)

The resulting flush solutions were concentrated in the concentrator feed makeup tank, combined with other flush solutions that were also concentrated and then diluted with glass formers and, when this mixture met specifications, transferred to the melter feed hold tank for feeding to the melter. For flushing the melter feed hold tank, that vessel was used as its own catch tank, after which the flush solution was transferred to the concentrator feed makeup tank for evaporation/concentration and subsequent sampling. After this material was combined with other flush solutions and glass formers, it was transferred back to the melter feed hold tank for feeding to the melter. (WVNSCO 2002a)

The various flush solutions that were used for other equipment in the Vitrification Cell and the waste tank farm, including nitric acid as well as water, were transferred into the concentrator feed makeup tank and the melter feed hold tank, with these liquids also having a decontamination effect by loosening deposits (WVNSCO 2002a). In addition, the external surfaces of the two vessels were washed down using high-pressure demineralized water from a spray wand, which was suspended

³³ 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, including the subject vessels, prior to completing a controlled shutdown of the melter.

from the cell crane. Flushing results were monitored primarily by visual inspections and dose rate measurements. (WVNSCO 2002f)

Visual Inspection Results

A camera with underwater lighting was used to obtain still images of the vessel internal surfaces to determine the effectiveness of the flushing. Figure 4-2 shows before and after flushing conditions in the two vessels.



Figure 4-2. Before and After Flushing Images of the Two Vessels. (The shiny surfaces in the after images illustrate the effectiveness of the high-pressure washing in these WVDP photos taken with the underwater camera.) (from WVNSCO 2002c)

Inspection before flushing showed substantial amounts of dried slurry in both vessels as can be seen in Figure 4-2. Inspection after flushing completion showed essentially no visible deposits, with visibility sufficient to show fabrication weld beads and threads on bolts.

Dose Rate Reduction and Reduction in Residual Radioactivity

Radiation detectors positioned using special fixtures to monitor decontamination progress showed that flushing reduced dose rates, although the amount of dose rate reduction was masked to some degree by other radiation sources in the cell, which included filled HLW canisters. Dose rates measured near the head of the concentrator feed makeup tank dropped from 200 to 8 R/h. Dose rates near the melter feed hold tank dropped from 250 to 22 R/h³⁴.

NOTE

Additional dose rate measurements made in February 2004 (WVNSCO 2004c) showed lower levels, with a maximum of 2.25 R/h on the concentrator feed makeup tank and a maximum of 2.39 R/h on the melter feed hold tank. The 2004 data, which were used in vessel characterization (WVG 2011), are indicative of a decontamination factor for the flushing of 89 for the concentrator feed makeup tank and 105 for the melter feed hold tank. (For the concentrator feed makeup tank, the decontamination factor is based on the initial measurement of 200 R/h divided by the final measurement of 2.25 R/h and, for the melter feed hold tank, the initial measurement of 250 R/h was divided by the final 2.39 R/h.)

An estimate of the flushing effectiveness can also be made by comparing the estimated residual activity in each vessel before flushing and after all of the flushing was completed. Table 4-4 shows estimated residual cesium 137 in the two vessels before flushing and after completion of flushing, when the vessels were drained to the extent practical.

Table 4-4. Vessel Flushing Effectiveness in Terms of Estimated Cs-137 Removal

Condition	CFMT Remaining Inventory (Ci)	CFMT Reduction Factor (%)	MFHT Remaining Inventory (Ci)	MFHT Reduction Factor (%)
Before Flushing ⁽¹⁾	1,500	NA	1,200	NA
After All Flushes	96.5 ⁽²⁾	93	103 ⁽³⁾	91

LEGEND: CFMT = concentrator feed makeup tank, MFHT = melter feed hold tank, NA = not applicable

NOTES: (1) The activity in each vessel before flushing began was estimated in the following manner: (a) the dried slurry coating observed on the vessel interior surfaces before flushing was assumed to average 0.250-inch thickness over the upper one-third of the vessels, a conservatively low value based on pre-flush visual inspection results; and (b) the Cs-137 concentration in this material was assumed to be the same as that in the last batch of slurry sent to the concentrator feed makeup tank before the vitrification system flushing began (batch 75), which had the second highest Cs-137 concentration (1.16E+04 $\mu\text{Ci}/\text{cm}^3$) among the feed material batches (Kurasch 2011).

(2) From Table 2-2 above.

(3) From Table 2-3 above.

The estimates in Table 4-4 should be considered to be order-of-magnitude estimates. As can be seen in Figure 4-2, the sludge buildup in some areas was much greater than 0.250 inch. The

³⁴The pre-flush dose rates were recorded on May 1, 2002 (WVNSCO 2002b). The post-flush dose rates were recorded on July 15, 2002 after all of the high-pressure spray flushing had been completed (WVNSCO 2002d).

area values used in the estimates did not include the areas of the internal baffles. These factors suggest that the before-flushing estimates in Table 4-4 are low.

Evaluation of conditions before and after flushing and considering the measured dose rates and the estimates on Table 4-4 indicate that flushing removed more than 90 percent of the residual cesium 137 inside the concentrator feed makeup tank and the melter feed hold tank. That is, the overall decontamination factor for the flushing performed – the “direct” flushes using the high-pressure spray apparatus and the “indirect” flushes associated with other vitrification flush solutions passing through the two vessels – was of the order of 10 based on the estimated residual sludge removal or 100 based on the reduction in measured dose rates, with the latter considered to be a more accurate estimate due to the uncertainty in the sludge removal estimates.

The flushing Plan (WVNSCO 2002a) identified the expected conditions after the flushes for both vessels as *“Dried slurry deposits are expected to be removed from the surfaces accessible to the spray head, with some removal from protected areas.”* The significant differences between the before and after surface conditions inside the vessels, combined with the dose rate reductions and the residual radioactivity reductions summarized in Table 4-4, demonstrate that these objectives were achieved and that the flushes were effective in removing key radionuclides.

Additional information on the overall effectiveness of the various flushes in reducing residual radioactivity in the concentrator feed makeup tank and the melter feed hold tank is provided in Section 4.3 below, including a table that summarizes all of the flushes performed – the “direct flushes” using the high-pressure spray, the “indirect flushes” associated with other vitrification facility flush solutions that passed through the vessels, and additional flushes that were performed prior to removal of the vessels in 2004.

4.2.3 Mechanical Decontamination Using a Ball Mill

This process uses a combination of grinding and impact in a ball mill to separate radioactive contamination from the base material.³⁵ It 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. The WVDP concluded that the process could be optimized, but that this effort would change the proven technology to an experimental one, resulting in another disadvantage. The WVDP determined that this process was not technically practical because of these disadvantages

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.

(WVNSCO 2001). Given this conclusion, details of how to use the process on the two vessels and how to dispose of the resulting waste stream were not developed.

4.2.4 Chemical Decontamination

This method uses a chemical solution to dissolve the slurry material or leach radionuclides from the glass or slurry.

Like ball milling, this process was evaluated for potential use in decontamination of small pieces of equipment and other materials contaminated with HLW during the vitrification process. Evaluation included laboratory tests where specimens consisting of Inconel and stainless steel plates with glass annealed to their surfaces and pieces of pipe containing dried slurry were placed in various chemical solutions at different temperatures. Solutions tested included nitric acid, hydrofluoric acid, oxalic acid, sodium hydroxide, and water. (WVNSCO 2001)

These tests demonstrated that chemical dissolution could be an effective decontamination process for the equipment. However, the presence of the chemicals could have been incompatible with the recipe for producing an acceptable borosilicate glass matrix in the vitrification process. If this were to happen, the resulting out-of-specification HLW product would have had no approved pathway for disposal. This overriding disadvantage made chemical decontamination an unacceptable approach from a technical standpoint.³⁶

4.3 Economic Practicality Assessment

The assessment of the economic practicality of further radionuclide removal focused on the costs and benefits of performing additional flushing before shutdown of the vitrification system because the flushing process used was the only method among those evaluated for removal of key radionuclides that was determined to be technically practical. Section 4.2.2 showed that the flushing process performed:

- Removed more than 90 percent of the estimated residual radioactivity in the concentrator feed makeup tank, leaving approximately 96.5 curies;
- Removed more than 90 percent of the estimated residual radioactivity in the melter feed hold tank, leaving about 103 curies; and
- Removed visible dried slurry from the surfaces of both vessels.

