Waste-Incidental-to-Reprocessing Evaluation for the West Valley Demonstration Project Vitrification Melter - 12167

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ABSTRACT

The Department of Energy (DOE) has determined that the vitrification melter used in the West Valley Demonstration Project can be disposed of as low-level waste (LLW) after completion of a waste-incidental-to-reprocessing evaluation performed in accordance with the evaluation process of DOE Manual 435.1-1, *Radioactive Waste Management Manual*. The vitrification melter – which consists of a ceramic lined, electrically heated box structure – was operated for more than 5 years melting and fusing high-level waste (HLW) slurry and glass formers and pouring the molten glass into 275 stainless steel canisters. Prior to shutdown, the melter was decontaminated by processing low-activity decontamination flush solutions and by extracting molten glass from the melter cavity. Because it could not be completely emptied, residual radioactivity conservatively estimated at approximately 170 TBq (4,600 Ci) remained in the vitrification melter. To establish whether the melter was incidental to reprocessing, DOE prepared an evaluation to demonstrate that the vitrification melter: (1) had been processed to remove key radionuclides to the maximum extent technically and economically practical; (2) would be managed to meet safety requirements comparable to the performance objectives for LLW established by the Nuclear Regulatory Commission (NRC); and (3) would be managed by DOE in accordance with DOE's requirements for LLW after it had been incorporated in a solid physical form with radionuclide concentrations that do not exceed the NRC concentration limits for Class C LLW. DOE consulted with the NRC on the draft evaluation and gave other stakeholders an opportunity to submit comments before the determination was made. The NRC submitted a request for additional information in connection with staff review of the draft evaluation; DOE provided the additional information and made improvements to the evaluation, which was issued in January 2012. DOE considered the NRC Technical Evaluation Report as well as comments received from other stakeholders prior to making its determination that the vitrification melter is not HLW, does not require permanent isolation in a geologic repository, and can be disposed of as LLW.

INTRODUCTION

The West Valley Demonstration Project is located some 48 km (30 mi) south of Buffalo, New York on the site of the only commercial spent nuclear fuel reprocessing facility to operate in the United States. This site is owned by the New York State Energy Research and Development Authority. Here Nuclear Fuel Services, Inc. reprocessed irradiated nuclear fuel to recover uranium and plutonium from 1966 through 1972, producing approximately 2.3 million L (600,000 gal) of reprocessing wastes that were stored in two underground tanks. This waste contained approximately 1,150,000 TBq (31 million Ci) of radioactivity – mostly Cs-137 and Sr-90 and their progeny. [1] (Unless otherwise specified, information that follows is from reference 1.)

Plant Processes

The West Valley plant used the PUREX (plutonium-uranium extraction) process to chemically separate uranium and plutonium from fission products and unwanted actinides. The similar THOREX (thorium-uranium extraction) process was used for one fuel lot. The nuclear fuel that was reprocessed came from commercial power reactors supplemented with a quantity of nuclear fuel from the N-Reactor at DOE's Hanford site.

The liquid waste from reprocessing was transferred to the tank farm, which included four underground waste tanks: Tanks 8D-1 and 8D-2, each of of 750,000-gal capacity, and Tanks 8D-3 and 8D-4, each of 15,000-gal capacity. Tanks 8D-2 and 8D-4 were used to collect liquid waste from reprocessing, with the other two tanks serving as spares. Tank 8D-1 was later used by the West Valley Demonstration Project as a waste treatment tank.

The West Valley Demonstration Project Act

In 1980, Federal legislation to clean up the West Valley site was enacted in the form of the West Valley Demonstration Project Act [2]. The Act, among other things, required the Secretary of Energy to solidify the HLW from reprocessing for permanent disposal at an appropriate Federal repository and to dispose of the LLW and transuranic waste produced by the solidification of the HLW. It also provided for the State of New York to make the facilities and the HLW available to DOE for accomplishment of the West Valley Demonstration Project.

Following passage of the Act, DOE began to move forward with the project. New facilities for treatment and solidification of HLW were built, including a vitrification facility. The HLW treatment and solidification program spanned approximately 20 years. Two hundred seventy-five stainless steel canisters of vitrified HLW were produced, which remain temporarily stored at the site.

High-Level Waste Pretreatment

After detailed analysis, DOE elected to isolate and manage the high-activity waste separately from the low-activity portion. The reprocessing wastes were therefore pretreated before the HLW was vitrified. The pretreatment program consisted of: (1) supernatant processing, (2) sludge washing, and (3) zeolite transfer to underground waste Tank 8D-2. DOE consulted with NRC on the treatment processes, consistent with provisions of the DOE/NRC Memorandum of Understanding on the project.

Approximately 2.3 million L (600,000 gallons) of supernatant was pretreated. 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.

The PUREX sludge in Tank 8D-2 was washed by 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. A second wash of the PUREX sludge was performed to further reduce the amount of sulfates in the high activity waste prior to vitrification.

