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Rev: 1730-2

DP-82-125-4

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WASTE MANAGEMENT PROGRAM TECHNICAL PROGRESS REPORT

OCTOBER—DECEMBER, 1982

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E. I. du Pont de Nemours & Co.
Savannah River Plant and Laboratory
Aiken, SC 29808

PREPARED FOR THE U. S. DEPARTMENT OF ENERGY UNDER CONTRACT DE-AC09-76SR00001

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Printed in the United States of America

Available from

National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161

Price: Printed Copy A10; Microfiche A01

DP-82-125-4

Distribution Category: UC-70

**WASTE-MANAGEMENT PROGRAM
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DP--82-125-4

DE83 017050

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Published: July 1983

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ABSTRACT

This quarterly report provides current information on operations and development programs for the management of radioactive wastes from operation of the Savannah River Plant and offplant participants. The studies on environmental and safety assessments, in situ storage or disposal, waste form development and characterization, process and equipment development, and low-level waste management are a part of the Long-Term Waste Management Technology Program. The following studies are reported for the SR Interim Waste Operations Program: surveillance and maintenance, waste concentration, low-level effluent waste, tank replacement/waste transfer, and solid waste storage and related activities.

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FOREWORD

This quarterly report provides information on the current status of plant operations and research and development programs for the management of radioactive wastes at the Savannah River Plant and, starting this quarter, several offplant participating laboratories. These efforts are funded by the Office of Defense Waste and Byproducts Management (DWBM) of the U.S. Department of Energy.

This technical progress report consists of two general sections: Long-Term Waste Management Technology and SR Interim Waste Operations.

The section on long-term waste management technology summarizes research and development studies on the immobilization of defense high-level waste at all three U.S. sites (SRP, ICPP, and RHO). Immobilization of the waste consists of placing it in a high-integrity form with a very low potential for radionuclide release.

The basic plan for meeting the goal of immobilization of the DOE high-level waste is (1) to develop technology to support a realistic choice of waste form alternatives for each of the three (SRP, ICPP, and RHO) DOE sites, (2) to develop product and processing technology with sufficient scale-up to provide design data for full-scale facilities, and (3) to support R&D for construction and operation of immobilization facilities.

Current offplant participants in the long-term high-level waste management program are:

- Idaho Chemical Processing Plant (ICPP)
- Materials Characterization Center-PNL (MCC)
- Materials Review Board-ANL (MRB)
- Rockwell Hanford Operation (RHO)
- Sandia National Laboratories (SNL)

In the reference design process for SRP high-level waste, the salt will be dissolved, and the resulting solution will be used to slurry the sludges from the tanks where they are currently stored. The salt solution will be separated from the sludge by centrifugation. More than 99% of the ^{137}Cs and residual ^{90}Sr will be removed from the salt solution by ion exchange. The remaining liquid will be evaporated to salt cake (mostly sodium nitrate) that contains only a small fraction of the total biologically hazardous radioactive material. The ^{137}Cs and ^{90}Sr will be eluted from the ion

exchange resin and combined with the sludge (containing most of the ^{90}Sr and ^{239}Pu). The combined eluate and sludge will be incorporated into borosilicate glass and shipped to a Federal repository.

The Defense Waste Processing Facility (DWPF) is being designed to incorporate the Savannah River Plant (SRP) defense waste into borosilicate glass according to the reference design process.

The current operational status for the liquid waste treatment and storage facilities at the SRP is reported for this quarter. The research and development program in support of interim tank and storage of liquid and crystallized plant waste comprises (1) analysis of the present tank system, (2) development of improved processes and equipment, (3) corrosion studies, and (4) analysis of environmental effects.

A concurrent program is developing technology for long-term management of these wastes. Because the long-term plan includes solidification and temporary storage in a retrievable storage facility, interim treatments must permit retrieval of the waste and incorporation into a high-integrity form. The SRP waste management program requires that all sludge be removed from waste tanks before tank retirement.

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HIGHLIGHTS

Long-Term Waste Management Technology

The Preliminary Safety Analysis (PSA) for the DWPF was transmitted to DOE-SR for information and comment. The PSA was issued in October. Page 13

Comment letters were received on the Waste Form Selection Environmental Assessment that supported the selection of borosilicate glass as the DWPF waste form. Page 13

A figure-of-merit analysis showed that alternative strategy 3a was the highest ranking alternative strategy for the long-term management of high-level radioactive wastes generated at the ICPP. Page 14

The feasibility of microwave radiowave energy as a method for in situ drying of residual liquid in Hanford single-shell tanks is being evaluated. Page 16

Waste tank corrosion studies are being made to define composition and storage temperature conditions such that the average corrosion rate will not exceed 1 mil per year over a 50-year storage period. Page 16

The metallographic preparation of samples from the first in a series of six test cesium capsules has been completed at Oak Ridge National Laboratory. Page 18

The potential impact of isotope recovery upon existing and future Hanford waste management plans and costs are being evaluated on a total system basis. Page 18

Viscosity profiles between 1050 and 1200°C are being measured on glasses prepared with various mixed calcine types. Page 19

Test methods for nuclear waste forms and metallic engineered barriers are being codified. Page 20

Bid invitations for supply of 500 lb of simulated waste glass were sent to nine potential manufacturers. Page 22

The fifth experimental run of the large slurry-fed melter (LSFM) was completed after 14 days of continuous operation. The entire system performed with no major problems. Page 23

Surges in the flow of off-gas from a slurry-fed melter occur because of unsteady evaporation of liquid that collects on and possibly within the crust covering the surface of the molten glass. Page 42

The first large melting test with simulated high-aluminum waste was completed successfully. Forty-one-hundred gallons of a 40 wt % solid slurry of Frit 165 and high-aluminum waste were fed to the pilot-scale ceramic melter to produce 8.5 tons of glass during a ten-day period. Page 54

Two evacuated canister tests have qualitatively confirmed theoretical predictions and demonstrated that the evacuated canister technique could successfully drain a glass melter in the Defense Waste Processing Facility. Page 58

Argon purging appears to be useful in prolonging the life of Inconel® 600 thermocouple sheaths in a glass melter. Page 67

The Hydro-Sonic® scrubber is a viable replacement for the primary washable deep-bed filter. The fractional decontamination performance of a single-stage Hydro-Sonic® scrubber for submicron particles between 0.1 to 1.0 μm is roughly an order of magnitude better than a two-stage WDBF. Page 82

As part of the overall effort for assessing the chemical durability of SRP waste glass forms, a variety of leaching tests are being performed under anticipated repository conditions. Page 82

Recently, a physical and mathematical model of the leaching process was developed. Emphasis is now on confirming the model by more detailed studies of glass corrosion. Page 92

The viscosities and resistivities of glasses made from Frit 165 and three waste simulations have been measured. At 1150°C the viscosities range from 35 poise for high-iron waste to 180 poise for high-aluminum waste. The resistivities at the same temperature range from 2.9 ohm-cm for high-iron waste to 3.6 ohm-cm for TDS-3A waste. Page 101

Chemical Oxygen Demand, a titrametric technique of analyzing the reducing strength of melter feeds, has been applied to laboratory samples and to actual feeds for large-scale melters. The precision of the technique and agreement between measured and predicted COD values are excellent. Page 124

For the in-tank decontamination of cesium-137 and strontium-90 and soluble plutonium or sodium titanate to be successful, the precipitate must be stable during decontamination. Tests have verified that the precipitate is stable to both radiation and chemicals and that no appreciable decomposition will take place on a full cycle multibatch in-tank decontamination. Page 133

The long-term consequences of past and present SRP burial ground practices are being studied at the SRL. The current view is that some remedial action, such as emplacement of additional soil cover, will be required to place the current burial ground in a satisfactory low-risk, low-maintenance decommissioned condition for the future. Page 139

How to provide greater confinement disposal for the most significant fraction of SRP waste is being investigated. The greater confinement disposal fraction must be identified and characterized. Page 147

The revised final 10CFR61 requirements for land disposal of LLW, published on October 23, 1982, comparing 10CFR61 with current SRP practices and with the design and operation of a new greater confinement disposal facility is outlined. Page 152

The solid waste incineration facility for testing has operated for 400 hr and burned 25,000 lb of waste. The incinerator processed a wide variety of simulated wastes. The resulting ash reduced the storage volume of the waste by a factor of 20. Page 159

An extensive effort to update the CAB cost estimate for the DWPF sludge facilities was completed. The estimated project cost was determined to be 880 million dollars. Page 185

SR INTERIM WASTE OPERATIONS

Heavy rains flooded Tank 11H on September 19, 1982. A program to prevent future flooding was initiated. Page 187

Beginning in August, radioactivity up to 50 times above background has been detected intermittently in the air exhausted from the annuli of Tanks 29H and 31H. Page 187

The Equipment Engineering Department recommended the installation of two turn-key cathodic protection systems to protect underground tank farm piping in both F and H Areas. Page 188

The Equipment Engineering Department concluded that the corrosion phenomenon observed at Cleanout Port No. 3 was localized, and all defective piping was removed. Page 190

A three part program is being developed to investigate the F- and H-Area seepage basins with respect to their organic and inorganic content, influx volume and sources, evaporation and seepage rates, and the controlling seepage mechanism. Page 193

Tank 24H salt removal remains suspended until January 1983 when dissolved salt solution will be transferred to Tank 48H for the in-tank salt decontamination demonstration. Page 194

Tank 20F density driven salt removal has been completed. A total of 714,000 gal of salt (66% of the total) has been removed from Tank 20F. Page 194

An additional 126,000 gal of well water was added to Tank 24H in October to complete water addition for the second water wash. The second wash slurry started settling on September 27. Page 195

The well water addition for the third water wash was completed October 30; 540,000 gal of water was added. Sludge settling began October 30 after the slurry was sampled. Page 196

Construction of the facility to demonstrate annulus cleaning was completed. Page 197

Construction of the salt decontamination demonstration facility is about 90% complete. Page 197

The SRL saltcrete demonstration is scheduled for March 1983. Page 201

SRP has been allotted \$125,000 to be used for developing an inventory work off plan for retrievably stored contact — handled TRU waste. Page 202

The beta-gamma incinerator at SRL was disassembled. The primary and secondary chambers, spray quencher, and most of the duct work were relocated to the H-Area demonstration site in November and erected in December. Page 203

Conceptual design studies for an incinerator to volume reduce an estimated 5000 ft³/yr of noncertifiable Pu-238 contaminated waste were reviewed in October. Page 204

The alpha disassembly and decontamination facility, a FY-1986 project, is currently estimated at \$36,000,000 for construction. Because of a lack of support for this level of funding, a task force was formed to research the possibilities of constructing the facility within alternative sites. Page 204

HIGH-LEVEL WASTE TECHNOLOGY

ENVIRONMENTAL AND SAFETY ASSESSMENTS

Preparation of environmental documentation continued in support of construction and operation of the Defense Waste Processing Facility (DWPF).

The Preliminary Safety Analysis (PSA) for the DWPF was transmitted to DOE-SR for information and comment. The PSA was issued in October to cover the start of DWPF construction activities. An Executive Summary of the PSA was prepared and Chapter 11 was revised from Technical Specifications to Safety Limits and Control Requirements. The Chapter 11 change was made because of different approval requirements for Technical Specifications.

A system for revising the PSA was prepared and is being reviewed to allow an orderly preparation of the Final Safety Analysis, which is required to be approved before operation of the DWPF.

DWPF Waste Form Selection

Seven comment letters (including four no comments) were received on the Waste Form Selection EA* that supported the proposed finding of no significant impact on the selection of borosilicate glass as the DWPF waste form. The comments received support DOE's preliminary finding that the selection of borosilicate glass as the waste form for SRP wastes will not result in any significant impacts. Based on the analyses in the environmental assessment and in the DWPF EIS (DOE/EIS-0082) and after consideration of comments received during the public review period on the proposed finding, the Department determined on December 21 that the proposed action will not have a significant impact on the environment. This final determination fulfills NEPA requirements on the selection of the DWPF waste form.

Environmental Monitoring During DWPF Construction

An environmental program was developed and recommended for conduct during DWPF construction. The principal sampling and

* Environmental Assessment - Waste Form Selection for SRP High-Level Waste (DOE/EA-0179).

measurement steps indicated by the proposed program have been implemented.

Environmental and Strategy Assessment — ICPP

The "Environmental Evaluations Document," IDO-10105, was printed and released for UC-70, DOE-ID, and ENICO internal distributions.

An "Advisory Board" to the Figure-of-Merit (FOM) Committee was assembled to review criteria and definitions to be used in a FOM analysis of alternative strategies for the long-term management of ICPP high-level wastes. The Advisory Board discussed current waste management practices, philosophies, and public concerns.

A FOM analysis was performed to evaluate and determine the relative suitability of alternative strategies for the long-term management of high-level radioactive wastes generated at the Idaho Chemical Processing Plant. The results of this FOM analysis will be used in addition to regulatory and DOE policies to identify two or more preferred alternative strategies based on current knowledge. The waste forms identified for the alternative strategies are only to be considered as representative for that strategy and not as selected forms to implement the strategy.

The six alternative strategies included in this FOM analysis were: (1) leave waste in place; (2) retrieve waste, modify to pellets (2a) or glass (2b) and dispose onsite; (3) retrieve waste, modify to glass (3a) or to stabilized calcine (3b) and dispose offsite; (4) retrieve waste, separate actinide fraction, dispose actinide-depleted waste onsite and dispose actinide fraction offsite; (5) delay retrieval of waste, after delay, retrieve waste, modify (to a glass) and dispose offsite; (6) delay decision on calcined waste, modify new liquid waste and dispose offsite (combining portions of 3 and 5).

The FOM analysis was conducted by first selecting and defining criteria on which each alternative strategy must be evaluated. The criteria used in this analysis were grouped under these categories: (1) technology, (2) costs, (3) risk, (4) regulatory constraints, and (5) sociopolitical issues.

A relative weight, based on consideration of importance, was assigned to each category and criterion. Numerical scores were assigned to each alternative strategy according to how well a criterion was met. The FOM score was then calculated for each alternative strategy by multiplying the criterion weighting factor by the score which was received by that alternative strategy. The procedure was repeated for all criteria, and the numerical scores were summed to produce the figure-of-merit score.

A comparison of the FOM scores shows that the alternative strategy (3a) of retrieving the waste, modifying to a glass and disposing offsite and the alternative strategy (2b) of retrieving the waste, modifying to a glass and disposing onsite were the highest ranking strategies. These alternative strategies were awarded 756 and 716 points of a possible 1000. A second group which falls within about a 100 point spread (597 to 499) includes all other alternative strategies evaluated, except the actinide separation strategy (4) which received the lowest score (450).

A sensitivity analysis was performed to determine the impact on ranking of the alternative strategies. Fifty-five sets of criteria weights were used to reflect plausible variations in their importance. Throughout the sensitivity analysis there were only minor variations in the ranking of the alternative strategies, and strategies (3a) and (2b) ranked higher than the other strategies considered. Alternative strategy (3a) received the highest score in 53 of the 55 sets. This remained true except when the cost category was weighted very heavily relative to the other categories.

Based on the results of the FOM analysis the ranking for the alternative strategies investigated was:

FOM Score Out of 1000 points	Ranking
756	1. Alternative 3a: Retrieve waste, modify to a glass, dispose offsite
716	2. Alternative 2b: Retrieve waste, modify to a glass, dispose onsite
597	3. Alternative 5: Delay retrieval of waste, modify to a glass, dispose offsite
568	4. Alternative 3b: Retrieve waste, modify to a stabilized calcine, dispose offsite
540	5. Alternative 6: Delay decision on old waste, modify new waste, dispose offsite
505	6. Alternative 1: Leave waste in place
499	7. Alternative 2a: Retrieve waste, modify to pellet, dispose onsite
450	8. Alternative 4: Retrieve waste, separate actinide fraction, dispose actinide-depleted fraction onsite, dispose actinide fraction offsite.

Additional reviews and cost estimates have been provided to DOE-ID in support of the president's Defense Waste Management Plan. This additional information provided for a reduction in disposal cost per canister and a reduction in the total waste volume output of future facilities.