4.3.1 Economic Practicality of Additional Flushing

The assessment considered the benefits and costs of one additional flushing cycle for each vessel. For conservatism, it assumed that this additional flushing cycle would have removed 90 percent of the activity remaining in that vessel at the conclusion of the flushing that was actually performed³⁷. To help establish the context for the cost-benefit assessment, this section begins with a more detailed discussion of the flushes performed and their effectiveness in removing key radionuclides.

Direct and Indirect Flushes

Table 4-5 summarizes the flushing of the two vessels, both direct flushes using the high-pressure spray and indirect flushes that involved temporary storage and agitation of vitrification

³⁶ Note that the concentrator feed makeup tank was actually flushed with sodium hydroxide solution before removal as discussed in Section 4.3 below, but the flush solution was sent to Tank 8D-4 since the vitrification system had already been shut down.

³⁷ Ninety percent effectiveness would have been very unlikely in these circumstances, given the extensive previous flushing which removed all of the visible sludge from both vessels.

system flush liquid of decreasing radioactive concentrations or additions of uncontaminated water. The table also includes a final flush of the concentrator feed makeup tank that was performed just prior to removal of the vessels from the vitrification facility in 2004. The batch numbers refer to batches of material made up in the concentrator feed makeup tank.

A total of 66 batches of HLW slurry numbered 10 through 75 were sent to the concentrator feed makeup tank to be prepared for vitrification, along with two batches of lower-activity vitrification system decontamination solutions identified as batches 76 and 77. Batch 76 contained about 51 percent of the radioactivity in batch 75 and batch 77 contained about 11 percent of the radioactivity in batch 75 (Kurasch 2011).

Table 4-5. Summary of Direct and Indirect Flushes Performed ⁽¹⁾

Concentrator Feed Makeup Tank	Flush 1 (indirect)	Batch 76, which included material from acid flush of Tank 8D-4, stored in the vessel and periodically stirred for five months between January 2 and June 1 of 2002. After the prepared batch 76 material was transferred from the CFMT to the MFHT, 2,000 liters (530 gallons, at least 10 inches in the vessel) of water were added to the CFMT and transferred to the MFHT.
	Flush 2 (direct)	Spray nozzle deployed inside CFMT on May 2, 2002 per WO 63499 through Nozzle F, operated for two cycles (55-60 minutes). ⁽²⁾ Dislodged material becomes part of batch 76 material.
	Flush 3 (indirect)	After transfer of batch 76 from the CFMT to the MFHT, at least 530 gallons (2,000 liters) of uncontaminated water transferred to CFMT and on May 31, 2002 was sent to MFHT. ⁽⁴⁾
	Flush 4 (indirect)	Batch 77 periodically stirred by the CFMT's agitator for two months (June 5, 2002 through July 31, 2002).
	Flush 5 (direct)	Spray nozzle deployed inside CFMT on June 14, 2002 per WO 65670 through Nozzle F, operated for two cycles. ⁽⁵⁾ Dislodged material became part of batch 77 material.
	Flush 6 (direct)	Spray nozzle deployed inside CFMT on June 18, 2002 through Nozzle F, operated for two cycles. Dislodged material became part of batch 77 material.
	Flush 7 (indirect)	Water-diluted MFHT Heel after airlift ⁽⁶⁾ 16 from the final production canister WV-412 and the MFHT water flush from September 10, 2002 transferred to the CFMT on September 10, 2002, raising the level in the CFMT from 58 inches to 86 inches. This dilute liquid stirred by the CFMT agitator periodically between September and December 2002, while the liquid level in the CFMT is reduced to about 34 inches.
	Flush 8 (direct)	In late December 2003, with vessel level at about 34 inches, about 55 gallons of 50% sodium hydroxide solution and 30 gallons of non-radioactive water added to CFMT. Agitator operated for 30 minutes and this material remained in vessel for about one month ⁽⁷⁾
	Flush 9 (direct)	In late January 2004, the CFMT agitator was run for at least one hour and the CFMT contents transferred to Tank 8D-4. About 200 gallons of utility water transferred to the CFMT, the agitator operated for at least one hour, and the vessel contents transferred to tank 8D-4, leaving a heel of about 55 gallons. ⁽⁸⁾

Table 4-5. Summary of Direct and Indirect Flushes Performed ⁽¹⁾ (continued)

Melter Feed Hold Tank	Flush 1 (indirect)	Batch 76 transferred from the CFMT to the MFHT, filling the MFHT to the 101.7 inch level. Batch 76 material periodically stirred and fed to the melter for 16 days until June 14, 2002.
	Flush 2 (direct)	Spray nozzle deployed inside MFHT Nozzle E on July 8, 2002 per WO-65708 and operated for two cycles. Decontamination solution collected and transferred to the CFMT to become part of batch 77. ⁽³⁾
	Flush 3 (direct)	Spray nozzle deployed inside MFHT Nozzle E on July 11, 2002 per WO-65708 and operated for two cycles. ⁽³⁾ The resulting decontamination solution transferred to CFMT became part of Batch 77.
	Flush 4 (direct)	Spray nozzle deployed inside MFHT Nozzle R on July 11, 2002 per WO-65708 and operated for two cycles. ⁽³⁾ Resulting decontamination solution transferred to CFMT became part of batch 77.
	Flush 5 (indirect)	Batch 77 material transferred from CFMT to MFHT on July 31, 2002, filling MFHT to the 98 inch level. Batch 77 material periodically stirred and fed to the melter for 10 days (August 5 through August 14, 2002.)
	Flush 6 (indirect)	Water added continuously to the MFHT after airlift 8 of the last production canister, diluting the batch 77 material for final airlifts 9 through 16. These final 8 airlifts take 30 hours during which water is added and the contents are periodically stirred by the MFHT agitator. This was completed on August 14, 2002. ⁽⁹⁾
	Flush 7 (direct)	On September 10, 2002, MFHT heel transferred to the CFMT. MFHT subsequently filled with 22 inches of water and periodically stirred for a minimum of two hours, after which contents transferred to the CFMT that same day. Afterwards, the MFHT level instrument showed that the vessel was empty.

LEGEND: WO = work order

- NOTES: (1) This table was compiled from vitrification system records, including the completed work orders listed in notes (2), (3), (4), and (6).
- (2) WVNSCO 2002b.
 - (3) WVNSCO 2002d.
 - (4) WVNSCO 2002i.
 - (5) WVNSCO 2002c.
 - (6) Air pressure was used to raise the molten glass level in the vitrification melter discharge cavity to send the molten glass into the waiting HLW canister. This process was called airlifting.
 - (7) WVNSCO 2003.
 - (8) WVNSCO 2004d.
 - (9) WVNSCO 2002h.
 - (10) Water and dislodged waste from spraying the inside of the concentration feed makeup tank became part of the material batch inside the vessel at the time the spray was performed.

As can be seen in Table 4-5, the concentration feed makeup tank received three separate, two-cycle flushes with high-pressure spray, was indirectly flushed four times, was flushed with small amounts of caustic solution and water, and, finally, flushed with approximately 200 gallons of water in early 2004. The melter feed hold tank received three separate, two-cycle flushes with high-pressure spray, was indirectly flushed three times, and, finally, flushed with uncontaminated water and the heel from the concentrator feed makeup tank. Figure 4-3 illustrates the direct flushes of the two vessels using the high-pressure spray apparatus.