Following the completion of sludge washing, final preparations were made to complete the installation of the HLW transfer system that linked all three underground waste storage tanks which contained HLW (Tanks 8D-1, 8D-2, and 8D-4) to the Vitrification Facility. To facilitate waste removal, waste transfer pumps were installed in Tanks 8D-1, 8D-2, and 8D-4.

Vitrification of the High-Level Waste

The Vitrification Facility was 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 1 provides a simplified diagram of the vitrification system.

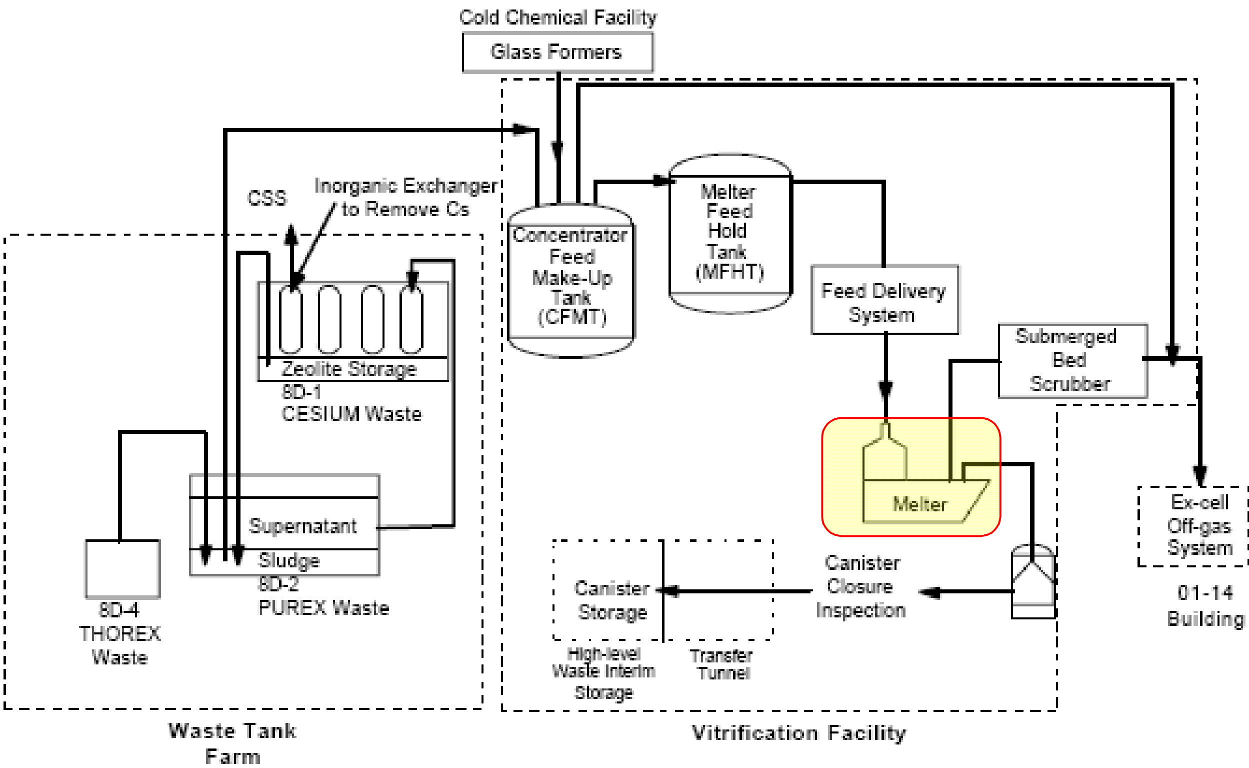

Fig 1. Simplified diagram of the vitrification system

As can be seen in Figure 1, the waste was consolidated in Tank 8D-2. From there it was pumped to the concentrator feed makeup tank in 66 separate batches numbered 10 through 75. Inside the concentrator feed makeup tank – a cylindrical vessel with an agitation system – the waste was combined with the heal from the previous batch and vitrification process recycle streams and mixed with glass formers to achieve the required waste form composition. Waste batches were transferred to the melter feed hold tank, which also had an agitation system to maintain homogeneity, and slowly fed to the vitrification melter.

Inside the melter, the waste slurry was heated by joule heating using electric heaters to an operating temperature of 1,050 to 1,150°C (1,900 to 2,100°F). The water in the slurry evaporated and the remaining solids calcified. The calcined waste and glass formers melted in a pool of homogeneous molten glass that was mixed by natural circulation. At regular intervals, glass was poured from the melter discharge port into a HLW canister positioned in a turntable fixture beneath the pour spout. A technique call airlifting, in which pressurized air was introduced to push molten glass through the pour spout, was used to fill the canisters, with typically about 15 airlifts required over a period of approximately 63 hours to fill one canister. Figure 2 shows the general arrangement.