IN SITU STORAGE OR DISPOSAL

Residual Liquid Solidification — RHO

The objective of this task is to engineer and demonstrate solidification of Hanford defense waste liquids for in-tank disposal, with minimum use or need for interim double-shell tank storage.

Work is continuing to evaluate the feasibility of microwave radiowave energy as a method for in situ drying of residual liquid in Hanford single-shell tanks. The immediate efforts are evaluating a frequency of 70 MHz. Frequencies above 100 MHz have been eliminated by prior screening tests. The penetration range and permittivity values at the longer wave length are being determined. The immediate goal is to optimize the frequency required to penetrate wastes to the tank bottom to ensure thorough drying.

Following optimization of the frequency, work will be initiated on equipment design for transmission of energy into a tank.

A draft engineering study on solidification of customer waste has been completed. Preliminary results indicate microwave drying of a semihydrated salt placed in a single-shell tank is the most cost effective processing alternative. Processing the customer waste in the evaporator to 10% solids for sulfate waste and to 12% solids for phosphate waste followed by grout immobilization in single-shell tanks or near-surface vaults represented the highest life cycle cost of all alternatives evaluated.

Waste Tank Corrosion Studies — RHO

Waste tank corrosion studies are being made to define composition and storage temperature conditions such that the average corrosion rate will not exceed one mil per year (mpy) over a 50-yr storage period.

The corrosion testing program continued at PNL. The program is investigating the relationships of corrosion rate to the following variables:

- time (3 time periods)
- temperature (5-6 temperatures)
- steel type (2 steel types)
- steel condition (2 conditions stressed and unstressed)
- waste type (3 general waste types)
- waste composition (up to 9 components per waste type).

Of the sixteen reports scheduled for completion during the 2-yr corrosion program, three were produced during this quarter. These are:

- The 12-month low temperature double-shell slurry tests
- The 4-month low temperature future plutonium-uranium extraction (Purex) tests
- The 4-month low temperature customer waste tests.

Results of the 12-month low temperature double-shell slurry tests indicate that the average corrosion rate at 100°C for A-516 steel (used in single-shell tanks) is 0.11 mpy and for A-537 steel (used in double-shell tanks) is 0.14 mpy. The maximum observed corrosion rates at 100°C for these two steels were 0.59 mpy and 0.68 mpy, respectively. These corrosion rates are somewhat lower than those observed during the 4-month tests. This phenomenon was expected, since corrosion rates are typically higher during the early states of a corrosion test. The time dependence of the corrosion rates will be evaluated after the 8-month corrosion test data are evaluated (first part of the next quarter). The corrosion rates at the lower temperatures were typically lower than those at 100°C. Statistical fluctuations will be discussed after the 8-month test data are available. No cracking or pitting was observed.

The results of the 4-month low temperature future Purex tests indicate the mean penetration rate for A-537 ranges from 0.16 mpy to 0.95 mpy for all but two of the 36 compositions tested averaged over all temperatures. The two exceptions are dilute solutions (0.001M OH⁻ and 0.01M NO₂⁻) which indicated higher corrosion rates between 6.58 and 6.75 mpy. The upper range was as high as 16 mpy. This compositional range of high corrosivity is easily avoided by tank farm specifications. This information could have significant impact upon other operational activities such as line flushing procedures. The point is made that dilute solutions can be much more corrosive than concentrated solutions. With the exception of the two compositions discussed above, the highest corrosion rate observed at 100°C was 0.59 mpy. After the 8-month and 12-month tests are completed (July 1983) the time-dependence will be addressed, and the temperature dependence discussion will be expanded.

The results of the 4-month customer waste tests indicate that the mean penetration rates range from 0.30 mpy to 0.8 mpy at 50°C for 20 of the 23 test compositions. The other three compositions are dilute solutions (0.0001M OH⁻ and 0.01M NO₂ plus others) and are similar to the dilute solutions showing high corrosion rates in the future Purex tests. These compositions have mean penetration rates ranging from 1.0003 mpy to 3.5 mpy at 50°C. The maximum penetration rate was 5.0 mpy at 50°C. After the 8-month tests and the 12-month tests are concluded, the time-dependence of the corrosion rate will be addressed.

Cesium Capsule Corrosion Studies — RHO

The objective of this task is to evaluate the compatibility of the waste encapsulation and storage facility (WESF) produced ¹³⁷CsCl with 316L stainless steel under the conditions that would be encountered in a deep geologic repository or in a near-surface burial. For this study, it is assumed that the maximum CsCl metal interface temperature to be encountered by the capsules in a repository is 450°C.

During this quarter the metallographic preparation of samples from the first in a series of six test capsules has been completed at ORNL. Photographs have been taken and will be shipped to PNL for analysis. The second test capsule completed testing and was shipped to ORNL in December.

Previously, two capsules were destructively evaluated immediately after being filled with cesium chloride. The results showed that little or no attack of the 316L stainless steel occurred during that period when the high temperature molten cesium chloride was in contact with the inner capsule wall. These results provide the baseline condition against which the observations on the six capsules aged at 450°C for long periods can be compared.

WASTE FORM DEVELOPMENT AND CHARACTERIZATION

Alternative Waste Fixation Assessment — RHO

The objective of this task is to assess cost impacts of alternative waste fixation processes with and without isotope recovery.

The potential impact of isotope recovery upon existing and future Hanford waste management plans and costs are being evaluated on a total system basis (i.e., source to final disposal). Consistent cost estimation will allow better comparison of alternatives and use of the data for decision making. The information will

allow Rockwell to provide consistent input to DOE defense waste planning and establish priorities for waste fixation tasks. A draft report is nearing completion, and cost estimates are being finalized. The report will be issued for review during the second quarter.

A high-level waste disposal strategy document is being prepared for Hanford waste. The objective of this strategy is to establish at Hanford waste immobilization facilities capable of demonstrating and implementing the waste management processing disposal objectives. There are three key factors to the strategy.

- Make maximum use of existing facilities and equipment in achieving final disposal, including production of useful by-products from waste streams.
- Immobilize for deep geological repository disposal only that portion of the high-level waste which fits the definition of high-level waste or TRU waste per DOE Order 5820. To this end provide a facility with maximum processing flexibility and throughput capacity to handle the various waste streams at Hanford within a fixed cost envelope of \$85 to \$105 million.
- Immobilize the residual liquid waste resulting from processing the high-level waste above as a low-level waste form per DOE Order 5820. To this end provide a low-level waste immobilization facility(ies) to process for surface or near-surface disposal.

Waste fixation assessment technology is being focused on evaluating existing and future waste sources as feed to these facilities while minimizing the need for interim storage.

Glass Waste Form Studies -- ICPP

Many of the waste storage bins at ICPP contain mixed calcine types. These calcines and calcines formed from high sodium content waste combined with other types will result in more complex calcines to be vitrified. An important processing aspect of vitrifying these wastes involves the effects of complex calcine chemistry on melt viscosity. Thus, viscosity profiles between 1050 and 1200°C are being measured on glasses prepared with various mixed calcine types. Profiles are being obtained with Formula 127 glass prepared with zirconium-sodium blend calcine, fluorinel calcine, and fluorinel sodium blend calcine. Profiles are also being obtained on Formula 532 glass prepared with 2.5:1 alumina-sodium blended waste and fluorinel-sodium blend calcine. Work was started on these studies, which will provide valuable processing information about the mobility at reasonable melt temperatures of the glasses formed by these wastes and Frits 127 and 532.

Sulfur and cadmium will be constituents of future ICPP calcines. The behavior of cadmium, which is toxic under certain conditions, and sulfur, which can be volatilized from glass melts or, under other conditions separates as a separate phase, is important for process information. This Formula 127 glass has been prepared with simulated fluorinel calcine containing both sulfur and cadmium. Likewise, Formula 532 glass has been prepared with the same calcine and 2.5:1 alumina-sodium waste containing sulfur. The glasses prepared from these mixtures are homogeneous and have completely vitrified the calcine reactants. Samples of these glasses are currently undergoing chemical analysis for cadmium and sulfur to determine the behavior of these elements during vitrification.

Test Method Development — MCC

The codification of test methods for nuclear waste forms and metallic engineered barriers is a major function of the MCC. Progress during this reporting period is described here.

- MCC-1 Static Leach Test (Radioactive). Quartz leach containers are being tested. Problems have been encountered in providing a noncontaminating seal that will satisfactorily prevent leachant loss at 90°C.
- MCC-3 Agitated-Powder Leach Test Method. Because the original design of the roller ovens was found unreliable, the roller assembly was modified and has proven to be sturdier and very reliable.

A series of tests to determine the particle size effects was conducted with the result that two different particle sizes are now being specified for use in the MCC-3 test.

- MCC-4 Low Flow Rate Leach Test Method. This test method was resubmitted to the Office of The Chairman-Materials Review Board (OTC-MRB) in June 1982 and returned after review for further modifications. These modifications were made and the method resubmitted.

MCC-4 is considered a very important leach test. The type of data generated by this test method is shown in Figure 1. Note that a steady state is apparently attained after about 2 months leaching at 90°C at flow rates of 1.79 and 15.4 mL/day. The steady-state leachant concentrations are a function of flow rate, increasing as flow rate decreases for elements such as boron and silicon. Data obtained with the MCC-5 soxhlet leach test method (flow = 1.4 L/day) and the MCC-1 static leach test (zero flow) are shown for comparison in Figure 1. The leaching behavior of calcium is a function of solubility limitations that

are strongly influenced by increasing pH; thus its behavior is much different from boron and silicon, as shown in Figure 1.

- **MCC-6 Method for Preparing and Characterizing Actinide-Doped Waste Forms.** This method is presently the only MCC test method for simulating long-term radiation effects on waste forms, based on the assumption that alpha and alpha-recoil particles provide the predominant sources of significant radiation effects. A series of tests are now under way to investigate the effects of purely ionizing radiation on waste forms and related materials. Specimens are irradiated in the gamma field of a ^{60}Co irradiation facility and are examined periodically for property changes. Results to date show no macroscopic swelling, but color centres are induced in all materials.

FIGURE 1. Effect of Flow on Leachate Composition

- MCC-12 Test Method for Determining Density Changes in Actinide-Doped Waste Forms. This method was first submitted to OTC-MRB on April 19, 1982. After accommodating some comments by the OTC-MRB, the method was resubmitted with a request for full approval. At the end of the reporting period, MCC-12 was ready to be distributed to the Procedures Panel of the MRB for a letter ballot on full approval.
- MCC-101 Unstressed, Static-Immersion Corrosion Test Method. This test method was first submitted to the OTC-MRB on August 27, 1982.
- MCC-102 Unstressed, Flowing-Immersion Corrosion Test Method. This test method was first submitted to the OTC-MRB on August 13, 1982.
- MCC-103 Stress-Corrosion-Cracking Susceptibility Test Method. This test method was approved as a provisional test method by letter ballot of the MRB Procedures Panel on July 12, 1982, provided certain minor modifications were incorporated into the wording of the test method. These modifications were made, and at the end of the reporting period MCC-103 was ready to be submitted to the full MRB for letter ballots.
- MCC-104 Stress-Corrosion-Cracking-Rate Tension Test Method. This test method was first submitted to the OTC-MRB on September 30, 1982.

Reference Glass Materials - MCC

Did invitations for supply of 500 lb of simulated waste glass were sent to nine potential manufacturers. A test sample of the specified glass formulation, composition shown in Table 1, was prepared at the National Bureau of Standards (NBS) and analyzed at NBS and PNL. The sample was homogeneous and easily melted.

A contract with the NBS for characterizing the glass, which will be manufactured as about 50 billets, is in the final stages of negotiation. The NBS will document the homogeneity of the glass within and between billets. The glass will be used to produce approved reference materials (ARM) for use with MCC test methods. The first ARM prepared from the glass will be for use with the MCC-1 static leach test method.

TABLE 1**Composition of Glass for Which Bids Have Been Tendered**

<u>Component</u>	<u>Concentration, wt %</u>	<u>Component</u>	<u>Concentration, wt %</u>
SiO ₂	46.75	SrO	0.47
B ₂ O ₃	10.96	ZrO ₂	2.10
Na ₂ O	10.00	MoO ₃	1.96
Li ₂ O	5.00	Cs ₂ O	1.36
Al ₂ O ₃	5.75	BaO	0.66
ZnO	1.51	P ₂ O ₅	0.57
TiO ₂	3.43	Nd ₂ O ₃	5.78
CaO	2.31	CeO ₂	1.41

PROCESS AND EQUIPMENT DEVELOPMENT**The Fifth Run of the SRP Large Slurry-Fed Melter****Introduction**

The fifth experimental run of the large slurry-fed melter (LSFM) was completed on June 29 after fourteen days of continuous operation. The primary objectives of LSFM-5 were to:

- Demonstrate long-term operation of melter and off-gas system under steady-state conditions.
- Demonstrate steady-state melting with 45 wt % solids feed.
- Further examine the effects of electrode power skewing and determine the optimum power skewing ratio.
- Study the effect surfactant in the scrubber liquid has on the plugging potential in the washable deep bed filter.
- Characterize the off-gas stream with emphasis on cesium and semivolatile deposits.
- Determine the extent of melter entrainment under various vapor space temperatures and power skew ratios.

The entire system (feed, melter, and off-gas) performed well during LSFM-5 with no major problems occurring which resulted in a complete shutdown. Extending the length of the run enabled time

to reach "steady-state" during short periods which not only made operation smoother and easier but also improved the quality of data. Highlights of the run are listed below:

- Approximately 7500 gal of 45 wt % slurry containing Frit 131 and TDS simulated waste were fed to the melter producing over 35,000 lb of glass.
- Excluding outages, total feed time was 318 hr for an average melt flux of 9.17 lb/hr-ft². Peaks of up to 10 lb/hr-ft² were obtained for short periods.
- Total nonfeeding time including can changes was limited to 18.5 hr. This corresponds to a 94% attainment.
- One hundred and thirty-six off-gas pressure surges of greater than two times the normal melter off-gas flow were recorded. Of these, eight surges were 5X or greater.
- The washable deep bed filter (WDBF) operated the length of the run without a single plug occurrence.

Figure 2 is a diagram of the complete melter, off-gas, and feed systems showing the various components that were in service during LSFM-5. The only change from the LSFM-4 system was the removal of the ruthenium adsorber bed from the process.

Discussion

Feed Preparations

The liquid slurry feed used during LSFM-5 was a uniform mixture of simulated defense waste sludge and glass formers (frit), Table 2. Three 2500 gal batches of sludge were then prepared in the slurry mix evaporator (SME Runs 3, 4, and 5) located in 678-G. Formic acid was added and the sludge refluxed for the required time to reduce the oxides MnO₂, CaCO₃, Ni(OH)₂, KOH, and SrCO₃ to their corresponding formated states: Mn(CHO₂)₂, Ca(CHO₂)₂, Ni(CHO₂)₂, KCHO₂, and Sr(CHO₂)₂. Glass Frit 131 was then added as a water slurry on a 62:28 ratio weight of dry frit to weight of dry sludge solids. Finally, the mixture was concentrated by heating and evaporation, adjusted to 45 wt % total solids by water dilution, and transferred to the slurry hold tank (SHT) located in 675-G.

The design yield stress and consistency for slurry feed is 50 to 250 dynes/cm² and 5 to 50 cp. Slurry feed during LSFM-5 was relatively thin at 50 to 80 dynes/cm² and 7 to 17 cp. Rheology of the slurry varied with each SME batch corresponding to variations in the frit particle size. This suggests that decreasing the frit particle size increases the yield stress of the slurry.