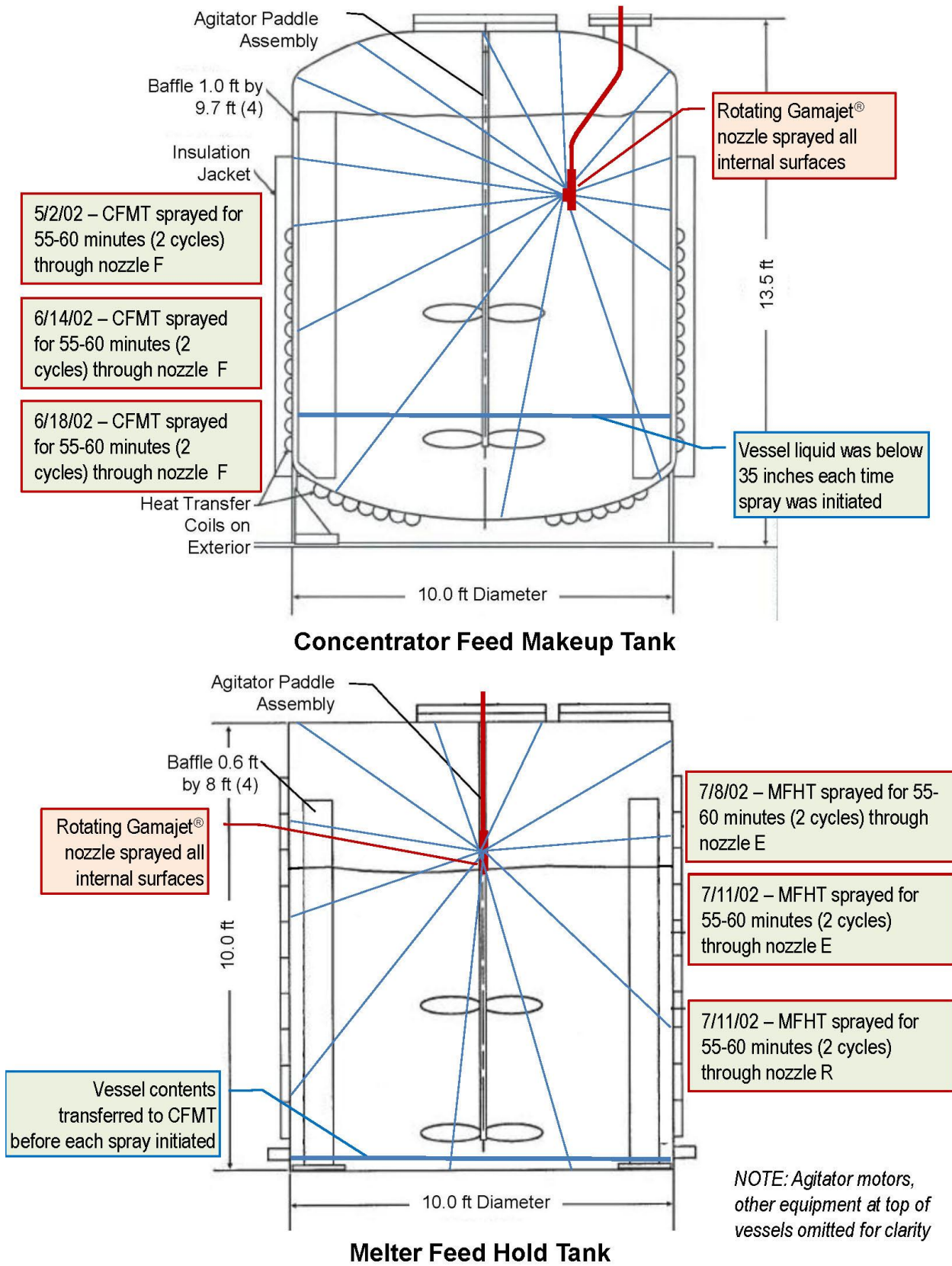


Figure 4-3. High-Pressure Spray of the Concentrator Feed Makeup Tank and the Melter Feed Hold Tank

Figure 4-4 illustrates how liquid levels in the concentrator feed makeup tank varied in the first seven months of 2002 when batches 76 and 77 were in the vessel and the direct flushes and some of the indirect flushes were taking place. The figure shows wide variations in the liquid levels over this period. Note that the increases in the figure were from incoming material transfers and the decreases were from evaporation, except for the identified transfers to the melter feed hold tank.

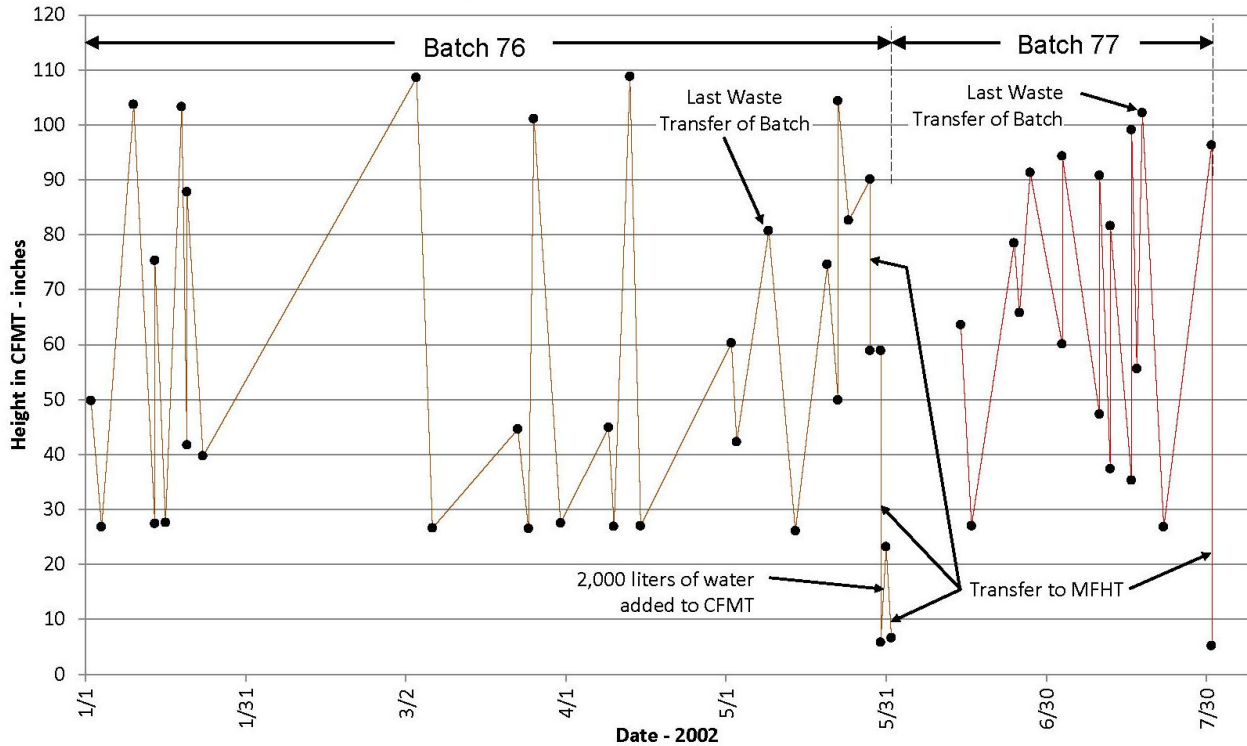


Figure 4-4. Variations in Concentrator Feed Makeup Tank Liquid Levels (WVNSCO 2002h)

Based on the results of the visual inspections, the reductions in dose rates, and the reductions in estimated residual radioactivity discussed previously, it is clear that the direct and indirect flushes effectively removed key radionuclides from the two vessels. The high-pressure spray had the greatest effect. The indirect flushes also had some effect.

The overall results show major decreases in residual radioactivity. Figure 4-5 illustrates the overall reductions in the maximum measured dose rates and the estimated cesium 137 activity resulting from the direct and indirect flushes shown in Table 4-5.

Although the figure plots the reduction in Cs-137, all key radionuclides were removed in approximately the same proportions based on the similarities of the radionuclide distribution in the material removed from the vitrification melter in the two evacuated canisters³⁸ to the radionuclide distribution in batches 74 and 75, the last batches of HLW slurry sent to the concentrator feed makeup tank. The measured dose rates are primarily from cesium 137 since it is the dominant gamma emitting radionuclide in the vessels as can be seen from Table 2-2 and Table 2-3.