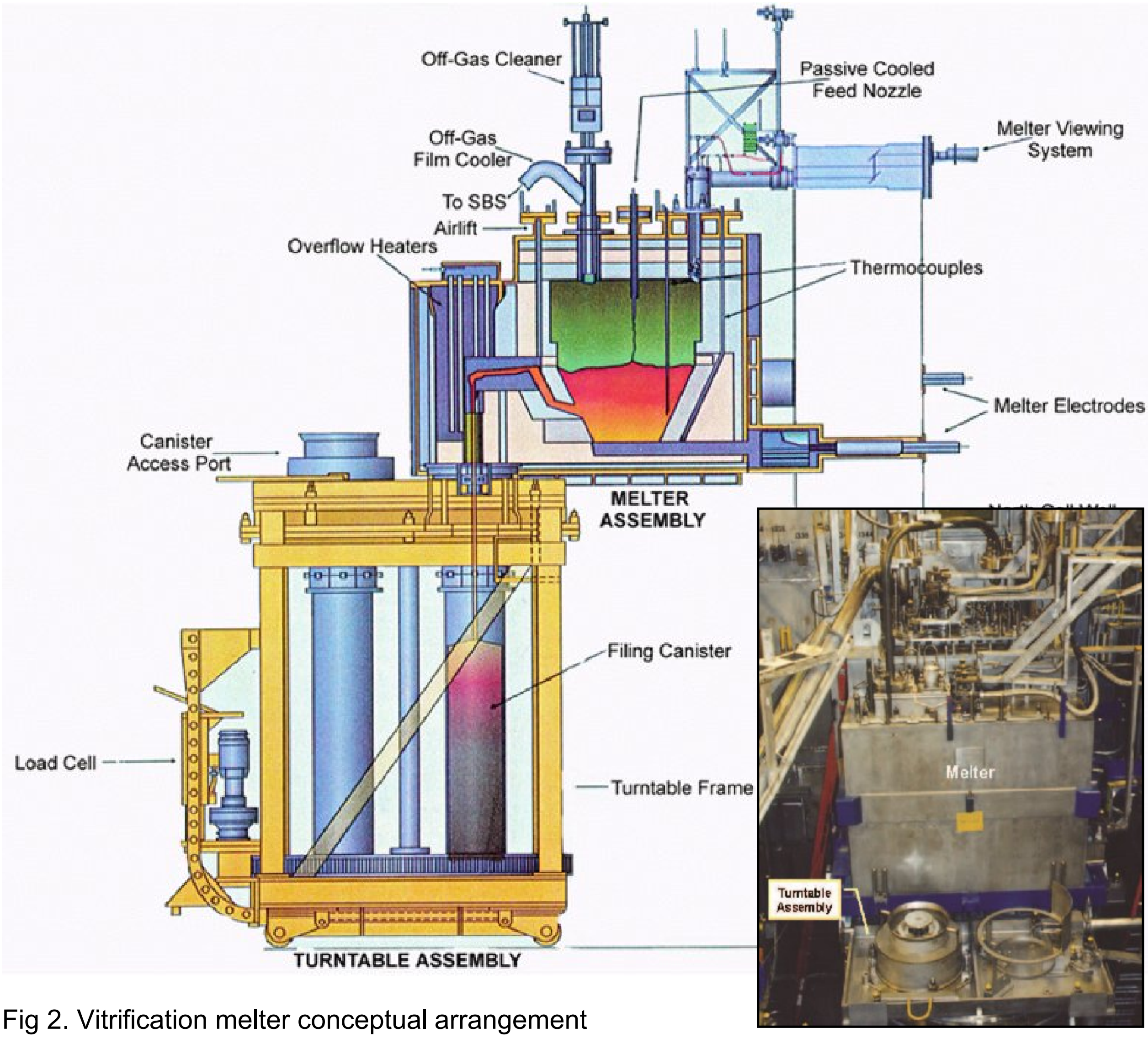

Fig 2. Vitrification melter conceptual arrangement

Melter Design

The vitrification melter was constructed of Inconel, lined with refractory brick, and encased in a stainless steel cooling jacket. The outside resembles a box approximately 3.1 m (10 ft) on each side. The unit weighs approximately 48,000 kg (106,000 lb). The melter cavity, as indicated in Figure 2, is shaped like an inverted, truncated rectangular pyramid surmounted by a rectangular tank. The melter cavity held approximately 860 L (227 gal) of molten glass when filled to its normal working level of 66 cm (26 in). The slurry was heated by three electrodes, one of which served as the cavity bottom as shown in Figure 2.

The melter was built with two separate glass discharge chambers, each equipped with removable heaters, because of the potential for buildup of solidified glass in the discharge area. This design feature proved to be invaluable when the discharge chamber being used became plugged with hardened glass near the end of operations. The second discharge chamber was placed into service and used to complete vitrification.

Melter Decontamination

Vitrification operations began in in July 1996 and concluded in August 2002. Given the complexities of the process, DOE commissioned a Vitrification Completion Team composed of representatives from DOE, the New York State Energy Research and Development Authority, West Valley Nuclear Services (DOE's site contractor), and NRC to review issues surrounding the ability to complete vitrification operations. This team developed an approach for retrieving waste from the underground waste tanks, washing and characterizing the residual tank materials, and flushing the vitrification system prior to completing a controlled shutdown of the melter.