FIGURE 2. Flow Diagram of 675-G Melter Process

TABLE 2

Slurry Compositions and Rheologies

Component	SME Batch, wt %		
	1	2	3
Fe(OH) ₃	40.8	36.4	37.0
Al(OH) ₃	16.4	16.1	15.0
Mn(COOH) ₂	14.9	13.0	14.2
SiO ₂	7.5	12.6	11.8
Ca(COOH) ₂	8.4	7.7	7.8
Zeolite	4.8	4.7	4.4
NaNO ₃	2.4	3.9	4.3
Ni(COOH) ₂	4.3	3.7	4.0
KCOOH	-	0.7	0.5
Na ₂ SO ₄	0.4	0.7	0.7
SrCOOH	-	0.4	0.4
CsOH	0.1	0.1	0.1
Rare Earth*	0	0	0
Sludge: Frit	38.7:61.3	37.6:62.4	37.7:62.3
Frit 131 Composition	64% (1)	87% (3)	100% (4)
(Ramtec)	36% (3)	13% (2)	-
pH	7.26	7.55	7.43
Mesh Size	67% (-80+200)	61% (-80+200)	65% (-80+200)
	33% (-200)	39% (-200)	35% (-200)
<u>Feed Rheology During the Run</u>			
Wt %	45.16 ±0.12	45.01 ±0.15	45.16 ±0.28
Density, g/cc	1.40 ±0.02	1.40 ±0.03	1.42 ±0.03
Yield Stress, dyne/cm ²	51 ±6	83 ±14	65 ±16
Consistency, cp	7 ±1	17 ±3	15 ±3

* Not added to sludge for LSFM-5

Feed Systems

The current DWPF reference feed system calls for a one-inch recirculation loop with a takeoff line to the melter. Alternative methods for feeding, however, are being investigated at the Equipment Test Facility (ETF). During LSFM-5, slurry was fed to the melter using the pressurized feed system (PFS) which, as the name describes, utilizes a pressurized vessel to force the feed into the melter.

A flow diagram of the LSFM feed system is presented in Figure 3. Typically, slurry is transferred from the SME in 678-G to the 5000-gal slurry hold tank (SHT) in 675-G. A centrifugal pump, defined for the DWPF, has not yet been developed. Therefore, the transfer pump, to be tested at the Engineering Test Facility (ETF) beginning in January, is not suitable for the recirculation loop. Instead, a Wilden diaphragm pump is used to circulate the slurry through approximately 150 ft of one-inch stainless steel pipe. Slurry is continuously drawn off the recirculation loop into the 100-gal peristaltic feed tank (PFT) at about the same rate as the melter feed rate (typifying the DWPF reference system).

From the PFT, the slurry is transferred to the 15-gal pressurized feed tank (PRFT) using a diaphragm pump. Air pressure is applied to the tank forcing the slurry through approximately 100 ft of 3/8-in. stainless steel tubing. The feed rate is manually controlled using a backpressure valve to adjust the tank pressure. A weight scale is used to monitor the contents of the tank while a magnetic flowmeter measures the feed rate.

During feeding to the melter, the PRFT was refilled with slurry at five minute intervals, each refill lasting approximately ten seconds. As the level in the tank dropped, the feed rate would fall off slightly. Conversely, when the tank was refilled the rate would rise. Because the system operated by manual control, it was not possible to eliminate this 5% feed rate fluctuation. By centering the middle of each crest and trough at the desired feed rate, the "long term" effect of the fluctuations were eliminated resulting in a semiconstant feed rate.

A peristaltic pump was available as a backup feed system and was used at the beginning of the run when problems occurred with the pressurized feed system. Although not a canyon type system, the peristaltic pump is a reliable system for melter experiments.

Sixty-two feed outages (nonfeeding periods) occurred during the run for a total downtime of 18.5 hr. Over the 330 hr of operation, this corresponds to a 94% attainment. Table 3 summarizes the outages according to the problem that occurred, the majority of the time is due to canister changes.

FIGURE 3. LSFM Feed System

TABLE 3**Feed Outage Summary**

Total Run Time		330 hr
Feed Outages:		
Feed nozzle plugs	(7)	1.3 hr
Feed nozzle flushings	(12)	0.3 hr
Canister changes	(10)	12.7 hr
Melter pressurizations	(27)	0.5 hr
Cold-cap burn-off (overfeeding)	(2)	2.7 hr
Others	(4)	1.0 hr
Totals	(62)	18.5 hr
Total Feed Time		311.5 hr
Attainment		94%

Melter Systems

The melter performed well over a variety of operating conditions during LSM-5. No adverse effects of long-term operation were detected over the 14-day test. Besides demonstrating long-term, steady-state melter operation, the effects of electrode power skewing on melt rate and melting behavior were examined. Power skewing refers to the intentional setting of an uneven power input per electrode pair so that one pair furnishes the majority of the power to the glass. The intent is to increase melt rate by increasing convection currents in the melt pool and to distribute more power to the melter bottom.

Power Skewing Test

During the run, four separate power skewing ratios were tested (3:1, 10:1, 1:3, and 1:1). These ratios refer to the ratio of the power produced by the footed electrode pair (BD) to that of the nonfooted pair (AC). On the LSM, a 3:1 power skewing distributes approximately 7% more power to the melter bottom while a 10:1 skew only produces an additional 3%. Because of electrode geometry, it is not possible to obtain a 0.7 to 1.3 power skewing which is required for the DWPF melter.

In general, even (1:1) and moderate (3:1, 1:3) power skewing produced better melting conditions than the extreme power skewing (10:1). However, the lower results with extreme skewing are more a

function of the LSFM electrode placement and geometry and not from actual power skewing. A melt rate of 9.5 lb/hr-ft² was demonstrated with the even and both moderate power skews. Melting behavior was very similar for all three settings. Table 4 lists the specific melt rates obtained for each canister filling period. Extreme skewing produced unstable conditions at a similar feed rate. Normally, this is detected by oscillations in the liquid level trace and an increase in off-gas surges. Both indications are evidence of trapped gases beneath a thick cold-cap layer. A third method for detecting instability is through visual observations of the melt surface.

Figure 4 shows typical temperature profiles within the melt pool for each of the power skew settings while pouring at approximately 9 lb/hr-ft². These measurements were taken from thermocouples positioned at various heights in a thin-walled thermowell. Temperatures near the bottom of the melter are similar while those near the top vary substantially. Thus, the thermal boundary layer and heat transfer coefficient between the cold cap and bulk glass are significantly different for each power skew ratio.

TABLE 4

LSFM — 5 Melt Rates

Can	Power Skew Ratio BD/AC ^A	Vapor Space, °C	Glass Produced, lb	Pour Time, hr	Rate, lb/hr-ft ²
1	3/1	800	3192	28.75	9.25
2	3/1	800	3322	27.78	9.96
3	3/1	800	3280	27.58	9.91
4	3/1	700-800	2700	26.42	8.52**
5	10/1	800	3136	29.12	8.97**
6	10/1	800	2888	29.25	8.22**
7	1/3	800	2676	26.75	8.33
8	1/3	800	3410	30.08	9.44
9	1/1	800	3537	31.75	9.28
10	1/1	800	3464	30.7	9.40
11	1/3	700-800	3400	29.73	9.53**
Total			35005	371.91	9.17

* BD is the footed electrode pair.

** Feed rates were lowered after unstable conditions occurred.

FIGURE 4. LSFM-5 Power Skewing Temperature Profile

Because the extreme (10:1) power skewing has the largest boundary layer, the corresponding melt rate should be and was the lowest (at 8 to 8.5 lb/hr-ft²). Furthermore, the remaining three ratios have similar profiles which would result in similar melting behavior and rates. Data in Table 5 also agree with this. The table lists the average power inputs for each power skew. The average electrode power input includes the sum of both electrode pairs while the total power input also includes lid heat power. Both values show that less power was required with the 10:1 power skewing, while the three remaining skew ratios resulted in higher and similar heat inputs. Thus, heat transfer to the cold cap (and heat loss to the vapor space) is hampered with extreme power skewing to the footed electrodes, probably because heat input is skewed toward the melter bottom when it is needed more near the melt surface. This also indicates that additional convection cells in the LSM caused by power skewing do not significantly increase heat transfer to the cold cap. Experience has shown that each individual melter reacts differently to a change in operating conditions. Factors effecting melter performance include electrode placement, melter geometry, and actual applied power skewing.

TABLE 5

Power Input Versus Power Skewing

<u>Power Skew Ratio*</u>	<u>Average Electrode Power Input, kW</u>	<u>Average Lid Heat, kW</u>	<u>Total Power Input, kW</u>
10/1	76	84	160
3/1	80	87	167
1/1	88	85	173
1/3	86	83	169
<u>800°C vapor space</u>			

* kW footed electrode/kW nonfooted electrodes

Vapor Space Temperature Test

During the majority of the run, the vapor space temperature in the melter plenum was maintained at 800°C by four sets of resistance lid heaters. On two occasions, the vapor space temperature was lowered to 700°C to determine the effect on melting behavior. This occurred after a 9.5 lb/hr-ft² melt rate was demonstrated for 3:1 and 1:3 power skew ratios (800°C vapor space temperature). Melting capacity was exceeded at the lower temperature for both power skewings and resulted in very unstable conditions (90% to 100% cold-cap coverage and frequent, large off-gas surges). Raising the vapor space temperature to 750°C did not appear to improve

conditions. Apparently, the 9.5 lb/hr-ft² melt flux is near the maximum for the melter at these operating conditions:

45 wt % solids feed slurry (Frit 131)

1150°C bulk glass temperature

800°C vapor space temperature

In general, lowering temperatures will reduce melter capacity.

Cold-Cap Behavior

Visual observations of the cold cap also indicate near maximum melter capacity. For each of the three better power skewing ratios, cold-cap coverage ranged from conservative (60 to 70% of melt surface) for the majority of the run to near complete coverage (80 to 90%) for short periods. Although occasionally it appeared that the melter was approaching unstable conditions (indicated by a large cold cap), the situation always improved. Thus, it appears that the melter operated near or at its maximum, for the given conditions.

Cold-cap behavior was similar to that of previous runs. Rivers of slurry formed and would flow toward a particular side of the cold cap. This would continue for a few minutes until that area dried out. Then the flow would shift directions to a different side. Normally, two to four active vent holes ranging from one to 6-in. in diameter were present in the cold-cap surface with a light orange foam surrounding it. The 45 wt % feed slurry spread over the melt surface similarly to the previously fed 40 wt % slurry. As before, feeds entered approximately 8-in. off-center causing the cold cap to center on the southwest quadrant of the melter. This did not appear to effect the melt rate substantially, however. It is possible that melt rates would increase slightly if the feed entered on-center; however, because of melter design, center feeding on the LSFM is not feasible.

During periods of instability (such as when the vapor space temperature was lowered to 700°C) the cold-cap coverage increased to >90%, and vent holes were either nonexistent or small and few in number. Trapped gases accumulated beneath the rigid cold-cap layer until the pressure built-up enough to rupture the surface releasing a surge of off-gas. This pattern tended to repeat itself over a 1- to 2-hr time period during both unstable and marginal melter operating conditions.

Melter Operational Problems

During the entire run, a deflector was used on the feed nozzle to help direct feeds toward the melter center (the feed nozzle port is about 11-inches off-center). The deflector consisted of an uncooled section of tubing protruding from the bottom of the feed nozzle angled toward the melter center. Problems occurred when dried feed material built-up on the deflector restricting the flow (sometimes causing complete pluggage). Pluggages occurred seven times for no apparent reason and an additional three times during outages in which the feed nozzle was not adequately flushed. This problem will possibly be eliminated in the future by removing the uncooled section of tubing and redesigning the bottom portion of the nozzle to avoid the buildups.

A second problem occurred when all of the thermocouples in the main thermowell failed from an electrical short between the thermocouples which resulted in erroneous glass temperature readings. Temperature control was shifted temporarily to the secondary thin-walled thermowell. Additional electrical insulation surrounding the thermocouple wires will eliminate this problem in the future; however, this does indicate the importance of having a second and separate thermowell to ensure adequate melter temperature control.

An additional minor problem involved heat sensitive components in the lid heater and riser signal transmitters. Initially, instrumentation on both indicated higher power usage than was actually supplied. The riser heater indicated full power when riser melt temperatures were 15°C below the specified value (1125°C). Both problems were traced to the heat sensitive transducers and were easily corrected.

Finally, as in past runs, the pour spout temperature remained below specification at only 800°C despite a full-power input to the heating element. No operational problems were caused by this, but additional heat is needed in this area to adequately pour in higher viscosity glass such as the new Frit 165 glass. Prior to LSFM-6 a new Kanthal® (The Kanthal Corporation) electrical resistance heater element will be installed to provide an adequate amount of power to heat the spout to 1050°C. This requires the addition of a secondary transformer to adequately power the new element as well as a slight modification of the pour spout area to adapt to the new element dimensions.

Post Run Melter Data

At the end of LSFM-5 a series of samples were taken to characterize the glass left in the melt tank. After feeds were stopped, 2 to 3 hr were required for complete cold-cap burnoff to a clear melt surface. Then a spinel probe, melter bottom sample, and a

melter surface sample were taken. There was no measurable amount of spinel accumulation and no crystals were found in the melter bottom sample. An analysis on the glass surface sample showed a sulfur content of 0 wt %. This agrees with visual observations of the melt surface which indicate there is not a sulfate layer present. A spinel probe, melter bottom sample and melt surface sample will be taken directly after all future LSFM runs to determine if a sulfate layer or spinels and metals are accumulating in the LSFM.

Off-Gas System

The off-gas system performed well during LSFM-5 with minimal problems. High air inleakage rates continued to exist, however, inhibiting optimal performance. By lowering the melter vacuum to 1-in. water column, the inleakage was reduced substantially compared to previous runs. (LSFM-4 inleakage rates averaged approximately 800 lb/hr at 2-in. vacuum versus 400 lb/hr at 1-in. vacuum for LSFM-5.) Nonetheless, use of additional air dilution to remain below lower explosion limits (LEL) was not necessary.

Off-Gas Surges

Because of frequent periods of unstable melter operation during the run, a relatively high number of off-gas surges were observed. Using a Pitot tube located approximately 30-ft downstream from the melter, the size, duration, and frequency of the off-gas surges were recorded for the length of the run.

Results from analysis of the data revealed 136 surges of greater than twice the calculated process off-gas flow (excluding purges and air inleakage) over the 14 days of feeding. Of these, eight surges were greater than 5X. Table 6 summarizes the surge data according to peak size.

TABLE 6

Summary of Off-Gas Surge Data

Size*	Number Surges	Base Duration, min	35% Duration, min
2X-3X	81	2.9	0.44
3X-4X	33	2.7	0.35
4X-5X	14	4.1	0.56
>5X	8	8.1	**

* Based on calculated process flow for 9.0 fps

** Insufficient data

The surge typically rises rapidly to the peak approaching a step function, then decays linearly to approximately 35% of the peak height. From that point the decay is exponential to the base line. This decay can be conservatively approximated by a straight line. Six data points were recorded for each peak enabling determination of mean heights and durations. These values are also summarized in Table 6.

While the size of a surge is important for determining off-gas system capacities, the frequency of surges seems to provide an indication of melter stability. Periods of stable and unstable operation can be distinguished by a definite pattern in off-gas surges. In conjunction with the bubbler level indicator, over feeding situations can be detected before serious problems arise.

A cold-cap coverage of 80 to 90% of the melter surface is desired to utilize as much surface area as possible for melting (which is 12 ft² for the LSM). These limits were set to prevent forming a rigid cold cap under which evolving gases could build up pressure forcing an excessive discharge of glass. For this reason, at least one vent hole for gases to escape through must always be visible (through melter view ports) in the cold-cap layer to be considered stable operation. On the other hand, if no vent holes are open in the cold cap, the possibility of a sudden release of gases increases, which frequently causes an over pressurization in the melter, resulting in unstable conditions. Surge patterns are good evidence of melter stability, and is especially useful when visual access to the melter surface is unavailable (for example, if the canyon melter boroscope failed).

Unstable conditions are generally depicted by a cyclic pattern of a period of little activity (surges $\ll 2X$), followed by a period of moderate surge activity (surges of around $2X$) and finally a large surge (4 to $6X$).

Stable periods are apparent by frequent surges of 2 to $3X$ in magnitude, which indicates the steady venting of gases. Average frequencies of surges during stable and unstable periods are listed in Table 7. Notice the frequency of 2 to $3X$ surges is higher for stable versus unstable conditions, whereas 4 to $5X$ surges occur more frequently during unstable periods. This further confirms the accuracy of surge patterns as a model of melter stability.

Off-gas surge data has been proven to be a valuable tool for both sizing off-gas equipment and predicting melter stability.