³⁸As noted previously, two evacuated canisters were used to remove molten glass from the melter cavity at the conclusion of the vitrification program. Comparing 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 other key radionuclides were removed in slightly higher proportions than Cs-137.

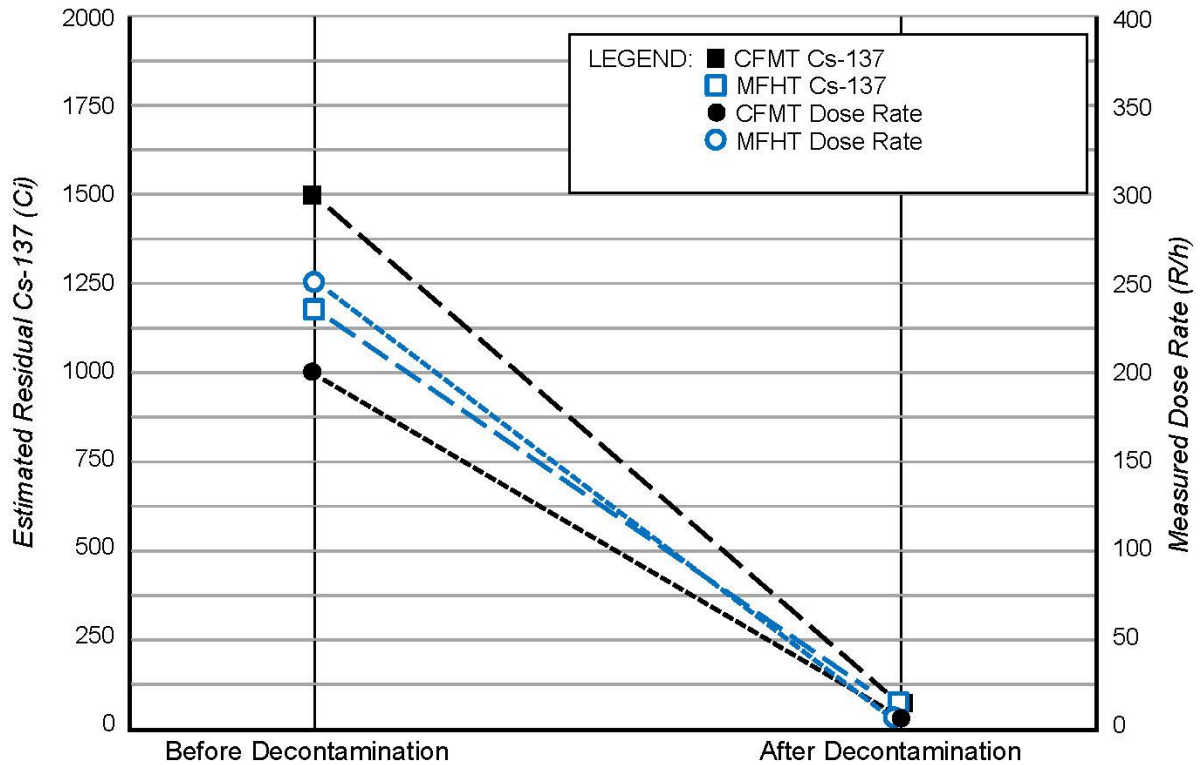


Figure 4-5. Vessel Flushing Effectiveness

Potential Benefits

The benefits from the hypothetical additional flushes would have been limited for the following reasons:

- The dose rates on the outside of the waste package in its present condition are low – a maximum of 16 mR/h on contact with the side of the concentrator feed make up tank and a maximum of 5 mR/h on contact with side of the melter feed hold tank (WVES 2011b). Compliance with radiological control program requirements in handling 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 less shielding would have been approximately the same.
- The flushes actually performed left both vessels 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 vessel waste packages 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 vessel waste packages at the LLW disposal facility without further decontamination would 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.

A single monetary benefit would have been realized had the hypothetical additional flushes been performed: The shielded containers for the vessels could have been designed and constructed of lighter weight steel, which would have reduced costs associated with materials, fabrication, and transportation by approximately \$200,000.³⁹

Costs of Additional Flushing

A total of approximately \$1 million in additional direct costs (in 2002 dollars) would have been involved for each additional direct flush.

One additional direct flush cycle for each vessel would have produced approximately 1,400 gallons of additional liquid to be processed through the vitrification melter. Even though this volume would have been reduced by evaporation, processing of the remaining liquid would still have required one additional processing cycle which would have produced one more canister of vitrification waste from the vitrification melter. 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 year⁴⁰.

Another factor in considering the costs of processing of additional flush solutions was the limited vitrification melter service life. By the time the flushes had been completed, the melter was near the end of its useful service life. A failure of the melter would have, for all practical purposes, stranded radionuclides within the melter.⁴¹

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

4.3.2 Summary and Conclusions

The technical practicality assessment showed that flushing the vessels with water using high-pressure spray was the only method among the three methods evaluated that would be technically practical. This method proved to be effective. The discussion of the economic practicality of additional flushing while the vitrification process was still operational showed that the cost of an additional high-pressure spray flush would have far outweighed the benefits.

³⁹ 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)

⁴⁰ In addition, there would be indirect life-cycle costs. Interim onsite storage of one additional canister would have cost around \$35,000 per year based on actual and predicted annual costs of interim storage, including surveillance and maintenance. The estimate does not include the cost of ultimate disposition. (Kurasch 2011)

⁴¹ As discussed previously, the WVDP used processing of decontamination solutions and the evacuated canister system to remove residual radioactive material from the melter. Those processes were effective in removing residual waste from the melter, but could only be used for waste in a molten glass form, before the glass solidified. It would not have been feasible to process decontamination solutions or use the evacuated canister system in the event of a failure of the melter, because upon melter failure, the residual glass in the melter cavity would solidify, thereby precluding use of these processes to clean the melter and effectively stranding the solidified waste in a solid form within the melter.

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 concentrator feed makeup tank and the melter feed hold tank waste packages 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 vessel waste packages 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 vessel waste packages will meet criteria for disposal as low-level radioactive waste.

Key Points

- Management of the concentrator feed makeup tank and the melter feed hold tank waste packages will meet safety requirements comparable to the performance objectives of 10 CFR 61, Subpart C.
- The concentrator feed makeup tank and the melter feed hold tank waste packages 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 concentrator feed makeup tank and the melter feed hold tank waste packages 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 concentrator feed makeup tank and the melter feed hold tank and their operational histories. Section 2 also explains how these vessels were characterized for residual radioactivity, with Tables 2-2 and 2-3 providing the total activity estimates in the final waste form. As noted previously, each vessel has been placed in its own custom made shipping container and the void spaces in the vessels and in the shipping containers 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, 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 vessel waste packages meet 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 vessel waste packages meet these criteria if the waste packages 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*,⁴² 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 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.⁴³

⁴² 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. The technical purpose and scope of the order remain the same.

⁴³ Appendix A of this draft evaluation demonstrates that the DOE performance objectives provide safety requirements comparable to the NRC performance objectives at 10 CFR 61, Subpart C and that 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.