The vitrification system flushes involved flushing Tank 8D-1, Tank 8D-2, the waste header, Tank 8D-4, and other system equipment with nitric acid solution and water and flushing the concentrator feed makeup tank and the melter feed hold tank with high-pressure water spray. For the vitrification melter, this process involved feeding material with lower than normal radioactivity concentrations (the decontamination solutions from the flushes) so it could be processed to reduce the radionuclide inventory in the molten glass pool.

The last batch of HLW slurry feed material processed in the melter was batch 75. The total volume of decontamination solutions used in the vitrification system flushes was approximately 833,000 L (220,000 gal). This material, after evaporation, made up two additional batches of feed material designated batch 76 and batch 77. Eight HLW canisters were produced from this material. The second step in decontamination of the melter involved extracting as much molten glass (the Batch 76 and Batch 77 material in the melter cavity) as practical using two evacuated canisters. These canisters were about the same size as those used for HLW but equipped with a special L-shaped "snorkel" assembly and placed under vacuum.

As shown in Figure 3, the canister was positioned over the melter and the snorkel was inserted into the molten glass pool in the melter cavity where an aluminum plug in the bottom of the snorkel melted allowing the molten glass to be drawn into the canister. The snorkel for the first evacuated canister reached within 15 cm (6 in) of the cavity bottom and the snorkel for the second evacuated canister within 5 cm (2 in) of the bottom.

Fig 3. Evacuated canister removing molten glass from melter cavity

Two evacuated canisters had been provided with one intended to serve as a backup. Both were actually used and together were able to remove approximately 88 percent of the molten glass in the melter cavity. The residual glass heel in the cavity after use of the evaluated canisters was determined to be less than 20 cm (8 in).

Following use of the evacuated canisters, the melter was shut down. The electrodes were de-energized and the remaining glass heel hardened.

Melter Characterization

The melter was characterized for residual radioactivity considering three separate areas: the melter cavity, the plugged discharge port, and the melter exterior. Characterization of the first two areas involved using measured dose rates to determine the amount of Cs-137 present and scaling factors based on sample analytical data to estimate the amounts of other radionuclides. The amount of residual radioactivity on the outside surfaces was estimated using the maximum measured surface activity.

Dose rates inside the melter were measured by lowering an unshielded Ludlum Model 133-7 Geiger-Mueller detector through nozzles located above the hardened glass. Approximately 90 Sv/hr (750 R/hr) was measured directly above the residual glass in the melter cavity and approximately 4.8 Sv/hr (40 R/hr) near the plugged discharge cavity. The analytical data used came from laboratory analysis of glass shard samples from the two evacuated canisters.

Two geometry models were used to calculate dose conversion factors, which were combined with the measured dose rates to estimate the Cs-137 activity. The model for the cavity was prepared using the QAD-CGGP-A computer code, a point-kernel code for calculating fast neutron and gamma ray penetration through various shield configurations developed by Atomic Energy of Canada, Ltd. The model for the plugged discharge port was developed using Megashield™, a Windows-based code from WMG, Inc. used to perform point-kernel shielding calculations.

Table I shows the resulting estimates, which are as of October 2004. These estimates total approximately 170 TBq (4570 Ci) and are conservative compared to a later estimate as discussed below.

Table I. Vitrification Melter Residual Activity Estimate

Nuclide	Activity (GBq)	Activity (Ci)	Nuclide	Activity (GBq)	Activity (Ci)
C-14	7.84E-01	2.12E-02	U-235	1.39E-02	3.76E-04
K-40	3.03E+00	8.19E-02	U-238	8.32E-02	2.25E-03
Mn-54	3.17E+00	8.57E-02	Np-237	2.29E-01	6.20E-03
Co-60	3.08E+00	8.33E-02	Pu-238	2.53E+01	6.84E-01
Sr-90	9.14E+03	2.47E+02	Pu-239	5.88E+00	1.59E-01
Zr-95	6.11E+01	1.65E+00	Pu-241	1.15E+02	3.12E+00
Tc-99	4.10E-01	1.11E-02	Pu-242	4.14E-04	1.12E-05
I-129	[a]	[a]	Am-241	1.11E+02	3.00E+00
Cs-137	1.59E+05	4.31E+03	Am-242m	3.38E-03	9.16E-05

Table I. Vitrification Melter Residual Activity Estimate

Nuclide	Activity (GBq)	Activity (Ci)	Nuclide	Activity (GBq)	Activity (Ci)
Eu-154	4.48E+01	1.21E+00	Am-243	1.30E+00	3.50E-02
Th-228	1.51E+00	4.09E-02	Cm-242	2.71E+00	7.33E-02
Th-229	[a]	[a]	Cm-243	6.22E-01	1.68E-02
Th-230	1.35E-02	3.65E-04	Cm-244	7.88E+00	2.13E-01
Th-232	1.48E-02	4.01E-04	Cm-245	5.74E-01	1.55E-02
U-232	1.85E-02	5.01E-04	Cm-246	6.54E-02	1.77E-03
U-234	3.62E-01	9.81E-03			

a. The amounts of these radionuclides are insignificant based on sample analytical data.