TABLE 7

Frequency of Off-Gas Surges Per Hour

Surge Magnitude	Melter Operation	
	Stable	Unstable
<2X	0.61	0.27
2X-3X	0.45	0.12
3X-5X	0.13	0.0073
4X-5X	0.025	0.055
>5X	0.008	0.027

Washable Deep Bed Filter (WDBF)

The WDBF was the primary off-gas scrubbing device for the process, handling the bulk of the particulate removal. The unit consists of a twin-staged arrangement of high-density polypropylene pads (3-in. thick), each with an individual cocurrent rinsing water spray. Recently, the plugging problem in the WDBF has been greatly diminished as a result of using a surface-active agent (surfactant) in the filter rinse water. During LSFM-5, no pluggages occurred over the 14-day length of the run.

From past experience Zonyl® (Du Pont), a surface-active agent, has shown favorable results in reducing the plugging potential in the WDBF. At a concentration of 0.001%, the surfactant lowers the surface tension of the filter rinse water from 72 to 38 dynes/cm. This enables the water to penetrate the tiny pores of the filter pad more efficiently providing a better rinse.

Because the shipment of surfactant was late arriving, the unit was bypassed by the first 35 hr of the run. After it was put in service with the surfactant added to the rinse water, the WDBF exhibited varying pressure drops. The large variations generally coincide with changes in the scrubber tank vacuum.

As the vacuum decreased (approached zero), the off-gas velocity through the WDBF also decreased, thus causing the pressure drop to fall. Several periods of gradual increases in pressure drop are evident which seem to indicate a slow buildup of material on the filter pads. Although insufficient data were collected to make accurate correlations, the pressure drop across the WDBF is closely related to scrubber liquid solids content. Entrained solids from the melter are carried over to the well-mixed scrubber tank where they approach a constant level in the liquid because of periodic dumps. Changes in feed rates or melter conditions have some effect on the percent solids. One instance occurred, between 250 and 300 hr, where the percent solids sharply dropped, then leveled out.

The WDBF pressure drop curve followed the same trend during that period which qualitatively illustrates the effect scrubber liquid percent solids has on WDBF pressure drop (and plugging potential). Furthermore, the surfactant appears to reduce the rate of increasing pressure drop. Therefore, by maintaining the rinse water solids content below 0.1% and by using a surfactant, plugging problems in the WDBF can be practically eliminated.

In future runs, additional scrubber liquid data will be collected to more closely examine the relationship to WDBF pressure drop. Because sample pumps failed, DF measurements could not be obtained.

Off-Gas Cooling

The steam cooling system (SCS) was used during LSFM-5 to cool the off-gas stream exiting the melter to 350°C. Except during can changes (where steam is shutdown), the system operated well holding the temperature relatively steady. The steam flow rate was regulated manually and, as required, step changes in flow were made to control the off-gas temperature.

Between the hours 229 and 233, steam was lost due to a boiler failure. Even after the supply was regained, the flow was very erratic making it difficult to control the temperature as well. During this period, an increasing pressure drop in the off-gas line exiting the melter was observed which generally indicates the formation of a plug in the pipe. Upon completion of the run, the line was opened for inspection revealing a hard, glassy material plugging 80% of the pipe. Lengthwise, the plug extended from the opening to approximately 3 in. into the pipe. Downstream of the plug was a hard, friable, crusty layer ranging from 1/4- to 1/2-in. thick which could be easily knocked off with a probe.

The formation of the plug seems to have resulted from the 4-hr steam outage. During this time, pipe temperatures rose to well over 600°C which is sufficient to promote adhesion of entrained materials. The plug also seemed to cause enough restriction of the off-gas flow to reduce the system's capability to handle surges. This would explain the number of melter pressurizations during the last 80 to 90 hr of the run.

Ejector Venturi Quencher (EVQ)

The purpose of the EVQ is to quench the hot off-gas stream to 50°C, thus removing the majority of the water vapor and the larger particulates. During LSFM-5, cooled water was circulated through the unit at a rate of 60 to 65 gpm using a 80-hp centrifugal pump.

The EVQ was designed to be oversized and normally operates with a positive pressure drop. As a result, almost all of the off-gas surges from the melter are damped out at the quencher. The EVQ pressure drop is also a function of the gas velocity through the unit which varies with scrubber tank vacuum. Generally, as the vacuum increases, velocity increases, which in turn causes the EVQ ΔP to rise.

Condenser Performance

The shell-in-tube condenser unit follows all of the primary scrubber units to remove any residual water and semivolatile vapors. Using a chilled brine medium, the condenser cools the off-gas stream to 10°C. The condenser performed well during LSFM-5 and greatly reduced the exhauster load to pump water.

Iodine Adsorber Bed (IAB)

Because no radioactive iodine is involved in simulation experiments, the IAB essentially provides typical adsorber pressure drop in the system. In fact, the ruthenium adsorber bed was bypassed to reduce the overall system pressure drop.

Periodically, inadequate operation of the condenser unit or preheaters results in the collection of water in the IAB packing which produces a higher than normal pressure drop. This is usually remedied by simply draining the bed.

During LSFM-5, the IAB pressure drop showed a steady decline. This is probably due to the gradual removal of residual water in the packing which seems to lessen the restriction. By the end of the run, the pressure drop had leveled out at around 5 in. of water.

Off-Gas Sampling Analysis

Melter Entrainment Studies

Quantitative results of melter entrainment studies revealed a two-fold increase in entrainment for vapor space temperature of 800°C versus 700°C. Results are listed in Table 8. Similar results were obtained at PNL. Varying power skew ratio did not appear to have any noticeable effect on entrainment.

TABLE 8**Melter Entrainment Measurements**

<u>Melt Flux</u>	<u>Vapor Space Temperature</u>	<u>DF</u>	<u>% Entrainment</u>
9 lb/hr-ft ²	800°C	28	3.6*
9 lb/hr-ft ² **	700°C	58	1.7*

* Calculated from off-gas particulate concentration data.

** Overfeeding situation.

Combustible Gases

A continuous percent combustibles analyzer was utilized during LSM-5 to monitor combustible gas (CO, H₂) content in the off-gas stream. Because of high air inleakage rates, the combustibles content was unmeasurably low. However, during off-gas surges, a rise in the content was observed, but always remained below 10% of the lower explosive limit (LEL). Sample bombs were also collected and analysis revealed a high amount of O₂ and N₂ (air) and an undetectable amount as CO and H₂.

Cesium Deposit Studies

Total filter samples were used to determine the amount of cesium entering the 55-ft section of 6-in. off-gas piping between the melter and the EVQ. The quantity of cesium removed from the off-gas (filtered) and the amount collected in the scrubber liquid (Table 9) were calculated from the increase in the cesium concentration of the scrubber liquid plus the amount lost in the discharged liquid. Typically the liquid level in the scrubber tank increases as the water vapor (generated at melter and steam cooling) was condensed by the EVQ. Periodically, the scrubber tank automatically discharges a specified amount when the liquid level reaches a maximum.

As seen in Table 9, only 11 to 19% of the cesium entering the off-gas line was found in the scrubber liquid.

Because of pump problems, a cesium balance across the WDBF could not be obtained. In investigations during SMOG-3, however, a maximum of 0.337 g/hr of cesium exhausted through the stack was measured. Since the WDBF was not in service during an earlier run, this number could be used as a conservative estimate for the cesium exiting the scrubber tank. In doing this, the data in Table 10 were generated.

TABLE 9

Quantities of Cesium Entering Off-Gas Line and Scrubber Liquid

Canister	Total Cs Entering OG Line, g*	Total Cs in Scrubber Liquid, g	Percent Cs Throughput
3	45	6.8	15
4	43	7.1	17
5	41	6.2	15
6	51	5.4	11
7	47	5.1	11
10	46	8.7	19

* Determined by total filter samples (may be off by as much as 50%, but appears reasonable).

TABLE 10

Off-Gas Line Cesium Deposits

Canister	Total Exiting Scrubber Tank, g*	Total Cs Exiting OG Line, g**	Total Cs Deposited in OG Line, g
3	9.3	16.1	28.9
4	8.9	16.0	27.0
5	9.8	16.0	25.0
6	9.9	15.3	35.7
7	9.0	14.1	32.9
10	<u>10.9</u>	<u>19.1</u>	<u>26.9</u>
Average	9.5	16.1	29.4

* Calculated by multiplying 0.337 g Cs/hr times feeding hours per can.

** Sum of Column 3 of Table 9 and Column 2 of Table 10.

Table 10 reveals that an average of 29.4 g of cesium per canister may have deposited in the off-gas line during LSFM-5 (number may be off by as much as 50%). Considering eleven canisters will be filled, this corresponds to a total of 323.40 g (0.71 lb) of cesium may have accumulated in the line over the 14-day run.

Deposit samples removed from the off-gas line at the completion of LSFM-5 analyzed as high as 1.5 wt % cesium near the melter to 0.30 wt % at the EVQ. Over 166 g of entrainment and semivolatile deposits were removed from the line by vacuuming, which works out to approximately 3 g/ft. Although this is not a substantial amount, the possibility of cesium accumulation may present problems (especially in the canyon). By increasing line velocities to around 60 to 70 ft/sec, however, the majority of the deposits would probably be transported through the line to the quencher and scrubber tank. Further experiments are planned to obtain more accurate measurements of off-gas line deposits.

Semivolatile Deposits Studies

Neck deposit coupons were inserted into several canisters during the run to collect semivolatile deposits. After the canisters were filled, inspection of the coupons revealed a thin, white coating. Analysis of the material showed high content of NaCl as well as some cesium and sulfates.

At the completion of the run, the pour shroud was overhauled to install larger heating elements. Again, a white powdery material, as much as 10-mil thick, coated the shroud. Also, a hard yellow deposit was discovered on the surface of the refractory between the pour spout and the melter shell. A similar yellow deposit was previously analyzed to be high in sodium sulfate. Apparently, hot vapors coming in contact with the refractory produces the reaction to form these deposits. Samples of actual shroud deposits were sent for analysis, and the results will be reported when available.

Off-Gas Flow Surges from the Slurry-Fed Melter

Introduction and Summary

Surges in the flow of off-gas from a slurry-fed melter occur from the unsteady evaporation of liquid that collects on and possibly within the crust covering the surface of the molten glass. During the fifth run of the slurry-fed melter at the ETF, flow surges were measured by a Pitot tube located in the melter off-gas line. Data from the 14-day run yielded information concerning:

- The pattern of surge activity as an indicator of approaching melter instability.
- Surge characteristics needed for off-gas system design.
- Applicability of a Pitot tube as a detector of flow surges.

Discussion

Surge Patterns

Continuous recordings of off-gas flow during the experimental run revealed distinguishable surge patterns which correlate with melter operating conditions. The slurry-fed melter was fed a 45 wt % waste slurry containing Frit 131 for over 330 hr. Several tests were performed during this time, such as varying power skew ratios and plenum temperatures, to evaluate their effect on melter stability. Pertinent run information is presented in Table 11.

TABLE 11

Off-Gas System Operating Conditions

Melter vacuum	-1.0 in. water
Air inleakage	400-500 lb/hr
Plenum temperature	700-800°C
Cooled off-gas temperature	300-380°C
Steam for cooling	200-450 lb/hr
Velocity in 6 in. line	48 fps
Surge velocities	>96 fps
Washable deep bed filter	in service 300 hr

Patterns of frequent, low magnitude (2 to 3X) surges were representative of stable melter conditions. As unstable conditions approached, the surge behavior changed to a cyclic pattern, beginning with a period of little or no surging, followed by a brief period of moderate surging (2 to 4X), and finally a large surge (>5X).

During filling of each canister (taken to be one period), the intent is to keep melter operating conditions constant. A stable period was defined as a canister fill during which no changes in melter operating parameters were necessary. Conversely, an unstable operating period was a canister fill during which a judgment was made that feed rate needed to be reduced or power boosted.

The magnitude and frequency for stable and unstable melter operating periods were analyzed. Table 12 lists the frequency of surges per hour for several multiples of the base line flow (which was 10 excluding air inleakage and steam injection). Notice that the frequency of the moderate surges (2 to 3X) is higher during stable periods indicating the steady release of gases. On the other hand, high magnitude surges (>4X) occur more frequently during unstable periods signifying the formation of a rigid, impermeable crust layer. During operation, however, these trends are not readily apparent and do not provide as useful an indicator of stability as the surge pattern previously mentioned.

TABLE 12

Frequency of Off-Gas Surges Per Hour

<u>Magnitude</u>	<u>Melter Operation</u>	
	<u>Stable</u>	<u>Unstable</u>
<2X	0.61	0.27
2X-3X	0.45	0.12
3X-4X	0.13	0.073
4X-5X	0.025	0.055
>5X	0.008	0.027

Apparently, the melter can operate indefinitely with the cyclic response, yet the pattern is a definite indication that an unstable, or overfeeding situation is nearing. Small increases in feed rate or decreases in auxiliary lid heat on the order of 10% lead to a positive indication of unstable conditions in which pressure begins to build under a rigid crust layer. Output traces from a liquid-level detector (bubbler type) immersed in the glass pool exhibit a vibrating response resulting from these unvented pressure buildups. Because a potential hazard could arise in which excessive pressure produces a massive discharge of glass, these conditions are undesirable. However, when closely monitoring both indicators, sufficient time is available to take corrective actions to prevent such an event.

Surge Characteristics

Another important aspect of off-gas flow behavior is the magnitude and duration of surges. This information provides a basis for the design of off-gas equipment such as sizing piping, scrubbers, and air dilution rates.

A total of 136 surges of greater than two times the calculated process flow was recorded during the experimental run. Of these, eight surges were >5X in magnitude. A breakdown of the number of surges according to magnitude is presented in Table 13.

TABLE 13

Distribution of Surge Magnitudes

<u>Surge Magnitude</u>	<u>Number</u>	<u>Percent of Total >2X</u>
<2X	136	100
2X-3X	81	60
3X-4X	33	24
4X-5X	14	10
>5X	8	6

Improved resolution of peaks was obtained with a high-speed recorder. The surge typically rises rapidly to the peak approaching a step function, then decays linearly to a point and from that point exponentially to the base line. The exponential decay can be conservatively approximated by a straight line. The percentage of the peak height at which the exponential decay begins varies with surge magnitude. Table 14 lists this percentage as a function of surge magnitude.

TABLE 14

Height of Exponential Decay Portion

<u>Surge Magnitude</u>	<u>Percent of Peak Height</u>
2X-3X	46
3X-4X	43
4X-5X	36

Average surge durations are listed in Table 15. The duration of the linear decay portion generally ranged from nine to 49 sec while the total peak duration including the exponential decay portion ranged from 15 sec to 6.5 min.

TABLE 15**Average Off-Gas Surge Durations**

<u>Surge Magnitude</u>	<u>Duration, min</u>	
	<u>Linear Decay Portion</u>	<u>Total</u>
2X-3X	0.44	2.9
3X-4X	0.35	2.7
4X-5X	0.56	4.1

Data were reviewed to determine a maximum credible surge to recommend for input to Bechtel's dynamic off-gas analysis. Surges of 5X or greater with durations between 30 sec and 1 min at 35% of the peak height, and 3 to 8 min at the base, were observed eight times during the run. Five of these surges were off-scale on the chart. Two surges that occurred when melter operation was not stable were 6X or greater. One of these was off-scale on the chart. Based on these data, the following basis for modeling studies for normal operation (seal pot not to relieve) was recommended.

Maximum Peak Height — 7X process off-gas flow (excluding purges and air inleakage)

Duration at 35% of peak height (2.5X) — 1 min

Duration at base — eight min

**Battelle-Pacific Northwest Laboratory Slurry-Fed Melter Test
June 14-23, 1982**

Good melting was obtained in the first melter test with feed slurry simulating unblended high-aluminum waste from H-Area storage tanks. A melting flux of 10.2 lb/hr-ft² was demonstrated for 115 hr with a 40 wt % solids slurry of Frit 165 and high-aluminum waste. The melt viscosity was 166 poise at 1150°C, the highest for a large slurry-fed melter test.