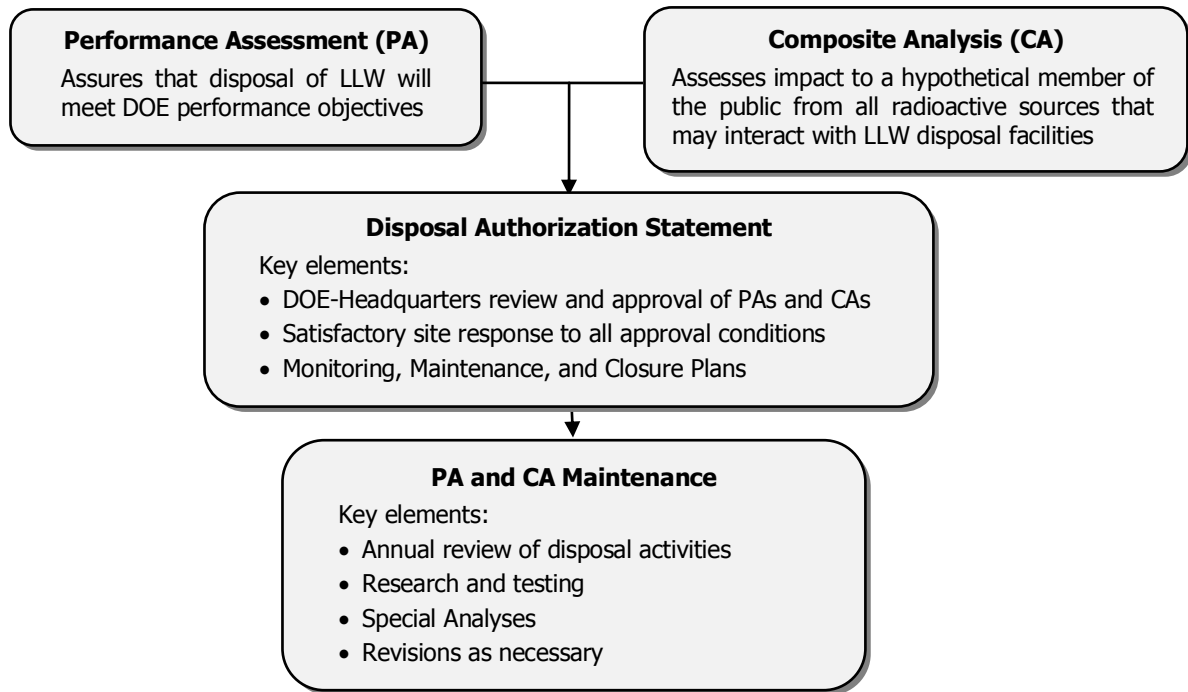


Figure 5-1. General Process Used to Ensure Performance Objectives Are Achieved

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.”

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 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/m²/s (0.74 Bq/m²/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.”

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 methodology⁴⁴. 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."

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 2012 (NST 2012a). This report addresses matters such as new or revised waste streams, monitoring results, research and development, the inventory estimates at planned closure, updated performance assessment results, and updated composite analysis results. It also identifies special analyses that were performed in the previous year⁴⁵.

As explained in the report of the annual review (NST 2012a), the updated Area 5 performance analyses provide 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:

⁴⁴ 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. Appendix B demonstrates that the DOE and NRC dose standards are comparable and shows that the State of Texas dose standards mirror those of NRC.

⁴⁵ 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.

- All pathways dose for members of the public, with the predicted peak annual dose of 1.9 millirem estimated to occur at 1,000 years after planned closure (i.e., in 3028, 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 of 0.045 millirem to a resident farmer at 1,000 years after facility closure; and
- The average Radon 222 flux density at the surface of the disposal units, which is predicted to reach a peak of 4.3 pCi/m²/s 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. Note that the estimates given are mean values. The 95th percentile values are also below the performance objective dose limits (NST 2012a).

Estimated Impact of Disposal of the Subject Vessels

Disposal of the concentrator feed makeup tank and the melter feed hold tank at the Nevada National Security Site Area 5 facility would have a negligible impact, if any, on facility performance for the following reasons:

- The vessel waste packages meet the waste acceptance criteria, with no radionuclide concentrations exceeding the waste acceptance criteria action levels as shown in Table 5-1 below.
- Screening of the waste profile (CHBWV 2011a) by the Nevada National Security Site has shown that no special analysis is required for this waste stream, confirming that the vessel waste packages do not have the potential for altering disposal site performance assessment assumptions, conceptual models, or performance assessment results.⁴⁶

A waste profile package for the subject vessels, which describes their characteristics 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 September 2011 (CHBWV 2011a). This 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 vessel waste packages for disposal, conditioned upon a determination that the waste packages are LLW in accordance with the waste-incident-to-reprocessing criteria in DOE Manual 435.1-1 (that is, this draft evaluation resulting in a final evaluation and determination that the vessel waste packages can be managed as LLW) (DOE 2011). Section 5.3 below discusses the waste acceptance process in more detail.

⁴⁶Note that a special performance assessment was performed for disposal of the vitrification melter waste package at the Nevada National Security Site Area 5 facility, which showed that the melter waste package would have a negligible impact on performance of the closed facility (DOE 2010). The vitrification melter waste package has been estimated to contain approximately 4,570 curies as of October 2004 (WVG 2004a). The concentrator feed makeup tank and melter feed hold tank waste packages are each about the same size as the vitrification melter waste package and each contains only about two percent of the activity in the melter waste package, with similar radionuclide distributions. Given this situation, it is evident that the impact of disposal of the two vessel waste packages will have negligible, if any, impact on performance of the Nevada National Security Site Area 5 facility since the vitrification melter waste package that contains much more radioactivity has been determined to have a negligible impact on performance of this disposal facility.

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. It included the following estimated dose for the post-institutional control period: a maximum annual dose of 0.0064 millirem per year to an adjacent resident, a small fraction of the 25 millirem per year limit⁴⁷. 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).

Consideration of the license limitations and the available performance assessment results for the WCS Federal Facility Waste Disposal Facility suggests that disposal of the concentrator feed makeup tank and melter feed hold tank waste packages at that facility would have negligible, if any, impact on disposal facility performance for the following reasons:

- As shown in Table 5-2 below, the activity in each of the vessel waste packages – 96.5 curies for the concentrator feed make up tank and 103 curies for the melter feed hold tank – would amount to only about 0.002 percent of the license limit for the Federal Facility Waste Disposal Facility.
- The 5.6 million curies in 26 million cubic feet of waste considered in the performance assessment would have an average activity density of 0.22 Ci per cubic foot, much higher than the activity density of the concentrator feed makeup tank waste package (approximately 0.03 Ci per cubic foot) and the melter feed hold tank waste package (approximately 0.04 Ci per cubic foot).

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.”

NRC in 10 CFR 61.42 sets forth the following requirements:

⁴⁷ This estimate is for the Federal Facility Waste Disposal Facility Canister Disposal Unit, where the vessel waste packages would be disposed of.

“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.”

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 equivalent 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.⁴⁸

Assessment of Area 5 Performance

The report of the most recent (2011) 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 2012a) 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:

- The drilling worker intruder scenario, with the predicted peak annual acute dose of 3.4 millirem 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 of 130 millirem 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.⁴⁹ (NST 2012a)

Estimated Impact of Disposal of the Subject Vessels

Disposal of the concentrator feed makeup tank and the melter feed hold tank at the Area 5 facility will have a negligible impact, if any, on facility performance with respect to inadvertent intruders, for the same reasons that disposal of these vessels would have a negligible impact, if any, on facility performance with respect to protecting members of the public. As discussed previously, this conclusion is based on the following factors:

- The vessel waste packages meet the waste acceptance criteria, with no radionuclide concentrations exceeding the waste acceptance criteria action levels as shown in Table 5-1 below.
- Screening of the waste profile (CHBWW 2011a) by the Nevada National Security Site has shown that no special analysis is required for this waste stream.

⁴⁸ 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 A.

⁴⁹ 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 (NST 2012a). 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.

Assessment of WCS Federal Facility Waste Disposal Facility Performance

The updated WCS performance (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 WCS annual limit⁵⁰.

Consideration of the license limitations and the available performance assessment results for the WCS Federal Facility Waste Disposal Facility suggests that disposal of the concentrator feed makeup tank and melter feed hold tank waste packages at that facility would have negligible, if any, impact on dose to the inadvertent intruder for the following reasons:

- As shown in Table 5-2 below, the activity in each of the vessel waste packages – 96.5 curies for the concentrator feed make up tank and 103 curies for the melter feed hold tank – would amount to only about 0.002 percent of the license limit for the Federal Facility Waste Disposal Facility.
- The 5.6 million curies in 26 million cubic feet of waste considered in the performance assessment would have an average activity density of 0.22 Ci per cubic foot, much higher than the activity density of the concentrator feed makeup tank waste package (approximately 0.03 Ci per cubic foot) and the melter feed hold tank waste package (approximately 0.04 Ci per cubic foot).