DEVELOPMENT OF THE DRAFT EVALUATION

DOE Manual 435.1, *Radioactive Waste Management Manual* [3], provides two processes that may be used to determine that wastes resulting from reprocessing are not HLW, the citation process and the evaluation process. The citation process is not intended for complex equipment such as vitrification melters so the evaluation process was appropriate for this case.[1]

The evaluation [4] was therefore developed to address the evaluation criteria of DOE Manual 435.1. These criteria, as described in Section II.B(2)(a) of the manual, can be used to demonstrate that wastes resulting from spent nuclear fuel reprocessing are not HLW and should be managed as LLW are as follows:

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

(2) Criterion 2 – the wastes will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, *Performance Objectives*; and

(3) Criterion 3 – the wastes are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of DOE Manual 435.1-1[2], provided the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize.

Before making a determination as to whether the vitrification melter would meet these criteria, DOE consulted with NRC and made the draft evaluation [4] available for public and state comment. The draft evaluation specified that DOE planned to dispose of the melter waste

[1] There are two other sets of waste-incidental-to-reprocessing criteria that may apply to DOE sites. One – criteria established by NRC under the West Valley Demonstration Project Act for the West Valley decommissioning [5] – applies only to West Valley, but does not apply to waste transported offsite for disposal. The other – in Section 3116(a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 [6] – does not apply to the West Valley site.

[2] Chapter IV describes requirements for management of DOE low-level waste.

package at either the Nevada National Security Site Area 5 Radioactive Waste Management Site or the Waste Control Specialists LLW disposal facility in Texas.

Content of the evaluation

The draft evaluation was organized in the following manner:

Section 1 was an introduction that addressed matters such as the purpose, scope, and background.

Section 2 described the waste stored in Tanks 8D-2 and 8D-4 at the conclusion of reprocessing and the HLW pretreatment process, along with the vitrification melter and its characterization.

Section 3 described the DOE Manual 435.1-1 waste-incidental-to-reprocessing criteria.

Section 4 described how key radionuclides had been removed from the vitrification melter to the maximum extent technically and economically practical.

Section 5 discussed how safety requirements comparable to NRC performance objectives in 10 CFR 61, Subpart C will be achieved, along with waste acceptance criteria for the potential disposal sites.

Section 6 showed that the radionuclide concentrations in the packaged vitrification melter are less than Class C concentration limits and explained that the vitrification melter will be managed in accordance with Chapter IV of DOE Manual 435.1-1.

Section 7 described planned consultation with the NRC and the opportunity for public and state comment.

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

Section 9 identified the references cited in the evaluation.

Appendix A provided copies of drawings for the vitrification melter and its shipping container.

Appendix B described management controls used to ensure quality in the draft evaluation.

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

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

Appendix E compared the criteria of Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 to the DOE Manual 435.1-1 waste-incidental-to-reprocessing criteria for information purposes.

DOE provided the draft evaluation to NRC for review and made it available for public and state comment in March 2011.

Addressing the First Criterion on Key Radionuclide Removal

Section 4 of the draft evaluation addressed the first evaluation criterion by identifying key radionuclides in the melter and then describing how these radionuclides had been removed to the maximum extent technically and economically practical. The key radionuclides were determined to be those long-lived and short-lived radionuclides listed in Tables 1 and 2 of the NRC's regulations in 10 CFR 61.55, three of which are important to the performance assessment of the Waste Control Specialists LLW disposal facility, along with four other radionuclides that are important to the results of the performance assessment of the Nevada National Security Site LLW disposal facility. These radionuclides were as follows:

H-3	Sr-90	Th-229*	Pu-238	Am-241
C-14**	Nb-94	U-233*	Pu-239*	Am-243
Co-60	Tc-99*,**	U-234*	Pu-240	Cm-243
Ni-59	I-129**	U-238*	Pu-241	Cm-243
Ni-63	Cs-137	Np-237	Pu-242	Cm-244

*Important to the Nevada National Security Site performance assessment.
**Important to the Waste Control Specialists performance assessment.

Evaluation of representative potential methods of removing key radionuclides showed that processing of decontamination solutions in the vitrification melter, using the evacuated canister system, and dismantling the melter were the only methods technically practical. Processing of decontamination solutions and the evacuated canister system were used and proved to be effective in removing key radionuclides. (Note that these bulk removal technologies did not selectively remove certain 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 based on a detailed cost-benefit analysis performed in 2004. This assessment also evaluated vitrification melter dismantlement and concluded that this approach also would not have been economically practical because of increased expense and worker radiation dose and other factors. The economic practicality assessment compared the impacts of additional flushing and vitrification melter dismantlement expressed as estimated monetary costs with the potential benefits expressed in terms of the monetary value of the reduction in collective radiation dose. (The economic practicality analysis was later revised as discussed below.)

Section 4 of the draft evaluation therefore demonstrated that further efforts to remove key radionuclides would have increased worker radiation dose without resulting benefits and that disposal of the vitrification melter without additional decontamination would have an insignificant impact on the performance of the disposal site in protecting the health and safety of workers and the public.