Introduction

Prior to June 1982 all large melting tests for the DWPF used feed slurry compositions based on blending wastes from the SRP storage tanks. A ten-day test was made in June 1982 to explore the

possibility of feeding the DWPF with unmixed high-aluminum wastes from H-Area storage tanks, and this test is reported herein. The test was made with Frit 165, the new improved frit developed for optimum vitrification of Savannah River defense wastes. The combination of high-aluminum waste and Frit 165 gives an estimated viscosity of 160 poise at 1150°C, the highest viscosity ever tried in a large slurry-fed melter for the DWPF. Vitrification of blended SRP wastes with Frit 165 gives a melt viscosity of 55 poise at 1150°C, and this combination had been used successfully in two large slurry-fed melting tests to demonstrate melting fluxes of 9.6 lb/hr-ft² with plenum heaters and 8.5 lb/hr-ft² without plenum heaters.

Equipment

Two feed systems were used to feed the melter with slurry from three makeup tanks. The test was started using a signal from a mass flowmeter to control the speed of a progressive cavity Moyno® (Moyno Products) pump. This feed system worked well with feed from two of the makeup tanks but plugged with the initial lot of feed taken from the bottom of the third makeup tank. The test was completed using a second feed system. This system used an air-driven double-diaphragm pump to circulate feed slurry in a loop, took the feed stream from the top of the loop, and controlled the feed rate with a signal from a mass flowmeter that adjusted air pressure to an air-operated pinch valve in the slurry loop. The plug in the original feed system occurred in the feed nozzle tube where the bore was 0.37 in. and slurry flow was partially blocked by a 0.125-in. O.D. thermocouple. This restriction was common to both feed systems and would have plugged with either system.

The plug was primarily due to coarse sandy material that probably entered the third makeup tank in one or more of the drums of purchased ferric hydroxide, accumulated in the bottom of the makeup tank, left the tank in the initial lot of feed removed, passed through 1/8-in. holes in the strainer, and entered a 254-gal tank in the first feed system. The coarse sandy material was removed by flushing and discarding the contents of the first feed system.

Other factors in plugging the feed tube were rapid settling of the high-aluminum feed slurry as compared to high-iron content material simulating blended DWPF feed and agglomeration of the fine-frit particles to form hard lumps in the shipping bags. Most, but not all, of the lumps dispersed during feed makeup. It may be necessary to specify dry frit, to ship and store frit in sealed containers, and to break up lumps of frit at the time of addition to the process.

The test melter, the PSCM, is shown in Figure 5 and was the same as used in earlier tests. Three methods were used for controlling electric power input to the glass by the electrodes as shown in Figure 6. Resistance feedback control of voltage was used initially while the slurry feed rate was being increased. Constant current control was used while the slurry feed rate was being held constant to obtain steady state conditions. Constant power control was used when a filled canister was being replaced with an empty canister and the melt level was changing. The control settings for all three methods were changed manually as deemed necessary to maintain the desired glass temperature of 1150°C as measured with thermocouples.

The plenum heaters were used for this experimental run. Temperature feedback control of voltage was used to control power input to the plenum heaters as shown in Figure 7 and maintain the heater sheath temperature maximum at 990 to 1000°C.

An air-atomized water spray was used to cool the off-gas. The water flow rate for the spray was adjusted manually to give off-gas temperatures ranging from 300 to 400°C.

Feed

Feed was an aqueous slurry of Frit 165 and W-Al-F waste sludge as shown in Tables 16 and 17.

The feed was based on 7.13 wt % frit and 28.7 wt % high-aluminum, W-Al,* waste solids were defined on simulated waste compositions for vitrification testing with omission of coal. W-Al waste solids have been changed to W-Al-F waste sludge by converting MnO_2 , CaCO_3 , Ni(OH)_2 , and NaOH to equivalent amounts of $\text{Mn(CHO}_2)_2$, $\text{Ca(CHO}_2)_2$, $\text{Ni(CHO}_2)_2$, and NaCHO_2 . W-Al-F waste sludge is designed to simulate the waste sludge in melter feed if unmixed high-aluminum waste from H-Area storage tanks is fed to the DWPF. Feed was prepared using the material in Table 17 except that MnCO_3 plus the stoichiometric amount of formic acid was substituted for $\text{Mn(CHO}_2)_2$.

Additional formic acid was used to lower the pH to 5.5. The amount of formate in $\text{Mn(CHO}_2)_2$, $\text{Ca(CHO}_2)_2$, $\text{Ni(CHO}_2)_2$, and NaCHO_2 is 1.75 g mol/lb of waste solids as defined on Stage 1 waste simulation. The amount of formic acid used to lower the pH was 0.19 g mol/lb of waste solids. The feed slurry was 40% solids and 60% water. The feed density was measured as 1.37.

* W-Al = high-aluminum waste, washed in hot caustic to remove 75% of the aluminum.

FIGURE 5. Melter Tank N-S Cross Section

FIGURE 6. Melter Power Control System

FIGURE 7. Plenum Heater Temperature Control System

TABLE 16**Composition of Frit 165**

<u>Components</u>	<u>Wt %</u>
SiO ₂	68.0
B ₂ O ₃	10.0
Na ₂ O	13.0
Li ₂ O	7.0
MgO	1.0
ZrO ₂	1.0
<hr/>	
Total	100.0

TABLE 17**Materials for 1000 kg of Glass Product**

2090 liters of aqueous slurry containing

726.2 kg Frit 165 (-200 mesh)
220.5 kg Al₂O₃·3H₂O
54.0 kg Fe(OH)₃
29.8 kg Zeolite (20 to 50 mesh)
13.2 kg SiO₂
55.1 kg Mn(CHO₂)₂
11.6 kg Ni(CHO₂)
6.1 kg Ca(CHO)₂
32.1 kg NaCHO₂
2.0 kg Na₂SO₄
7.4 kg HCOOH to lower pH to 5.5

Small amounts of stable elements were added as shown in Table 18 to obtain data on volatilization of fission products.

TABLE 18

Feed Additives for 1000 kg of Glass

<u>Compound</u>	<u>Weight, g</u>
$\text{Sr}(\text{NO}_3)_2$	420
Sb_2O_5	330
SeO_2	84
$\text{Cd}(\text{NO}_3)_2$	150
TeO_2	100
CsNO_3	730
Rare Earth Carbonates	4180

The glass product composition is given in Table 19.

TABLE 19

Glass Product Composition

<u>Components</u>	<u>Wt %</u>
Al_2O_3	15.0
Fe_2O_3	4.0
MnO	2.7
NiO	0.6
CaO	0.6
SiO_2	52.1
Na_2O	11.0
Na_2SO_4	0.2
B_2O_3	7.3
Li_2O	5.1
MgO	0.7
ZrO_2	0.7
Total	100.0

Results

The first large melting test with simulated high-aluminum waste was completed successfully. Forty-one-hundred gallons of 40 wt % solids slurry of Frit 165 and high-aluminum waste were fed to the PSCM to produce 8.5 tons of glass during a ten-day period. The test was started with the melter containing glass corresponding to vitrification of blended SRP defense wastes with Frit 165. This glass was flushed from the melter during the test by producing twelve melter throughputs of glass.

The melting flux was about 5 lb/hr-ft² on the first day of the test and about 6 lb/hr-ft² on the second day. The melting flux increased to 10 lb/hr-ft² by the 67th hr of the test after feeding slurry equivalent to 2.1 melter throughputs of glass. Increase in melting flux during the first three days of a melting test has been noted previously.

Slurry was fed to the melter at a rate of 76 L/hr for 115 hr to obtain a glass pouring rate of 36 kg/hr and demonstrate a melting flux of 10.2 lb/hr-ft² while providing part of the process heat with silicon carbide heaters in the plenum.

Near the end of the test the maximum melting flux was shown to be less than 12.2 lb/hr-ft² by slow accumulation of an excessive amount of unmelted material on the melt surface during a 40-hr period with a feed rate of 91 L/hr/lb and a glass pouring rate of 43 kg/hr.

The average electrode input was 97 V, 810 amps, and 79 kW, and the electrode temperature was 1025°C while feeding 76 L of slurry per hr. The plenum heater input was 44 kW with the heater thimbles at 900 to 1000°C. The average plenum temperature was 835°C.

The electrical resistivity of the glass was a little higher with high-aluminum waste than it was with blended waste as shown by the increase in melter glass tank resistance from 0.092 ohms in March 1982 to 0.120 ohms in June 1982.

Glass temperatures are given in Table 20 for a thermowell 3 in. above the floor

The cold cap was flexible and had one or two active gas vents during most of the period of feeding 76 L of slurry per hour. The cold cap covered 90 to 100% of the glass surface and the remainder was covered with orange bubbles. The cold cap thickened and became rigid while feeding 91 L of slurry per hour.

The amount of spinel or heavy crystals on the floor of the melting chamber was measured as 1/4- to 1/2-in. both before and after the test.

TABLE 20

Glass Temperatures

<u>Height Above Floor, inches</u>	<u>Temperature, °C</u>
0	1147
3	1152
6	1144
9	1139
12	1156
15	1052
18	846

The off-gas was cooled from 835 to 352°C with 7.5 gal of water per hr. The water was injected counter-current to the off-gas as an air-atomized spray at the entrance to the off-gas line as shown in Figure 5. The water spray was effective in cooling the off-gas and thereby preventing plugging of the off-gas pipe with hard deposits, but three soft deposits formed at the spray nozzle and were removed during the ten-day test. The soft deposits were clusters of particles adhering to the spray nozzle and the pipe wall. The deposits were probably caused by poor atomization of the water spray. The soft deposits were easily removed and dislodged into the plenum with a 1/4-in. rod and an air lance applied manually.

The space above the melt and cold cap appeared free of smoke and fine glass fibers during this test with high-viscosity glass and plenum heaters. Some volatilization of semivolatiles was indicated by the growth of a white feathery structure in the plenum viewport nozzle and white powder deposits on the inside of the viewing windows. The white deposits on the inside of the canister necks were extremely thin and provided noticeably less coverage of the dark surface of the canister interior than in previous tests. This reduction in deposition in canister necks is attributed to the increase in glass viscosity causing less volatilization of sodium, cesium, chlorine, etc. The amount of cesium in the off-gas averaged 0.8% of that in the feed as compared to 8% for previous tests with formate feed.

The weight gain of a HEPA filter on a dry basis was 0.12% of the glass produced when the full flow of melter off-gas passed through the HEPA filter 6-ft downstream and the slurry rate was 76 L/hr. This result compares with filter weight gains of 0.13%

for periods with slurry feed rates of 67 and 89 L/hr during the March 1982 test and a filter weight gain of 0.069% with a slurry feed rate of 49 L/hr during the May 1982 test.

Data are given in Table 21 for the size distribution of solid particles in melter off-gas as determined with a cascade of three cyclone separators followed by an absolute filter. The cutpoint of a cyclone is the particle diameter for which the collection efficiency is 50%.

TABLE 21

Size Distribution of Particles

<u>Cutpoint, micron</u>	<u>Mass %</u>
16	60
6	6
1	23
Filter	31

Dilution of melting off-gas with inleakage air gave H_2 and CO concentrations below lower explosive limits even without oxidation as shown in Table 22. Most of the H_2 and CO was oxidized in the heated plenum as reported previously.

TABLE 22

Dry Off-Gas Composition, Volume %

<u>Component</u>	<u>Measured</u>	<u>Calculated Without Oxidation</u>	<u>Calculated With Oxidation</u>
H_2	0 to 0.5	2.5	0
CO	0 to 0.75	2.5	0
CO ₂	2 to 9.8	2.5	5.3
O ₂	16 to 20	19.4	17.8
A	-	0.9	0.9
N	73 to 77	72.2	76.0

The calculated off-gas concentrations are based on the melter feed composition from Table 17, the glass production rate of 36 kg/hr, and 16 scfm of noncondensable off-gas flow as measured with a Pitot tube downstream of two venturi scrubbers. The calculations are based on decomposition of 2 mols of formate to yield 1 mol each of H_2 , CO, and CO_2 .

The measured concentrations of H_2 and CO show increases during off-gas surges and indicate the possibility of exceeding lower explosive limits if oxidation is insufficient. The measured and calculated concentrations in Table 22 do not allow for off-gas surges. The largest off-gas surge during the June 1982 test was three times the normal flow, and the surge duration was five minutes from normal flow to peak flow and back to normal flow. During this surge the CO concentration went from a level of around 0.2% to a value above 0.5%, full scale of the meter; the indicated H_2 concentration went from less than 0.5% to 3.5%. Measurement of H_2 concentration is based on thermal conductivity, and the indicated value may be higher than the true value during a surge.

A small amount of molten salt was observed on the surface of the glass in the PSCM after the test. The molten salt was sampled and analyzed. The analyses are given in Table 23.

TABLE 23

Molten Salt Composition

<u>Component</u>	<u>Wt %</u>
SO_4	69
Cl	0.6
Ca	0.4
Li	3.8 to 9.8
Na	21 to 29

These data show that the molten salt was mostly Na_2SO_4 and Li_2SO_4 .

Analyses of the sulfur content of feed and glass samples indicate that the feed contained 3.8 kg Na_2SO_4 per 1000 kg of glass and the glass contained less than 0.9 kg Na_2SO_4 per 1000 kg of glass product. These numbers show that the feed contained almost twice the specified amount of Na_2SO_4 and that the glass contained less than 25% of the Na_2SO_4 fed to the melter.

The measured concentration of SO_2 in dry melter off-gas ranged from five to 390 ppm as shown in Figure 8. It was low on the first day of the test and increased until the third day.

Material balance calculations show that the amount of SO_2 in the off-gas corresponded to a yield from sulfate in the melter feed slurry of less than 2% on the first day and increased to 25% on the third day. The increase in SO_2 in off-gas indicates that more sulfate is being reduced and matches the increase in melting flux during the first three days of the test. The combination of sulfates and reducing agents increases melting rates in industrial soda-lime glass melting furnaces.

Glass pouring and canister filling were good with the high-viscosity glass product. Glass pour densities were calculated based on canister filling heights and canister weights before and after filling. The first canister had a glass pour density of 119 lb/ft^3 with a trough temperature of 1050°C and a face temperature of 900°C . The remaining canisters were filled with the temperature increased to 1100°C for the trough and 1000°C for the face. The glass pour density was 154 lb/ft^3 for the third and fourth canisters and 157 lb/ft^3 for the second and fifth canisters.

Evacuated Canister Tests with the Slurry-Fed Melter*

The removal of molten glass from a melter by use of an evacuated canister is an especially attractive melter drain technique successfully used on several occasions to drain glass from the small-cylindrical melter (SCM). Difficulties in applying the technique successfully on the calcium-fed melter (CFM)** provided impetus for development of a suitable evacuated canister theory which could be used to predict canister performance and thus guide evacuated canister design.

Two evacuated canister tests have been conducted using the slurry-fed melter (SFM) and Frit 131 + TDS-3A simulated waste glass. The tests qualitatively confirm theoretical predictions and demonstrate that the evacuated canister technique could be successfully used to drain a glass melter in the Defense Waste Processing Facility.

Design of the Evacuated Canister

The evacuated canister used for both of the tests is shown in Figure 9. The sides of the canister were rolled from 1/4-in.-thick 304L-SS, and the top and bottom each cut from 1/2-in.-thick 304L-SS.

* Previously known as the LSFM or JHRM.

** Previously known as the 1941 melter.

FIGURE 8. SO₂ Concentration in Melter Off-Gas

FIGURE 9. Side View of Evacuated Canister

The canister was designed to resist an implosive force of 14.7 psia at the temperature of the incoming glass stream, i.e., 1100 to 1150°C. The sides of the canister were insulated with a 1-in.-thick layer of Kaowool® Bulk Fiber (Babcock & Wilcox Co.); and the top and bottom each insulated with a 2-in.-thick layer of Kaowool® M-Board. This insulation served to keep the carbon steel secondary container thermally cool and structurally strong throughout the duration of each test. Lifting lugs for hoisting the entire apparatus up to the top of the melter were welded to the secondary container.