5.2.4 Protection of Individuals During Operations

The DOE requirements in DOE Manual 435.1-1, Section I.E(13), for protection of individuals 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.”

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 B.

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 458.1⁵¹ 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

⁵⁰ These estimates are for the Federal Facility Waste Disposal Facility Canister Disposal Unit where the vessel waste packages would be disposed of.

⁵¹ Section I, 1. E (13) of DOE Manual 435.1-1 cross-references 10 CFR Part 835 as well as prior DOE Order 5400.5. Prior DOE Order 5400.5 has been cancelled and replaced by DOE Order 458.1.

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. 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.

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 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 preparing the vessel waste packages 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 vessel waste packages 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 Part 835. These programs also comply with various DOE directives (including 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 2010). 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 one foot from the sides of the vessel waste packages range from 0.1 to 10 millirem per hour for the concentrator feed makeup tank and from 0.1 to 3.5 millirem per hour for the melter feed hold tank (WVES 2011b). Workers involved with handling of the waste packages 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 vessel waste packages 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 vessel waste packages and the short duration of the work to prepare the waste packages for shipment. Furthermore, the work associated with preparing the waste packages 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 vessel waste packages for shipment are well below the applicable limit⁵². This work will be performed within a radiologically controlled area within the WVDP security fence. Past

⁵² 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. The two vessels waste are essentially ready for shipment, except for loading them on the transport vehicles.

WVDP experience with similar waste management work indicates that potential doses to the public will be very low. In 2009, 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 2011b). The airborne pathway is the only pathway of interest for potential exposure to a member of the public from onsite work to prepare the subject vessels 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 vessel waste packages 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 vessel waste packages at the Nevada National Security Site are also expected to be very low. Operations to dispose of the vessel waste packages 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 vessel waste packages 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.⁵³

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:

"Disposal Facility Closure Plans (DOE Manual 435.1, Section IV.Q(1)(a) and (b)). 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]."

"Disposal Facility Closure (DOE Manual 435.1, Section IV.Q(2)(c)). Institutional control measures shall be integrated into land use and stewardship plans and programs, and shall

⁵³ The updated WCS performance assessment (WCS 2011) provides an estimated average dose to workers during normal operations at the Federal Waste Facility of 95 mrem per year.

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].”

As discussed in Appendix A, 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.”

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 2012a). The plan will ensure that the applicable requirements of DOE Order 458.1 (which replaced prior DOE Order 5400.5 referenced in DOE Manual 435.1-1) 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 effective dose equivalent. The airborne emissions limit of 10 mrem per year effective dose equivalent (40 CFR 61.92) also applies to emissions of radionuclides to the ambient air from DOE facilities.

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 Vessels Will Meet 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 vessel waste packages 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 vessels meet 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 provides specific radionuclide waste acceptance criteria for LLW (DOE 2012) 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 2012) 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 2012)

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.^{54,55}

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 2012),
- (2) Any alpha-emitting transuranic radionuclide with half-life over 20 years that exceeds 10 pCi/g,
- (3) Any radionuclide whose concentration exceeds one percent of the total activity concentration.

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

⁵⁴ 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 2012).

⁵⁵ 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 10 CFR 61.55, concerning classification of low-level waste, NRC includes 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 the U.S. Environmental Protection Agency'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 2012). Table E-1 provides radionuclide action levels for waste characterization and reporting purposes.

specified to these ends is verified by DOE in a comprehensive performance assessment program, as noted previously.

5.3.2 The Vessel Waste Packages

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 vessel waste packages to the Nevada National Security Site waste acceptance criteria and determined that it will meet these criteria. For example, the melter feed hold tank contained an estimated 16.8 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 (CHBWV 2011a).

5.3.3 The Vessels Meet 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 2012).

As described in Section 2, DOE has packaged each vessel in a custom-built, steel-shielded IP-2 container and the internal void spaces have been filled with low-density cellular concrete. The low-density cellular concrete is compatible with the waste acceptance criteria and will help reduce the possibility of disposal cell subsidence in the area of the buried waste containers by eliminating the container void spaces.

As noted previously, a waste profile package for disposal of the vessel waste packages (CHBWV 2011a) was submitted to the Nevada National Security Site and approved conditional upon the determination that the vessels are not HLW and can be managed as LLW (DOE 2011). Table 5-1 compares the radionuclide concentrations in the vessels with the radionuclide action levels for waste characterization and reporting provided in the Nevada National Security Site Waste Acceptance Criteria (DOE 2012). Radionuclides with concentrations exceeding one percent of the action level are highlighted in the table.⁵⁶

⁵⁶ As noted previously, the waste acceptance criteria document (DOE 2012) require 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 subject vessels were rigorously characterized by the WVDP.

Table 5-1. Vessel Radionuclide Concentrations in Bq/m³(1)

Nuclide	AL ⁽²⁾	CFMT ⁽³⁾	MFHT ⁽³⁾	Nuclide	AL ⁽²⁾	CFMT ⁽³⁾	MFHT ⁽³⁾
H-3	6.2E+11	NA	NA	U-232	4.3E+10	NR ⁽⁴⁾	NR ⁽⁴⁾
C-14	5.4E+15	NA	NR ⁽⁴⁾	U-233	8.2E+10	NR ⁽⁴⁾	NR ⁽⁴⁾
K-40	9.4E+10	NA	NR ⁽⁴⁾	U-234	1.3E+10	NR ⁽⁴⁾	NR ⁽⁴⁾
Mn-54	NA	NA	NR ⁽⁴⁾	U-235	1.1E+10	NA	NR ⁽⁴⁾
Fe-55	NA	NA	NA	U-236	2.8E+11	NA	NR ⁽⁴⁾
Ni-59	1.7E+14	NA	NA	U-238	3.5E+11	NR ⁽⁴⁾	NR ⁽⁴⁾
Co-60	1.6E+12	NR ⁽⁴⁾	NR ⁽⁴⁾	Np-237	3.4E+10	NR ⁽⁴⁾	NR ⁽⁴⁾
Ni-63	3.2E+14	NA	NR ⁽⁴⁾	Pu-238	1.8E+12	6.32E+06	1.60E+07
Sr-90	4.3E+11	1.36E-09	9.32E+09	Pu-239	5.1E+11	1.70E+06	3.98E+06
Zr-95	NA	NA	NR ⁽⁴⁾	Pu-240	5.2E+11	1.30E+06	3.04E+06
Tc-99	3.2E+09	NR ⁽⁴⁾	NR ⁽⁴⁾	Pu-241	5.8E+12	NR ⁽⁴⁾	NR ⁽⁴⁾
I-129	3.4E+09	NA	NA	Pu-242	3.7E+11	NA	NA
Cs-137	2.5E+11	1.16E+11	1.69E+11	Am-241	1.7E+11	3.02E+07	7.56E+07
Pm-147	NA	NA	NA	Am-243	5.8E+10	3.10E+05	6.86E+05
Eu-154	1.7E+12	NR ⁽⁴⁾	NR ⁽⁴⁾	Cm-242	NA	NR ⁽⁴⁾	NR ⁽⁴⁾
Th-228	4.3E+13	NA	NR ⁽⁴⁾	Cm-243	8.3E+11	1.89E+05	4.96E+05
Th-229	2.8E+10	NA	NA	Cm-244	3.4E+12	NR ⁽⁴⁾	NR ⁽⁴⁾
Th-230	6.0E+07	NA	NR ⁽⁴⁾	Cm-245	4.6E+10	NA	NA
Th-232	8.1E+09	NR ⁽⁴⁾	NR ⁽⁴⁾	Cm-246	9.2E+10	NA	NA

LEGEND: AL = action level, CFMT = concentrator feed makeup tank, MFHT = melter feed hold tank, NA = not available, NR = not required to be reported because the radionuclide is <1% of the action level or <1% of the total activity concentration.