Addressing the Second Criterion on Meeting Safety Requirements

Section 5 of the draft evaluation began by summarizing key DOE safety requirements related to disposal of LLW. It then compared each of these requirements with the similar safety requirements of the NRC and the State of Texas and showed DOE's safety requirements to be comparable.

Section 5 also provided a summary of the results of the most recent performance assessment for the Nevada National Security Site Area 5 facility related to protection of the general population and inadvertent intruders. The results of a special performance assessment that was performed to evaluate the potential impact of disposal of the vitrification melter in that facility were also described – this assessment showed that these impacts would be negligible.

The results of a performance assessment of the Waste Control Specialists Federal Facility Waste Disposal Facility related to protection of the general population were also discussed. Appendix C was included to provide a more detailed comparison of the comparability of DOE, NRC, and State of Texas requirements for LLW disposal. Appendix D was included to provide a more detailed discussion of the comparability of DOE, NRC, and State of Texas radiation dose standards that apply to LLW disposal.

Section 5 also discussed DOE waste acceptance criteria including the criteria for the Nevada National Security Site. It described how it had been determined that the vitrification melter waste package meets the Nevada National Security Site waste acceptance criteria (West Valley had submitted a waste profile for this waste steam to the Nevada National Security Site and this waste profile had been approved for disposal).

In addition, Section 5 discussed waste acceptance criteria for the WCS Federal Facility Waste Disposal Facility and how it would be established that the vitrification melter waste package meets these criteria if the waste package were to be shipped to that facility for disposal.

Addressing the Third Criterion on Management as Low-Level Waste

Section 6 demonstrated that the vitrification melter waste package will be in a solid physical form, will not exceed Class C concentration limits, and will be managed in accordance with DOE requirements as LLW. This section explained that the melter is packaged in a custom-built Industrial Package Type 2 shielded container made of steel that is 14.2 cm (6 in) thick on the sides and 10.2 cm (4 in) thick on the top and bottom. Prior to shipment, the container will be filled with low-density cellular concrete; the grouted container will weigh approximately 163 metric tons (360,000 lb).

Section 6 also contained a table showing that the Class C sum of fractions was approximately 0.95 for Table 1 radionuclides and approximately 0.046 for Table 2 radionuclides, demonstrating that Class C concentration limits were not exceeded. The sum-of-fraction approach was used because the melter contains a mixture of radionuclides. The Table 1 sum-of-fraction result was driven primarily by Am-241 and Pu-238. The calculations were deliberately conservative for several reasons, including, not accounting for radioactive decay that has taken place since melter characterization in 2004. (If this decay had been taken into account, the Table 1 sum of fractions would have been approximately 0.87. Later analysis showed that the conservatisms inherent in the calculations bounded uncertainties associated with the analytical data used for the estimates.)

Consultation with NRC and Review by Other Stakeholders

DOE consulted with NRC staff on the evaluation consistent with its guidance in DOE Guide 435.1-1, *Implementation Guide for Use With DOE Manual 435.1-1* [7]. The NRC review, which was performed in accordance with NUREG-1854, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations* [8], focused on the following general topics, as they related to the criteria in DOE Manual 435.1-1:

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

After several telephone conferences involving NRC staff, DOE staff, and DOE contractors, NRC submitted a request for additional information in May 2011 [9]. This request included a total of 17 comments on the melter inventory, key radionuclides, removal of key radionuclides to the maximum extent practical, and waste classification. DOE evaluated the comments and provided written responses in June 2011 describing changes to be incorporated into the evaluation [10]. Two examples of these changes follow.

Examples of Changes to the Evaluation from Consultation With NRC

One change entailed including another estimate of residual radioactivity in the melter using a different method that involved multiplying the estimated mass of residual glass (425 kg or 937 lb) by the radionuclide concentrations measured in samples taken from the evacuated canisters. This estimate showed a total activity of approximately 83 TBq (2,240 Ci), showing the original 170 TBq (4,570 Ci) estimate to be conservative.

Another change entailed an improved analysis of the costs and benefits of processing additional decontamination flush solutions in the melter. Figure 4 illustrates the conceptual model used in this analysis.

In this model, recorded data from vitrification records were used and total activity considered. (Note that melter input is slow and continuous while melter output is periodic, a factor that complicates determining the actual melter cavity radionuclide concentrations at a particular time.) Table II shows the results of the calculations made using this model.