Evacuated canister theory indicated that a reasonable choice for the suction pipe leading from the glass pool to the evacuated canister was 3-in. to 4-in. pipe. The suction pipe was thus made from 4-in. schedule 40 pipe (304L-SS), with a suitable flange attached (Figure 9) to control the depth to which the suction pipe could be inserted into the glass pool (approximately 11 in.). The suction pipe design required that the glass be sucked upward 6 ft from the submerged end, transported horizontally 12 ft, and then downward 2-1/2 ft before being discharged into the evacuated canister. No supplementary heating of the suction pipe was used to prevent premature solidification of glass as it flowed to the canister, since theoretical predictions indicated that the initial glass surge would reach the canister in approximately one second after melting of the aluminum plug.

A 1-in.-thick layer of Kaowool® Blanket was wrapped around the entire portion of the suction pipe that was external to the melter to reduce heat losses from the glass stream as it flowed through the suction pipe. No insulating blanket was used on the portion of the suction pipe that was inserted in the melter plenum because of warnings by the vendor that the organic binders in the blanket may flash in such a situation.

A threaded aluminum plug was screwed into a pipe coupling on the end of the suction pipe (Figure 9) to provide the necessary seal of the canister prior to evacuation.

Prior to each test, the evacuated canister was pumped down to a vacuum of 30-in. Hg as indicated on a 0- to 30-in. Hg vacuum gauge. No special precautions were taken to clean the interior of the suction pipe or evacuated canister before the test.

Evacuated Canister Test, 4/6/82

The canister was pumped down to 30-in. Hg vacuum before the test, and then valved off and disconnected from the vacuum pump. The vacuum in the canister dropped to 28- to 29-in. Hg at this time. Ten to fifteen minutes later when insertion of the suction pipe into the glass pool began, the vacuum had dropped to 24- to 25-in. Hg.

The aluminum plug in the submerged end of the suction pipe melted in approximately 6 min after insertion into the glass pool, which compares well with the calculated value of 3 to 7 min. The glass level in the melter dropped 4 to 6 in. in 6 to 8 sec after the plug melted, indicating that 550 to 800 lb of glass had been removed from the melter. Glass flow then stopped, and the vacuum gauge on the canister read 10- to 12-in. Hg. This was disappointing since approximately 1450 lb of glass* should have been sucked into the canister.

The maximum amount of glass that could be contained in the suction pipe is 268 lb, indicating that at least 280 to 530 lb of glass were discharged into the canister, and that premature glass solidification in the suction pipe was not responsible for stopping the glass flow.

The vacuum required to lift the glass the required minimum of 5 ft above the melt pool is 10.4-in. Hg. Unless the absolute pressure is lower than this (higher vacuum), flow will stop, which is what occurred. Thus, it appears that glass flow stopped prematurely because air leakage into the canister caused a subsequent rise in canister pressure as the air was both heated and compressed by the incoming glass stream.

Evacuated Canister Test, 6/9/82

No changes were made in the design of the evacuated canister, suction pipe, aluminum plug, or secondary canister. However, it was suspected that the leak in the first test probably occurred around the threaded aluminum plug. This time the joint between the plug and the pipe coupling (Figure 9) was coated with epoxy glue before the canister was evacuated. Air inleakage caused less than 1-in. Hg drop in vacuum when the canister was valved off from the vacuum pump and allowed to sit overnight. Just prior to the test however, the canister vacuum was pumped back to 30-in. Hg. Also, the evacuated canister and its auxiliary components were weighed on two separate scales (Table 24).

The evacuated canister was then hoisted to the top of the melter and the suction pipe inserted into the melt pool. A stop watch was used to measure the time for the aluminum plug to melt.

The initiation of glass flow was clearly evident by a slight jump in the suction pipe which corresponds to a conversion of momentum to force as the initial glass surge hit the first turn in the suction pipe. The termination of glass flow was marked by a

* Based on a glass density of 2.4 g/cm³.

TABLE 24

Data for Evacuated Canister Test 2

1. Weight of glass removed from melter:

	<u>Scale 1</u>	<u>Scale 2</u>
Weight of full canister, lb	4800	4900
Weight of empty canister, lb	2800	2900
Weight of glass, lb	2000	2000

2. Nominal glass temperature at start of test: 1100 to 1150°C.

3. Time to melt aluminum plug: 5 min, 42 sec.

4. Time for glass to flow into canister: 45 sec.

5. Evacuated canister vacuum:

Start of test - 30 in. Hg

End of test - 0 in. Hg

6. Volume of glass removed from melter: 10.5 ft³.

7. Volume of glass in canister immediately after test: 10.2 ft³.

8. Apparent glass density: 193 lb_m/ft³ or 3.10 g/cc.

9. Steady-state glass flow rate in suction pipe:

Measured - 44.4 lb_m/sec

Theoretical - 93.2 lb_m/sec @ 9 poise and 3.10 g/cc

- 48.5 lb_m/sec @ 17.3 poise and 3.10 g/cc

- 38.3 lb_m/sec @ 17.3 poise and 2.44 g/cc

sucking noise and observable agitation in the melt pool around the suction pipe, at which time the evacuated canister vacuum dropped to 0-in. Hg. Total elapsed time for glass flow was measured at 45 sec.

The volume of glass removed from the melter was calculated from the height of the glass film remaining on the bottom of the suction pipe after the test and the cross-sectional area of the glass pool corrected for the riser and electrodes.

After the test, the volume of glass sucked into the evacuated canister was calculated from the depth of the glass in the canister as measured immediately after the test and the cross-sectional area of the canister. The glass density was calculated to be 3.10 g/cm^3 at 1100 to 1150°C which is surprisingly high compared to the TDS value of 2.1 g/cm^3 and the preliminary value of 2.44 g/cm^3 for Frit 131 + TDS waste.

Using the TDS glass viscosity of 9 poise at approximately 1125°C and the measured density of 3.10 g/cm^3 , the theoretical steady-state mass flow rate of glass into the canister was calculated to be $93.2 \text{ lb}_m/\text{sec}$. This compares poorly with the measured flow rate of $44.4 \text{ lb}_m/\text{sec}$ (Table 24). However, using a glass viscosity of 17.3 poise for Frit 131 + TDS waste and again a density of 3.1 g/cm^3 , the theoretical mass flow rate was calculated to be $48.5 \text{ lb}_m/\text{sec}$, which differs from the measured flow rate by only + 9.2%. Using values of 2.44 g/cm^3 for density, 17.3 poise for viscosity, the theoretical mass flow rate was calculated to be 38.3 ft/sec , which differs from the measured flow rate by -14%.

The discrepancies in the glass density and viscosity data make it difficult to quantitatively assess the accuracy of the theoretical equations previously developed for predicting the performance of evacuated canisters. Glass samples will be removed from the evacuated canister in the near future to remeasure glass density and viscosity as well as check for errors by the vendor in chemical composition of the glass.

Qualitatively however, theoretical predictions regarding higher glass flow rates in the suction pipe with increasing pipe radius appear to be substantiated since earlier tests on the CFM with approximately 1-in.-OD suction pipes failed because of premature glass solidification in the suction pipe.

Conclusions

- One or more evacuated canisters could be used to drain a glass melter in the DWPF. Since the reference drain technique is currently a bottom freeze valve, the evacuated canister technique should be regarded as a backup or a potential replacement for the reference drain technique.
- Insulation of that portion of the suction pipe inserted into the melter plenum is no longer recommended because of possible flashing of the insulation.
- Supplementary heating of the suction pipe is not required to prevent premature glass solidification if the diameter of the suction pipe is large enough (4 in. is adequate for Frit 131 + TDS-3A simulated waste glass).
- A high vacuum in the evacuated canister when glass flows up the suction pipe begins is mandatory.
- Evacuated canister theory appears to be qualitatively substantiated, but must await resolution of sharp differences in glass property data before it can be quantitatively evaluated.

Argon Purging of Thermowells

The Defense Waste Processing Facility (DWPF) melter is designed to provide a two-year life time in the canyon. Any equipment replacement must be done remotely. Therefore, it is important to minimize the number of routine maintenance items. Thermocouples are items that will have to be routinely replaced. To prevent or reduce the failure incidence of a thermocouple because of oxidation of the metal sheath, the thermowell could be purged with argon. Tests were conducted on the small-cylindrical melter-1 (SCM-1) which demonstrated the feasibility of argon purging.

Experimental Procedures

Figure 10 shows the design of the thermowell used for the tests. The portion of the thermowell in contact with the glass and glass vapors was made from 3/4-in. schedule 40 Inconel® (Huntington Alloys, Inc.) 690 pipe. All Type K thermocouples used had Inconel® 600 sheaths. The purge guide for introducing the argon at the bottom of the thermowell was made from 1/8-in. schedule 80 304L-SS pipe. A Manostat flowmeter was used to regulate the flow of argon into the thermowell.

FIGURE 10. Thermowell Design

For the argon purged thermowell, argon was bled in through the purge guide at a rate of 29 mL/min. The argon did not cool the thermocouples. Each thermocouple and thermowell outside diameter was measured with a micrometer before and after the test for corrosion rate determination. The melter idled for 36% of the time. Total time in the melt pool was 22 days.

For the control test, the same thermowell design was used, only argon purging was eliminated. This test lasted 25 days and the melter idled for 64% of the time. The melter was fed with Frit 131 and TDS simulated waste slurry.

Results

Because of the short testing time and the differences in melter idle time, no comparison between thermowells concerning the true corrosion rate can be made. The diameters of the thermowells before and after testing are shown in Table 25. The average corrosion rate of the portion of thermowell in contact with the glass was 0.20 mils radial/day for the purge and 0.28 mils radial/day for the control. No changes in the diameter of the thermocouples could be measured.

There was a notable difference between the inside surface on the thermowells, thermocouples, and the stainless steel purge guides. These items are shown in Figure 11 in the same relative position as in the molten glass. The inside of the control thermowell was blackened by oxidation and pitted, whereas the purged thermowell remained clean and free of pits. These observations are also true for the thermocouples. Argon purging preserved the 304L stainless steel purge guide. The pipe in the control thermowell was completely oxidized below the melt line. Because of the inert atmosphere and high temperatures ($\approx 1150^{\circ}\text{C}$), one of the thermocouples in the argon purged thermowell fused to the stainless steel pipe, but this did not affect the accuracy of the thermocouple.

Conclusion

Argon purging appears to be useful in prolonging the life of Inconel® 600 thermocouple sheaths. More testing is required to determine if argon purging helps to prevent drifting of the thermocouples and to better determine corrosion rates of Inconel® 690 thermowells. With the small cylindrical melter-2 (SCM-2) it is possible to test at least two thermowells simultaneously. Each thermowell and thermocouple will be subjected to the same simulated waste glass, idle time, and other melt pool/melt plenum conditions.

FIGURE 11. Section Views of Thermowells

TABLE 25**Thermowell Diameter Measurements**

<u>Distance from Bottom, in.</u>	<u>Diameter Before, in.</u>	<u>Purged Diameter After, in.</u>	<u>Nonpurged Diameter After, in.</u>
1	1.050	1.037	1.035
2	1.050	1.041	1.035
3	1.050	1.042	1.035
4	1.050	1.041	1.035
5	1.050	1.041	1.035
6	1.050	1.042	1.035
7	1.050	1.046	1.042
8	1.050	1.050	1.050
9	1.050	1.050	1.050
10	1.050	1.050	1.050
11	1.050	1.050	1.050
12	1.050	1.050	1.050

Days in service: Purged 22
Nonpurged 25

Average corrosion rate: Purged 0.20 mils/day
Control 0.28 mils/day

The platinum-rhodium (Pt/Rh) thermocouples proposed for use in the DWPF will be tested in the SCM-2. Four Pt/Rh thermocouples with Inconel® 600 sheaths and two Pt/Rh thermocouples with platinum sheaths will be placed in purged and control thermowells to determine the possible life expectancy of these thermocouples.

**Performance Testing of a Single Stage Hydro-Sonic®
(Hydro-Sonic Systems) Scrubber**

Operation of continuous slurry fed melters has demonstrated that off-gas aerosols are generated by entrainment of feed slurry and vaporization of volatile species from the melt. As the slurry feed contacts molten glass various heat transfer regimes occur at this interface encompassing both nucleate and film type boiling. The results of this vigorous boiling activity is the dispersion of a portion of the feed into the melter plenum vapor space. The dispersed material is then transported from the melter with the bulk off-gas flow. This emission mechanism is referred to as entrainment.

Elements that volatilize from the melt originate in both the waste sludge and the glass-forming frit. As vaporization from the melt occurs, volatile material is transported to cooler regions of the melter vapor space by the bulk movement of off-gas. Cooling of the off-gas causes the size distribution of semivolatile species to undergo a shift towards increasing particle diameters. The primary mechanisms responsible for this type of particle growth are nucleation and condensation effects which can be estimated based on thermodynamic principles.¹

Measurements of the size distribution of melter off-gas aerosols have verified the existence of a bimodal size distribution that differs in chemical composition between modes. Table 26 is a summary of size distribution data obtained from tests conducted at the PNL with the pilot-scale ceramic melter (PSCM) processing a mixture of simulated defense waste sludge and frit.

Size distribution data presented in Table 26 were obtained with a series cyclone particulate sampling train located in the melter off-gas exhaust line. The melter aerosol is composed of two fractions -- a coarse fraction (entrainment) representative of the feed and a fine fraction representative of semivolatile emissions. As expected, the magnitude of the coarse fraction (entrainment) is related to the glass production rate. This trend was evidenced by the increase in mass fractions collected in the first stages of the cyclone sampler (for particles greater than 1 μm) as the plenum temperature was increased.

TABLE 26

Size Distribution of Melter Aerosols²

Test	Size Distribution, wt %				Glass Production Rate, kg/hr-m ²	Plenum Temp, °C	Frit
	16 μm	6 μm	1 μm	<1 μm			
PSCM-4	-	0.9	3.6	95.6	39	400	131
PSCM-5	13.5	3.9	20.2	62.2	50	520	131
PSCM-6	46.1	0.5	12.7	40.7	60	850	165
PSCM-7	7.5	3.2	9.4	79.9	43	300	165

Measurement of the size distribution of particles leaving the off-gas quencher has shown that the distribution is no longer bimodal. The majority of the aerosol is typically less than 1 μm in diameter and composed almost entirely of semivolatile material. Figure 12 is a summary of measured size distribution data of particles leaving the off-gas quencher. As evidenced by the data depicted in Figure 12, essentially all material (usually >95%) was less than 1 μm (aerodynamic diameter). This demonstrates the enhanced scrubbing that occurs during quenching. The effect of condensing steam for improving particulate scrubbing performance is an established method for scrubbing and is used in several commercial installations.³

The effects of steam condensation are that (1) the actual size of the off-gas particles are increased thus improving inertial capture and (2) temperature and vapor pressure gradients effectively increase deposition of particles.

Selection of Control Equipment

Knowledge of the size spectrum of melter off-gas aerosols is essential for the appropriate design engineering of wet scrubbing process equipment. In addition the design restrictions imposed by requirements for remote maintenance and canyon operation eliminates several control equipment options. Semivolatile submicron emissions generally are of the greatest radiological concern. Scrubber selection therefore should be limited to devices with high decontamination capability for submicron aerosols.

A review of existing particulate control devices which have a high removal efficiency for submicron particles are compared in Figure 13. Unfortunately, the majority of these devices are not suited for remote operation or radioactive off-gas processing service. As evidenced by the data presented in Figure 13, the Hydro-Sonic® scrubber is capable of attaining exceptionally high removal efficiencies for submicron sized particles.

Principle of Operation

The Hydro-Sonic® scrubber is a compressible fluid aspirated wet scrubber that has been proven to be extremely effective for reducing the concentration of submicron particles compared to other more complex devices.⁴ Particulate scrubbing is accomplished by injecting a liquid stream into the wake region of an expanding supersonic free jet. Rapid acceleration of the liquid phase results in complete atomization. Fragmentation and acceleration of the liquid phase and the high relative velocity created between the expanding free jet flow and the entrained off-gas flow results in

FIGURE 12. Aerosol Distribution of Quenched Melter Off-Gas

FIGURE 13. Comparison of Wet Scrubber Performance

efficient particulate scrubbing. Scrubber particles are then effectively grown by agglomeration and separated from the gas phase by a cyclone separator.

Test Description

Scrubber performance tests were structured to evaluate the dependence of scrubber collection efficiency on various process variables. Specific objectives of these tests were to evaluate the effect of ejector water flow rate, ejector air flow rate, and particle size distribution on particle capture capability. The test configuration is shown schematically in Figure 14.