- NOTES: (1) To convert Bq/m³ to pCi/L multiply by 0.027.
 (2) From the Table E-1 of the Waste Acceptance Criteria (DOE 2012).
 (3) From the vessel waste profile technical basis document (CHBWV 2011a). These values are based on the waste volume; values based on the waste package volume are lower.
 (4) The waste profile package (CHBWV 2011a) includes high and low activity range values for these radionuclides, which are not reportable.

Table 5-1 shows that the estimated concentrations of all radionuclides fall below the action levels.

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 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 2012) contains additional requirements related to waste disposal, including total waste volume limitations and total activity limitations for certain radionuclides. Table

5-2 shows representative requirements compared to the related parameters for the vessel waste package.

TABLE 5-2. Key WCS Federal Facility Waste Disposal Facility License Requirements

Requirement (Section)⁽¹⁾	License Limit⁽¹⁾	CFMT	MFHT	CFMT % of Limit	MFHT % of Limit
Total waste volume, ft ³ (§7.B)	8,100,000	2,885 ⁽²⁾	2,396 ⁽²⁾	0.036	0.030
Total activity, curies (§7.B)	5,500,000	96.5 ⁽²⁾	103 ⁽²⁾	0.002	0.002
Total C-14, curies (§5.D)	180	NA	0.0004 ⁽²⁾	NA	0.0002
Total Tc-99, curies (§5.D)	35	0.0042 ⁽²⁾	0.0008 ⁽²⁾	0.012	0.0023
Total I-129, curies (§5.D)	0.15	NA	NA	NA	NA

LEGEND: NA = not available

NOTES: (1) From the WCS license (TCEQ 2012) 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 characterization report (WMG 2011).

Table 5-2 shows that the volume of each vessel 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 packages are small fractions of the WCS limits.

If DOE were to elect to dispose of the subject vessels at the WCS Federal Facility Waste Disposal Facility, DOE would confirm that the waste packages meet 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 Form⁵⁷.

⁵⁷ Because the WCS facility is licensed to accept Class C LLW, DOE would expect that the vessel waste packages would be approved for disposal. However, because this waste stream 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 vessel waste packages.

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 concentrator feed makeup tank and the melter feed hold tank waste packages are 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 concentrator feed makeup tank and the melter feed hold tank waste packages are 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 concentrator feed makeup tank and the melter feed hold tank waste packages are in a solid physical form.
- The radioactivity in the concentrator feed makeup tank and the melter feed hold tank waste packages do not exceed Class C concentration limits.
- The concentrator feed makeup tank and the melter feed hold tank waste packages 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 concentrator feed makeup tank and the melter feed hold tank have been packaged in custom-built, steel-shielded IP-2 shipping containers and voids in the vessels and the space between the vessels and the inside of the containers have been filled with low-density cellular concrete. Hence, the vessel waste packages are in a solid physical form. (The vessels themselves are already in a solid physical form, as noted previously.)

Because the subject vessels contain 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 draft 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 draft 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 subject vessels before packaging was as described in Section 2.5.3. Table 6-1 shows the results of the waste classification calculations, which show that the vessel waste packages do not exceed Class C limits.

Table 6-1. Vessel Waste Classification Results With Respect to Class C limits⁽¹⁾

Waste Package	Percent of Class C Limit	
	Table 1	Table 2
Concentrator Feed Makeup Tank	4.515	1.960
Melter Feed Hold Tank	6.190	1.631

NOTE: (1) From WMG 2011, as calculated by the RADMAN code.

The calculations were performed using the weight and size of the vessels themselves; neither the grout nor the shipping container was considered for conservatism (WMG 2011) even though the mass of the grout – which was necessary for stabilization purposes and to encapsulate surface

contamination – could have been considered in accordance with applicable concentration averaging guidance (NRC 1995). The uncertainties associated with the data used to estimate the residual radioactivity associated with the two vessels and to calculate the sums of fractions were relatively small, as discussed in Section 2, so it is clear that the vessel waste packages do not exceed Class C concentration limits.

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 vessel waste packages 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 (DOE 2012). This Waste Profile (CHBWV 2011a) has been approved by the Nevada National Security Site (DOE 2011) if a final decision is made to send the vessel waste packages to that facility for disposal.

As noted previously, DOE may elect to send the vessels to the commercially-operated WCS Federal Facility Waste Disposal Facility in Texas. As demonstrated earlier in this draft evaluation, vessel waste packages are in a solid physical form (with voids in the vessel and the space between the vessel 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.⁵⁸ In addition, the State of Texas Class C concentration limits mirror the concentration limits in 10 CFR 61.55; consequently, disposal of the vessel waste packages 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 vessel waste packages at the WCS Federal Facility Waste Disposal Facility would meet the third criterion of DOE Manual 435.1-1, Section II.B.2(a).

⁵⁸ 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 is consulting with NRC concerning this draft evaluation and is making it available for public review and comment, including comment by the States of Nevada and Texas where the subject vessels might be disposed of as LLW.

DOE will consider NRC comments as well as comments from the public, including the States of Nevada and Texas, before finalizing the evaluation and before making any final determination as to whether the concentrator feed makeup tank and the melter feed hold tank meet the criteria in DOE Manual 435.1-1 for waste incidental to reprocessing, and thus are not HLW and are to be managed and disposed of as LLW pursuant to DOE's regulatory authority under the Atomic Energy Act of 1954, as amended.

8.0 CONCLUSIONS

Based on information provided in the preceding sections of this draft evaluation, DOE has reached the preliminary conclusion that the concentrator feed makeup tank and the melter feed hold tank are 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.

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

10 CFR 61.55, *Waste Classification*.

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

10 CFR Part 835, *Occupational Radiation Protection*.

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

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APPENDIX A

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/m²/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.”⁵⁹

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 Part 20, *Standards for Protection Against Radiation*. Because NRC has not revised 10 CFR 61.41 to reflect the more current methodology in 10 CFR Part 20, DOE’s requirements and those in 10 CFR Part 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 Part 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.”

⁵⁹ 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.⁶⁰ 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

⁶⁰ 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 vessel waste packages 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 packages would have to meet any associated waste acceptance criteria. Because the activity density (curies per cubic foot) in the vessels waste packages is much less than the average activity density used in the updated WCS performance assessment as discussed in Section 5.2.3 of the body of this draft evaluation, it is likely that disposal of the vessel waste packages would have a negligible impact, if any, on compliance with the 25 mrem per year intruder dose limit.

61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.”

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 Part 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 Part 20, *Standards for Protection Against Radiation*, which contains similar radiological protection standards for workers and the public.

Appendix B 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 B.

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:

“Disposal Facility Closure Plans (DOE Manual 435.1, Section IV.Q(1)(a) and (b)). 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].”

“Disposal Facility Closure (DOE Manual 435.1, Section IV.Q(2)(c)). 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 Part 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](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 Fall, 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, Vol. 67, 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 B

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 concentrator feed makeup tank and melter feed hold tank waste packages for shipment at the WVDP and handling of those waste packages when they are received at either the Nevada National Security Site or the WCS LLW disposal facility in Texas for disposal, assuming that the waste packages 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 A 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 draft 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), 20.1201(a)(1)(ii), 20.1201(a)(2)(i), 20.1201(a)(2)(ii), 20.1201(e), 20.1208(a), 20.1301(a)(1), 20.1301(a)(2), and 20.1301(b).⁶¹ These NRC dose limits correspond to the DOE dose limits in 10 CFR Part 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 draft evaluation, operations related to disposal of the subject vessels 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 Dose Standard Comparison

Table B-1 provides a crosswalk of the NRC, DOE, and State of Texas dose standards. This table shows that the dose standards applicable to DOE for individual workers and members of the public are comparable to those of NRC and Texas. The following information is provided to help explain information included in the table.