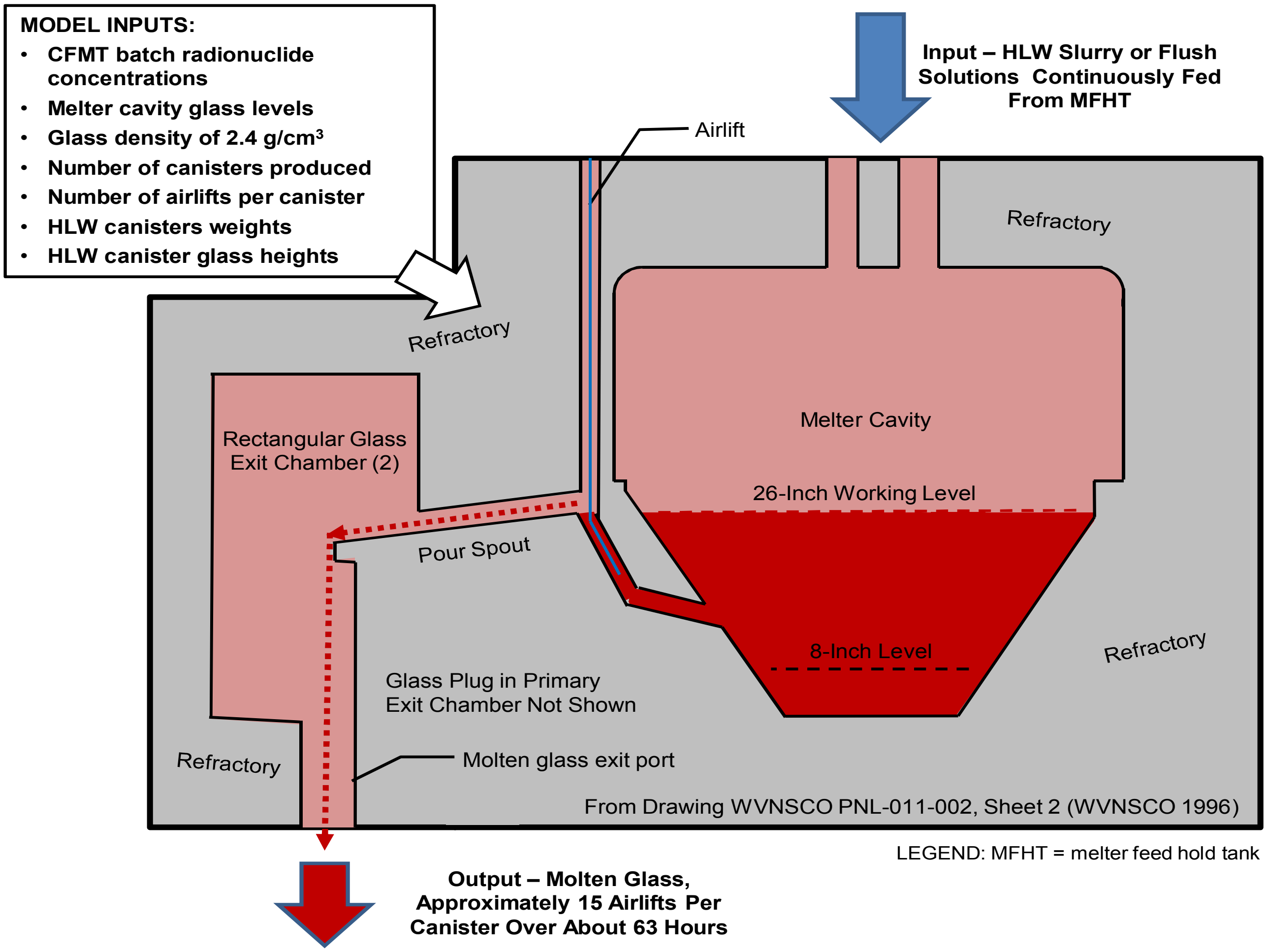

Fig 4. Conceptual model for processing additional flush solutions

Table II. Estimated Effectiveness of Processing Another Flush Solution Batch

Concentrator Feed Makeup Tank Batch and Last Canister Filled While this Batch Was Being Fed to the Melter	Glass Pool Activity	
	TBq	Ci
75 (last HLW batch, canister 267)	1,181	31,908
76 (first flush solutions batch, canister 272)	592	16,012
77 (second flush solutions batch, canister 275)	159	4,303
78 (Hypothetical flush solutions third batch, one hypothetical additional canister)	67	1,814

Figure 5 shows the results of the calculations graphically. The 30,000 Ci line in this figure equates to 1,110 TBq and the 5,000 Ci line equates to 185 TBq.