Selection of aerosol generating equipment was based on the requirement for a bimodal particle size distribution representative of actual melter off-gas. A coarse aerosol, representing entrainment, was obtained with a Syntron® (FMC Corporation) Model FC-T01 vibrating dust feeder using Chemaloy® (Ashland Oil, Inc.) Speculite ferric oxide. The feed rate was 1.26 g/sec. A submicron sized aerosol, representing semivolatile emissions, was provided by a hot particle generator (HPG). Submicron sized particles were produced by feeding an atomized solution of ferric nitrate, $\text{Fe}(\text{NO}_3)_3$, to the propane fired HPG. The solution was sprayed into the core of the flame where oxidation occurred. Exhaust streams from both aerosol generators were mixed and transported to the scrubber by the primary gas flow. The primary gas flow was provided by a pressure blower and saturated in an air washing column to simulate saturated melter off-gas. The gas delivery rate ranged from 2.8 to $5.7 \text{ m}^3/\text{m}^3/\text{min}$.

Aerosol distribution and concentration data were verified using cascade impactors and total filters at the scrubber inlet and discharge. A Brink BMS-11 cascade impactor was used to characterize the scrubber inlet distribution and a University of Washington Mark V source test cascade impactor was used at the scrubber discharge. Backup filter media for both impactors were Gelman Type A glass fiber filters. Figure 15 is a comparison between the size distribution of quenched melter off-gas and the simulated melter aerosol distribution employed for the scrubber evaluation.

Results and Discussion

Scrubber performance tests were conducted for inlet process gas temperatures of 35 to 50°C and aerosol loadings ranging from 3.4 to $794 \text{ mg}/\text{m}^3$ dry gas. Scrubber water injection rates were maintained in excess of the design condition of 0.35 kg water/kg process gas based on previous performance tests. Removal efficiencies expressed as decontamination factor, DF, (where $\text{DF} = \text{mg aerosol injected}/\text{mg aerosol discharged}$) are summarized in Table 27.

FIGURE 14. Schematic of Test Configuration Used for Scrubber Evaluation

FIGURE 15. Comparison Between Scrubber Test Aerosols

TABLE 27

Performance Test Results of Model 20 AE-H Hydro-Sonic® Scrubber

Particle Concentration gr/dscf*		Ejector Air Process Air	Ejector Water Process Air	Removal Efficiency, wt %	DF
Inlet	Outlet	kg/kg	kg/kg		
0.0094**	0.000320	0.28	0.52	96.60	20
0.0025**	0.000120	0.28	0.52	95.2	21
0.0118**	0.000470	0.28	0.52	96.02	25
0.2245†	0.000930	0.58	1.02	99.59	241
0.2642	0.000760	0.58	2.03	99.71	347
0.2778†	0.000400	0.58	3.18	99.86	694
0.2913††	0.000510	0.54	0.77	99.82	571
0.3468†	0.001200	0.47	2.57	99.65	289
0.1072†	0.000550	0.57	2.44	99.99	195
0.0015**	0.000090	0.57	2.44	94.0	17

* gr/dscf = grains/dry standard cubic foot

** HPG aerosol generator only

† HPG aerosol generator and coarse dust feeder

†† Coarse dust feeder only

For tests conducted with both aerosol generators operating the gas stream decontamination, DF, increased as water injection rate and separator swirl velocity were increased. Separator swirl velocity was controlled by an adjustable vane located at the inlet to the separator. Gas stream decontamination was effectively increased from a DF of 250 to 694 for roughly a threefold increase in water injection rate. Further increases in water injection rate could not be performed because of a limitation in the water supply pressure as shown in Figure 16. The effect of increased water injection is presently being evaluated on the SCM-2 installation.

The effect of increased water injection is not surprising. Calculations indicate that with particles 3 to 5 μ m and larger, particle size has a minimal effect on the measured and theoretical

FIGURE 16. Variation of Scrubber Decontamination Performance with Water Injection Rate

scrubbing efficiency. Increased water injection serves to improve overall collection efficiency by improving coverage. Since the transverse shear imposed by the jet on the fluid is constant, increasing the water rate results in a deeper penetration of the jet by the liquid phase, thereby improving coverage. Increased coverage results in improved scrubber efficiency — an effect that has been studied extensively.⁵

Increased air injection rate did not result in improved collection efficiency. This effect was attributed to the performance aspects of the supersonic free jet. In fact, high air injection rates should decrease efficiency since re-entrainment losses are increased, and operation of the supersonic nozzle off of design conditions (over expanded shock) results in poor velocity recovery.

Mass overall DF values were lowest when the HPG aerosol generator was used. The aerosol produced by this device is predominately submicron. The measured inlet particle size distribution for the HPG aerosol generator and the HPG aerosol generator and dust feeder are shown in Figure 15. Measured fractional collection efficiencies showed that the most penetrating particle size was less than 0.1 μm aerodynamic diameter. This was verified by theoretical calculations which show that the separation for particles in this size range is controlled by diffusional effects. A comparison between measured data from these tests and data from Reference 6 are shown in Figure 17.

FIGURE 17. Comparison Between Air and Steam Ejector Driven Scrubber Performance

Scrubber pressure drop was found to have been low compared to other types of high efficiency venturi scrubbers. A summary of measured overall scrubber pressure drop as a function of operating conditions is presented in Table 28.

TABLE 28

Summary of Scrubber Air Consumption and Pressure Drop

Ejector Air		Ejector Water, gpm	Process Air, scfm	Mixing Tube	Cyclone
psig	scfm			Pressure Drop, in. H ₂ O	Pressure Drop, in. H ₂ O
54	62	1.05	219	6.4	1.1
58	66	1.05	223	7.8	1.4
58	66	1.05	114	6.9	0.8
60	67	1.05	137	5.7	7.3
60	67	2.10	137	5.1	8.5
60	67	3.15	137	3.8*	8.0
72	81	1.30	170	4.9	9.7
60	67	3.15	173	10.4	0.0
82	87	3.15	164	7.9	6.8
82	87	3.15	164	7.9	6.8

* Inconsistent with other data, probably caused by a water slug in the manometer line.

High fractional removal efficiency for particles in the 0.1 to 1.0 μm range is essential for controlling the semivolatile emission. All types of gas-solids separations equipment experience a degradation in removal efficiency for submicron or near micron sized particles. This phenomena occurs because two or more separation mechanisms are operating yet none is dominating. This particle size range is where the particle penetration through the collection device is a maximum and the efficiency of the device a minimum. The minimum fractional collection efficiency of the Hydro-Sonic® scrubber could not be determined because the last cut stage of the cascade impactor was roughly 0.1 μm . As a practical matter this is the limit for all inertial separating devices. Figure 18 depicts the fraction collection efficiency performance of a single-stage Hydro-Sonic® scrubber compared to a single and two-stage 2-in.-thick WDBF. These data demonstrate that the most penetrating particle size for the WDBF is roughly 0.2 μm . This is not unexpected though since this device operates in the diffusional

FIGURE 18. Comparison of Melter Off-Gas Wet Scrubbers

regime. However, when the units are staged, a slight change in removal efficiency occurs (overall DF increases from 25 to 50, this means the second-stage DF is 2). A single-stage Hydro-Sonic® scrubber is roughly an order of magnitude better than the two-stage WDBF.

The measured mass overall removal efficiencies of the Hydro-Sonic® scrubber and WDBF were comparable. However, the degradation in removal efficiency for the Hydro-Sonic® scrubber occurs in a size range that can be controlled by conventional wet scrubbing equipment (i.e., particles $<0.1 \mu\text{m}$ diameter).

Conclusions

- The Hydro-Sonic® scrubber is a viable replacement for the primary WDBF.
- The fractional decontamination performance of a single-stage Hydro-Sonic® scrubber for submicron particles between 0.1 to $1.0 \mu\text{m}$ is roughly an order of magnitude better than a two-stage WDBF.
- The water consumption of a single-stage Hydro-Sonic® scrubber is 1/3 the requirement of a similarly sized WDBF.
- Scrubber operation is passive and not dependent on the primary off-gas flow rate.
- Air is not as effective as steam for attaining increased DF performance.
- Scrubber design is inherently simple and applicable for canyon operation.

The Effects of Lead on the Chemical Durability of SRP Waste Glass

As part of the overall effort for assessing the chemical durability of SRP waste glass forms, a variety of leaching tests are being performed⁷ under anticipated repository conditions. It is important to determine the performance of SRP waste glass as part of the repository system to develop cost effective waste packages. For example, we have already observed that leaching SRP waste glass in the presence of Type 304L stainless steel or Ticode-12, potential canister, or overpack materials, does not significantly affect waste glass durability. We have also noted that certain bentonite backfill mixtures can either accelerate or inhibit glass corrosion depending on clay composition. In addition,

leaching in simulated groundwaters and in the presence of host rock (including salt, basalt, shale, granite, and tuff) has generally reduced the amount of material leached from the glass, relative to deionized water.⁷

One of the more interesting effects that has been observed over the past year has been the effect of lead metal on the leaching of 76-68 commercial waste glass.⁸ The objective of the present work is to compile and interpret three separate studies performed at Savannah River Plant evaluating the effects of lead on leaching of SRP waste glass. In these studies lead was introduced in three different ways, using techniques that might have practical relevance to long-term waste immobilization. For example, lead was introduced into the glass by doping glass compositions with lead oxide during fabrication. Lead was also introduced separate from the glass by two methods; first, by doping leachants as may occur by intrusion of groundwaters picking up lead ions from a package backfill and second, by leaching glass in the presence of lead metal or lead-clad canister or overpack components. This work will ultimately be used with other studies currently in progress to better understand the "lead phenomenon" and possible applicability for optimizing performance of SRP waste glass during permanent storage.

The following summarizes the findings:

Summary

- The presence of lead metal during the leaching of SRP waste glass significantly improves the durability of the glass. This occurs even though the final pH of the solution rises to relatively high values.
- The presence of lead ions in a buffered leachant during the leaching of SRP waste glass generally improves the durability of the glass, but only slightly and not as much as the presence of lead metal.
- There is no significant difference in the durability of SRP waste glass for compositions containing no lead compared to compositions containing up to 4 wt % lead oxide.
- The reason that glass durability is improved for glasses leached in the presence of external lead is because of a lead-enriched surface layer that adsorbs on the leached glass surface.

Experimental Procedures

Waste Glass/Leachant Preparation, Leaching Conditions and QA

Waste glass compositions were made from approximately 70 wt % Frit 131 and 30 wt % TDS waste. Some compositions were doped with cesium and strontium while others were doped with 0 to 4 wt % lead oxide. The exact content of lead and other species contained within the glass after melting is given in Table 29. All glass compositions were melted at 1150°C for at least two hours before being cast into graphite molds. The solidified waste glass forms were then annealed, core drilled and sliced to a surface area of 400 mm², and each surface finished to 600 grit.

The volume of leachant used in each study was 40 mL to give a constant surface area of glass to volume of leachant ratio (SA/V) of 0.1 cm⁻¹. In most cases, deionized water was the leachant, but in some tests a solution buffered to pH 7 was doped with lead and used as leachant.

The three individual studies performed evaluating the effects of lead on leachability all used standard MCC-1 leaching procedures.⁹ This included performing leach tests at 90°C in triplicate using a SA/V ratio of 0.1 cm⁻¹. A standard glass composition was also leached as a reference case with no lead involved. Because of the different objectives of these studies, there were small variations in waste glass composition and leaching conditions as summarized in Tables 29 and 30 and described below:

- Reference Glass — No Lead Present

A reference composition of SRP 131/TDS waste glass was leached for 28 days using MCC-1 leaching procedures. Deionized water was used as the leachant.

- Lead-Doped Glass

SRP 131/TDS waste glass was doped with 1, 2, 3, and 4 wt % lead oxide. The lead-doped glasses were leached in the same manner as the reference glass.

- Lead-Doped Leachant

SRP 131/TDS waste glass was doped with 0.1 wt % nonradioactive Cs and Sr. These glasses were leached in solutions containing lead ions. To keep the pH of the solution constant to better evaluate the effects of lead ions alone, a pH 7 buffer was used as leachant in addition to a deionized water leachant as standard. The buffered leachant was composed of 0.1M tris

TABLE 29

Major Components of Waste Glass Compositions (131/TDS) Used in Lead Studies

Waste Glass Composition, wt %

	<u>SiO₂</u>	<u>B₂O₃</u>	<u>Na₂O</u>	<u>Al₂O₃</u>	<u>Fe₂O₃</u>	<u>MnO₂</u>	<u>MgO</u>	<u>Cs₂NO₃</u>	<u>SrNO₃</u>	<u>f_i* Pb</u>
I Reference glass (no lead)	46.8	9.40	12.57	8.76	12.45	3.08	0	0	0	0.0009
II Pb - Doped glass										
1 wt %	45.6	9.27	12.38	8.88	12.79	3.49	0	0	0	0.0076
2 wt %	45.3	9.07	11.79	9.66	12.58	3.22	0	0	0	0.0103
3 wt %	43.1	8.98	11.09	8.12	13.10	3.23	0	0	0	0.0118
4 wt %	42.2	8.68	11.04	7.61	12.93	3.17	0	0	0	0.0108
III Pb - Doped leachant	42.6	9.93	9.70	4.12	10.27	3.72	1.37	0.10	0.10	0
IV Glass/Metal Pb	45.8	10.84	9.57	5.43	13.20	4.06	1.41	0.10	0.12	0

* f_i - fraction of Pb in glass.

TABLE 30

Leach Rates and Leaching Conditions Used in Lead Studies*

Leach Rates, g/m²-d

	Mass Loss	Si	B	Na	Al	Fe	Mn	Mg	Net Lead in Solution, ppm	Leaching Conditions			
										Leachant	pH	pH	Time
I Reference glass (no lead)	0.40 0.12	0.53 0.13	0.64 0.42	0.60 0.43	0.56 0.00	0.02 0.00	0.02 0.38	0.00 0.19	0.028	DI water Tris Buffered Solution	4.95 6.99	9.57 6.96	28 d. 28 d
II Pb - Doped glass													
1 wt %	0.38	0.54	0.66	0.62	0.49	0.02	0.02	0.00	0.038	DI water	4.95	9.47	28 d.
2 wt %	0.38	0.46	0.51	0.49	0.49	0.04	0.05	0.00	0.177			9.41	
3 wt %	0.37	0.53	0.64	0.76	0.54	0.01	0.02	0.00	0.055			9.54	
4 wt %	0.36	0.52	0.63	0.53	0.53	0.01	0.02	0.00	0.30			9.53	
III Pb - Doped leachant	0.07	0.14	0.14	**	0.05	0.07	0.14	0.16	55	Tris Buffered Solution	7.05	5.2	28 d.
IV Glass/Metal Pb**	0.02	0.01	0.15	0.02	0.01	0.01	0.00	0.37	>1000	DI water	4.50	10.2	42 d.

* All tests run in triplicate using modified MCC-1 test procedures. Teflon® (Du Pont) leach vessels, 90°C and (SA)_{glass}/(V)_{leachant} = 0.1 cm⁻¹.

** Accurate Na determinations due to glass leaching could not be made because of the high initial Na concentration of the lead-doped Tris solution.

(hydroxymethyl) aminomethane and 0.1M hydrochloric acid containing a 0.1M concentration of lead nitrate. Leaching tests were conducted using MCC-1 procedures and lasted 28 days.

- Glass/Metallic Lead Interactions

SRP 131/TDS waste glass was again doped with approximately 0.1 wt % of nonradioactive Cs and Sr. MCC-1 leaching procedures were used. However, leaching tests were conducted in the presence of approximately 3 g of lead foil with a relative surface area of about 663 mm². Tests were continued for 42 days, using deionized water as leachant.

Analyses

As prepared waste glass solids were dissolved and analyzed by inductively coupled plasma spectroscopy (ICP) and atomic absorption (AA). Leachant and leachate solutions were analyzed by these same tools with Si, B, Na, Fe, Mn, Mg, Pb, and Sr concentrations determined by ICP and Cs by AA. Scanning electron microscope, electron microprobe analysis, and Auger spectroscopy were used to study surfaces of selected leached glasses.