2.1 Air Emissions Limit for Individual Member of the Public

The DOE is subject to and complies with the U.S. Environmental Protection Agency requirement in 40 CFR 61.92.⁶² As can be seen in the table, the annual limit of 10 mrem (0.1 mSv) is the same for each agency.

⁶¹ The “standards for radiation protection” in 10 CFR Part 20 (as cross-referenced in the performance objective in 10 CFR 61.43), which are relevant to this draft 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 draft 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 Part 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 subject vessels. Therefore, this limit is not discussed further in this draft 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 draft evaluation.

⁶² 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.”

Table B-1: Dose Standard Crosswalk⁽¹⁾

Topic	DOE	NRC	Texas
Annual air emission limit for individual member of the public	10 mrem (0.1 Sv) 40 CFR 61.42	10 mrem (0.1 Sv) 10 CFR 20.1101(d)	10 mrem (0.1 Sv) §336.304
Annual total effective dose equivalent for adult workers	5 rem (0.05 Sv) 10 CFR 835.202(a)(1)	5 rem (0.05 Sv) 10 CFR 20.1201(a)	5 rem (0.05 Sv) §336.305
Any individual organ or tissue annual dose limit for adult workers	50 rem (0.5 Sv) 10 CFR 835.202(a)(2)	50 rem (0.5 Sv) 10 CFR 20.1201(a)	50 rem (0.5 Sv) §336.305
Annual dose limit to the lens of the eye for adult workers	15 rem (0.15 Sv) 10 CFR 835.202(a)(3)	15 rem (0.15 Sv) §20.1201(a)	15 rem (0.15 Sv) §336.305
Annual dose limit to the skin of the whole body and to the skin of the extremities for adult workers	50 rem (0.5 Sv) 10 CFR 835.202(a)(4)	50 rem (0.5 Sv) 10 CFR 20.1201(a)	50 rem (0.5 Sv) §336.305
Limit on soluble uranium intake	2.4 mg/week 29 CFR 1910, Subpart Z	10 mg/week 10 CFR 20.1201(e)	10 mg/week §336.305
Dose equivalent to embryo/fetus	0.5 rem (5 mSv) 10 CFR 835.206(a)	0.5 rem (5 mSv) 10 CFR 20.1208(a)	0.5 rem (5 mSv) §336.312
Dose limit for individual members of the public (total annual dose)	100 mrem (1 mSv) DOE Order 458.1	100 mrem (1 mSv) 10 CFR 20.1301(a)	100 mrem (1 mSv) §336.313
Dose limit for individual members of the public (dose rates in unrestricted areas)	0.05 mrem/hr (0.0005 mSv) 10 CFR 835.602	2 mrem/hr (0.02 mSv) 10 CFR 20.1301(a)	2 mrem/hr (0.02 mSv) §336.313
Dose limits for members of the public with access to controlled areas	0.1 rem (0.001 Sv) 10 CFR 835.208	0.1 rem (0.001 Sv) 10 CFR 20.1301(b)	0.1 rem (0.001 Sv) §336.313
As low as reasonably achievable requirements	10 CFR 835.2, 10 CFR 835.101	10 CFR 20.1003	§336.2

NOTES: (1) Requirements from DOE's 10 CFR Part 835, NRC's 10 CFR Part 20, and State of Texas Administrative Code, Title 30, Part 1, Chapter 336, Subchapter D, along with the cited DOE Order. Dose limit differences appear in boldface.

2.2 Total Effective Dose Equivalent Limit for Adult Workers

As can be seen in the table, the annual limit of 5 rem (0.05 Sv) is the same for each agency.

2.3 Any Individual Organ or Tissue Dose Limit for Adult Workers

As can be seen in the table, the annual limit of 50 rem (0.5 Sv) is the same for each agency.

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

As can be seen in the table, the annual limit of 15 rem (0.15 mSv) is the same for each agency.

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

As can be seen in the table, the annual limit of 50 rem (0.5 Sv) is the same for each agency.

2.6 Limit on Soluble Uranium Intake

DOE uses the Occupational Safety and Health Administration permissible exposure limit for soluble uranium at 29 CFR 1910, Subpart Z (0.05 mg/m³). This limit equates to a soluble uranium intake of 2.4 mg/week as shown in the table, which is lower than the limit specified by the other two agencies.

2.7 Dose Equivalent to an Embryo/Fetus

As can be seen in the table, the dose limit of 0.5 rem (5 mSv) is the same for each agency. 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 dosimetry⁶³ is provided and used to track exposure carefully.

2.8 Dose Limits for Individual Members of the Public (Total Annual Dose)

As can be seen in the table, the annual limit of 100 mrem (1 mSv) is the same for each agency.

2.9 Dose Limits for Individual Members of the Public (Dose Rate in Unrestricted Areas)

DOE's regulation in 10 CFR 835.602 establishes the expectation that the total effective dose equivalent to individuals who enter controlled areas, without entering radiological areas or radioactive material control 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 (0.05 mSv 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.

2.10 Dose Limits for Individual Members of the Public With Access to Controlled Areas⁶⁴

As can be seen in the table, the annual limit of 0.1 rem (0.001 Sv) is the same for each agency. 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

2.11 As Low As Reasonably Achievable Requirements

As can be seen in the table, each agency has ALARA requirements.

⁶³ 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.

⁶⁴ 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.

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.” The DOE regulation at 10 CFR 835.101 requires a documented radiation protection program approved by DOE, which shall include formal plans and measures for applying the ALARA process to occupational exposure.

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.”

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.”

Conclusion

The DOE, NRC, and State of Texas definitions of ALARA are comparable.

3.0 References

Code of Federal Regulations

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

10 CFR Part 835, *Occupational Radiation Protection*.

29 CFR 1910, Subpart Z, *Occupational Safety and Health Standards, Toxic and Hazardous Substances*.

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

DOE Orders and Policies

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.

State Regulations

Texas Administrative Code, Title 30, Part 1, Chapter 336, *Radiation Substance Rules*.

APPENDIX C
**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 draft evaluation.

Appendix Content

This appendix describes the subject criteria in relation to the Department's plans for disposal of the concentrator feed makeup tank and melter feed hold tank.

Key Points

- Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 does not apply to the two vessels.
- However, disposal of the vessel waste packages 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 concentrator feed makeup tank and melter feed hold tank waste packages 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.⁶⁵

⁶⁵ 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 subject vessels,⁶⁶ 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 vessel waste packages 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 Subject Vessels Require 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 waste stream in a facility that is not a deep geologic repository will meet these

⁶⁶ 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)."

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.”⁶⁷

That is not the case here. As demonstrated in Section 4 of this draft evaluation, key radionuclides have been removed from the subject vessels to the maximum extent technically and economically practical. Moreover, the vessel waste packages 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 subject vessels 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 vessel waste packages 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 subject vessels do 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 vessels 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.”⁶⁸

Section 4, Table 4-3, of this draft evaluation identifies key radionuclides for the subject vessels. 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 draft evaluation describes how key radionuclides in the vessels 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).

⁶⁷ *Basis for Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center Tank Farm Facility* (DOE 2006).

⁶⁸ 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).

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.⁶⁹ 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 draft evaluation demonstrates that the vessel waste packages do 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 vessels have been packaged in shielded shipping containers which have been filled with low-density cellular concrete, and thus are in a solid physical form as discussed in Section 6. Section 4 of this draft evaluation further shows that management and disposal of the waste packages will meet safety requirements comparable to NRC performance objectives in 10 CFR Part 61, Subpart C. Given these considerations, management and disposal of the two vessels 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 is consulting with NRC concerning this draft evaluation, as well as making this draft evaluation available for public review and comment. DOE will consider NRC comments, as well as comments from the public, before finalizing the evaluation and before making any final determination as to whether the concentrator feed makeup tank and melter feed hold tank are or are not HLW. Accordingly, such consultation is 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*.

DOE Manuals

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

⁶⁹ 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.

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](http://info.sos.state.tx.us/pls/pub/readtac$ext.ViewTAC?tac_view=4&ti=30&pt=1&ch=336))

Other References

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.