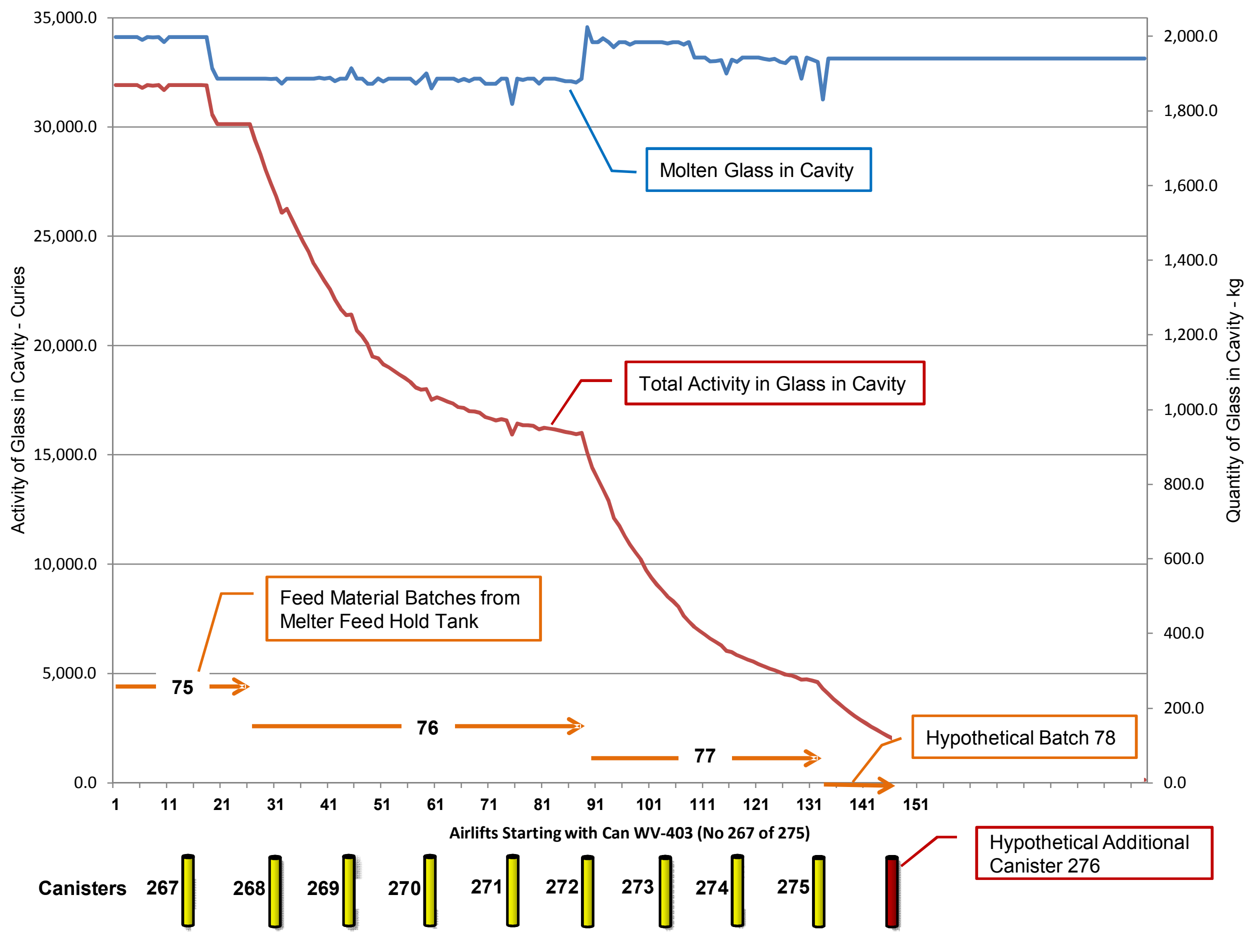

Fig 5. Predicted total activity in vitrification melter glass pool

The costs of processing one additional batch of flush solutions considering the results of the calculations based on this model were found to outweigh the benefits as with the analysis included in the draft evaluation. Only one benefit would have been realized had the hypothetical additional flush solutions been processed: the shielded container could have been designed and constructed of lighter weight steel, which would have reduced costs associated with materials, fabrication, and transportation by approximately $100,000. (There were no other benefits because of factors such as the low dose rates on the outside of waste package and the waste package already meeting the disposal site waste acceptance criteria.) A total of approximately $1 million in additional costs (in 2002 dollars) would have been involved considering that one additional flushing and processing cycle would have taken about two weeks to complete at a cost of approximately $1 million, based on vitrification system operating costs that were running $25 million to $30 million per year.

Another factor in considering costs of processing of additional flush solutions was the limited vitrification melter service life. A failure would have, for all practical purposes, stranded radionuclides within the melter since neither processing of flush solutions nor use of the evacuated canister system would have been feasible with the residual glass in the Melter cavity in solid form.

The NRC Technical Evaluation Report

NRC issued its Technical Evaluation Report [11] in September 2011. The executive summary of this report stated that:

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

Public Comments

Comments on the draft evaluation from public stakeholders fell into four categories:

- Legal basis and authority,
- The Waste Management Environmental Impact Statement,
- Concentration averaging, and
- Melter flushing solutions.

Several changes to the evaluation were made after consideration of the public comments. The DOE's written responses were posted on the West Valley Demonstration Project website and the DOE Office of Environmental Management website.

CONCLUSION

In January 2012, DOE issued the final waste-incidental-to-reprocessing evaluation [1] and made the determination that the vitrification melter was not HLW and may be disposed of as LLW at either the Nevada National Security Site Area 5 Radioactive Waste Management Site or the Waste Control Specialists radioactive waste disposal facility in Texas [12].

REFERENCES

1. "Waste-Incidental-to-Reprocessing Evaluation for the West Valley Demonstration Project Vitrification Melter," U.S. Department of Energy, West Valley, New York (January 2012).
2. West Valley Demonstration Project Act, Public Law 96-368 (S. 2443) (October 1, 1980).
3. DOE Manual 435.1-1, "Radioactive Waste Management Manual," Change 1, U.S. Department of Energy, Washington, D.C. (June 19, 2001).
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5. "Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 67, No. 22 (February 1, 2002).

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