Results and Discussion

- Leachability

The best overall leach rates (Table 30) were exhibited by glasses leached in the presence of lead metal, where leach rates, based on extraction of silicon, were 5 to 20 times lower compared to previous TDS/131 leach tests. Glasses leached in solutions containing lead ions also showed a slight beneficial effect. This was most apparent for elements other than silicon. These low leach rates occurred despite the extreme differences in final pH and the highly corrosive pH regimes of both tests. For example, for leach tests conducted in the presence of lead metal the final leachate pH rose to 10.2 while for the lead-doped leachant studies the final pH decreased to 5.2. These results show that something other than silica solubility has an important beneficial effect on improving glass durability.

There were no significant differences in leachability for glasses nominally containing 0, 1, 2, 3, or 4 wt % lead oxide. In addition, these glasses showed no significant improvement in durability compared to the other studies.

- Surface Layers

Surface layers were examined after leaching for glasses leached in the presence of lead metal and for glasses leached in lead-doped solutions. In studies performed using electron microprobe analysis (SEMQ), a thin 0.5 μm surface layer, very enriched in lead, was observed on each glass surface (Figure 19). Unlike previous surface layers found on leached SRP glass,¹⁰ the observed layers did not contain large amounts of Fe, Mn, and Mg as usually found. Instead, the lead concentration dominated on the glass surface compared to all other waste or frit elements. More detailed analysis using Auger electron spectroscopy (AES) confirmed the predominately lead enriched surface layer observed on a large scale by SEMQ. Elemental profiling into the adsorbed lead enriched surface layer was performed by argon ion milling in conjunction with Auger spectroscopy, Figures 20 and 21. For the lead foil glass interactions, the lead profile did not terminate abruptly but extended approximately 1 μm into the glass. While the lead concentration decreased, the iron concentration decreased, the iron concentration increased as sputter depth increased into the glass surface layer. These profiles are consistent with earlier analyses and microstructures. These data suggest that as lead is adsorbed onto the leached glass surface, a layer is formed with some lead diffusing toward the bulk glass. This coating of the glass surface by lead appears to affect the formation of surface layers observed in earlier studies, and seems to provide the most effective protective layer observed thus far.

- Leaching Mechanisms

The formation of an enriched surface layer which protects the glass underneath is taken into account in the general SRL leachability model.¹¹ In this model, a three-stage corrosion process is defined with the most important stage being the formation and stabilization by protective layer(s). Long-term glass corrosion is ultimately governed by diffusion of soluble species through the layer or layers and the saturation of species of interest in solution. Other protective layers enriched primarily in Mg and Cl have also been observed for SRP waste glass after leaching in some brine solutions.¹² In each case, the chemical durability of the product improved with time.

Pb PROFILE
A. LEAD/GLASS INTERACTION STUDY

Pb PROFILE
B. LEAD-DOPED LEACHANT STUDY

FIGURE 19. Lead Enriched Surface Layers

FIGURE 20. Elemental Depth Profiles of Waste Glass Surface Layers Using Auger Analysis

FIGURE 21. Elemental Depth Profiles of Waste Glass Surface Layers Using Auger Analysis

Structural Performance of Species of Interest During Leaching Tests Involving Lead

The significant beneficial effect of leach tests involving externally supplied lead is summarized graphically in Figure 22. These results are compared to internally doped lead glass and a reference composition involving no lead. In addition, the role that species play during the leaching operation is superimposed on the graph. The effectiveness of the lead surface layer in protecting the glass underneath can easily be noted for glass network formers (Si, B), intermediates (Al), and even glass modifiers (Na) in this graph. This indicates that the basic random network structure of waste glass is more stable or durable for glass leached in the presence of externally supplied lead. Hence, radionuclides that would be contained within this structure would be immobilized to a higher degree.

Chemical Durability of SRP Waste Glass -- Time/Temperature Dependence

Introduction and Summary

Over the past two years, the leaching behavior of Savannah River Plant (SRP) waste glass has been extensively studied to predict the long-term stability of the glass. Recently, a physical and mathematical model of the leaching process was developed, based on simple concepts of glass corrosion and protective layer formation which accurately describes the three-stage corrosion behavior of waste glass. Emphasis is now being placed on confirming the model by more detailed studies of glass corrosion. The effects of pH, SA/V ratio, surface finish, protective layers formed, time, and temperature are being studied in detail, and the model is being extended by applying it to other waste glass systems.

In this study the effects of time and temperature on the leaching of SRP waste glass are examined. Data obtained in previous studies on both laboratory scale and full scale, TNX-continuous melted glasses are used. Based on these data:

- The time dependence of normalized leach rates (based on Si) correspond relatively well with the behavior predicted by the SRL leachability model.
- Over the temperature range 40 to 150°C, leachants which are less aggressive to waste glass have higher activation energies for leaching, representing a higher barrier to glass dissolution.

FIGURE 22. Lead/Waste Glass Static Leach Tests

(MCC-1 Test Procedures, 90°C, SA/V = 0.1 cm — Deionized Water Leachant, Except for Lead-Doped Study Which Used a pH 7 Buffered Solution)

- The aggressiveness of the leachants evaluated in the present study is in the order: pH 3 buffer > pH 11 buffer > deionized water > MCC silicate water > MCC brine > pH 7 buffer. The pH 3 buffer is 40X more aggressive than the pH 7 buffer.
- Arrhenius plots of Cs and U leachability in three different leachants indicate differences in the leaching behavior of these elements compared to Si occur at elevated temperatures. These differences may result from changes in corrosion mechanisms or changes in oxidation state as in the case of uranium.

Discussion

Time Dependence

The normalized elemental mass loss (NL_i), Table 31, has been related to the time of leaching in unsaturated systems by the SRL leachability model as follows:

$$NL_i + \frac{B(NL_i)^2}{2} = K_r t$$

or NL at t (for Stage 2 leaching - network dissolution)
and NL at $t^{1/2}$ (for Stage 3 leaching - layer formation and diffusion controlled dissolution)

Plots of NL_i versus both t and $t^{1/2}$ for times of 3, 7, 14, and 28 days are shown in Figure 23 and summarized in Table 31. As noted in the figure, both curves fit the data relatively well. However, the correlation coefficient for $t^{1/2}$ is slightly better. In addition, these curves must go through zero at time = 0 which suggests that there are different mechanisms or transient effects evoked for very short-term leaching. These potential effects would be very large for the larger y-intercept assuming linear time dependence but less dominant for $t^{1/2}$ dependence. For time periods less than 3 days, where most standard leaching tests only begin, glass leaching behavior may be quite different. Work is currently in progress investigating the very early stages of glass corrosion as a function of parameters such as surface finish, temperature, and flow rate. These data will allow us to better understand short-term leaching behavior.

FIGURE 23. Normalized Leach Rates Based on Stage 2 and Stage 3 Leaching Behavior for TNX Glass

TABLE 31

Time Dependence of Leaching of SRP Waste Glass*

TNX Premixed Waste Glass May 1982					
$t^{1/2}$ Dependence			t Dependence		
t , days	NL(Si)	Calc	Diff	Calc	Diff
3	8.82	9.61	0.97	10.46	1.65
7	13.51	12.55	-0.96	12.20	-1.31
14	16.24	16.07	-0.17	15.26	-0.98
28	20.72	21.06	0.34	21.36	0.68
Std Dev			±0.75		±1.39
$NL(Si) = 3.22 t^{1/2} + 4.03$			$NL(Si) = 0.436 t + 9.16$		
Corr Coeff = 0.989			Corr Coeff = 0.980		

131/TDS Lab-Scale Waste Glass September 1981					
3	12.69	12.29	-0.40	13.31	0.62
7	14.98	16.11	1.13	15.61	0.63
14	21.70	20.69	-1.01	19.63	-2.07
28	26.88	27.16	0.28	27.69	0.81
Std Dev.			±0.92		±1.38
$NL(Si) = 4.18 t^{1/2} + 5.06$			$NL(Si) = 0.575 t + 11.59$		
Corr Coeff = 0.990			Corr Coeff = 0.977		

* MCC-1 static leach tests, 90°C, deionized water leachant

Temperature Dependence

A direct correlation between leaching behavior and activation energy has been observed. The activation energy (Q) and leachability (L) are related by $L = kQ/RT$, where T is the absolute temperature, k is constant, and R is the gas constant. Therefore, from the slope of a plot of $\log L$ versus $1/T$ the activation energy can be obtained. For glasses previously leached in buffered solutions of pH 3 and pH 11, the leachability based on Si was relatively high and the activation energies calculated were relatively low, about 5.5 kcal/mole (Table 32). The lowest leachabilities observed occurred after leaching in a pH 7 solution with a relatively high activation energy of 14.2 kcal/mole. In the intermediate case of deionized water, the leach rates based on Si were between these extremes corresponding to about 11 kcal/mole. Similar correlations were observed for brine and silicate leachants as summarized in Figure 24. This graph also shows the 90°C data for the full-scale CM glass leached in deionized water which falls on the sample curve for laboratory-produced glass also leached in the same medium. The activation energy cannot be precisely interpreted at this time because of the complex and varied leaching processes involved. However, in an oversimplified interpretation, the activation energy represents the low barrier to glass dissolution.

A relatively linear representation was obtained by plotting log of leachability of Cs versus T^{-1} for the three temperatures studied (Figure 25). At lower temperatures, deionized water was the most aggressive leachant corresponding to the highest leachability and lowest activation energy, followed by silicate water and finally brine. However, at approximately 110°C, the three curves intersect and the relative aggressiveness of the leachants reverse. The reasons for this behavior are not presently understood but may be related to changes in the dissolution rate and subsequent sorption characteristics of Cs as a function of temperature, or possible insoluble compound formation at higher temperatures.

A plot of log of leachability of uranium (Figure 26) with T^{-1} shows a sharp decrease in slope at higher temperatures. In addition, this plot is clearly not as linear as observed for other elements and as predicted by simple glass corrosion theory. This change in slope may be attributed to several factors including possible compound formation as well as changes in oxidation state-of-the-uranium. Higher temperatures favor a lower oxidation state-of-the-uranium and improved performance.

FIGURE 24. Activation Energies Based on Silicon Extraction

FIGURE 25. Activation Energies Based on Cesium Extraction

FIGURE 26. Activation Energies Based on Uranium Extraction

TABLE 32

Leachability Based on Si ($\text{g/m}^2\text{-day}$)*

	pH 3	pH 11	DI H ₂ O	Silicate	Brine	pH 7
40°C (Lab)	0.53	0.45	0.05	0.04	0.01	0.005
90°C (Lab) (TNX-CM)	2.32	1.40	0.96	0.56	0.32	0.060
150°C (Lab)	4.96	4.66	4.28	2.96	1.41	1.53
Q (kcal/mole)	5.55	5.80	11.05	10.69	12.29	14.22

* Based on MCC-1, 28-day static leach tests, September 1981.

Physical Properties of Frit 165/Waste Glasses

Summary

The viscosities and resistivities of glasses made from Frit 165 and three waste simulations have been measured. At 1150°C, the viscosities range from 35 poise for high-iron waste to 180 poise for high-aluminum waste. The resistivities at the same temperature range from 2.9 ohm-cm for high-iron waste to 3.6 ohm-cm for TDS-3A waste. The error in both the viscosity and resistivity values is 20%.

The specific heats, densities, and thermal conductivities have been estimated for Frit 165 glasses. These estimates are based on a comparison of predicted and measured values of the properties of Frit 131 glasses.

These values of the physical properties are all consistent with those recommended for use in design of the DWPF melter.

Introduction

Frit 165 was developed as a replacement for Frit 131 once direct slurry-feeding of the DWPF melter was adopted as a reference process. Large-scale tests in the pilot-scale ceramic melter at Battelle-Pacific Northwest Laboratory have since demonstrated the processability of Frit 165 waste glasses, and verified substantial improvements in leaching resistance and off-gas decontamination factors seen in laboratory tests.

This report gives viscosity and resistivity data over the range 800 to 1200°C for waste glasses made using Frit 165 with three waste compositions — high-iron, high-aluminum, and TDS-3A (an average composition).

Specific heats, densities, and thermal conductivities over the range 20 to 1200°C are also reported.

Viscosities and Resistivities

The glasses studied were prepared from Frit 165 (Table 33) and three different waste simulations; high-aluminum, TDS-3A (average), and high-iron (Table 34).

The measured viscosities and resistivities appear in Tables 35 and 36, respectively. These data were fit to the Fulcher equation:

$$\log_{10} \eta \text{ or } \log_{10} \rho = A + \frac{B}{T - T_0}$$

to give the parameters of Table 37.

The solid lines in Figures 27 and 28 were generated using the Fulcher equation; the points on each curve are measured values. The difference between the data points and the Fulcher curves are always less than the experimental error.

Table 38 gives the values of the viscosities and resistivities for these glasses using the Fulcher constants. At 1150°C the viscosities range from 35 to 180 poise, and the resistivities range from 2.5 to 3.6 ohm-cm.

Specific Heats

The measured specific heats of 28% TDS-131 glass at 25°C, 375°C, and 477°C¹³ are plotted in Figure 29, along with calculated values. The greatest deviation between the experimental and calculated values is 6% at 375°C. The experimental and calculated values are both compatible with published values of specific heats, and are within the range of experimental errors in published values for similar glasses¹⁴ (see Figures 30 and 31). Values calculated using the method of Reference 13 are shown in Table 39 for Frit 165 glasses. It has previously been shown that specific heat is a weak function of composition,¹³ e.g., it varies only 7% between high-iron and high-aluminum types.

FIGURE 27. Viscosities of Frit 165 Glasses

FIGURE 28. Resistivities of Frit 165 Glasses

FIGURE 29. Specific Heats of Frit 131 Glasses

FIGURE 30. Specific Heat of SiO₂ + Na₂O + Σ K_i Soda Lime Glass

FIGURE 31. Specific Heat of $\text{SiO}_2 + \text{B}_2\text{O}_3 + \text{EX}_i$ Borosilicate Glass

TABLE 33

Frit 165 Composition

Oxide	Wt %
SiO ₂	68.0
Na ₂ O	13.0
B ₂ O ₃	10.0
Li ₂ O	7.0
MgO	1.0
ZrO ₂	1.0

TABLE 34

Simulated Calcine Compositions

Component	Amount, wt %		
	W-Al	TDS-3A	W-Fe
Fe ₂ O ₃	13.8	47.3	59.1
MnO ₂	11.3	13.6	4.0
Zeolite	10.2	10.2	9.7
Al ₂ O ₃	49.3	9.5	1.4
NiO	2.0	5.8	10.1
SiO ₂	4.5	4.1	2.9
CaO	0.9	3.5	4.0
Na ₂ O	5.0	3.1	5.9
Coal	2.3	2.3	2.1
Na ₂ SO ₄	0.7	0.6	0.8
Frit/ Calcine Ratio	71.3/28.7	70.2/29.8	70.2/29.8

TABLE 35

Measured Viscosities of Frit 165 Glasses

W-Al		TDS-3A		W-Fe	
T, °C	η , poise	T, °C	η , poise	T, °C	η , poise
1169	150	1164	60	1166	33
1120	240	1115	88	1116	43
1074	380	1067	140	1069	62
1024	660	1020	240	1020	98
976	1200	973	390	973	160
925	2200	923	830	923	290
878	4600	876	1900	875	620
830	11000	827	4900	825	1400

TABLE 36

Measured Resistivities of Frit 165 Glasses

T, °C	ρ , ohm-cm		
	W-Al	TDS-3A	W-Fe
1200	2.8	3.0	2.4
1100	3.8	4.5	3.5
1000	6.5	7.4	5.7
900	10	12	8.9
800	20	27	19

TABLE 37

**Fulcher Parameters for Viscosities and Resistivities
of Frit 165 Glass**

Property	Waste	A	B, °C	T ₀ , °C
Viscosity	W-Al	-1.5662	3876.2	138.01
	TDS-3A	-0.9462	2216.3	349.97
	W-Fe	-0.6307	1659.3	387.47
Resistivity	W-Al	-1.5568	2646.1	-128.61
	TDS-3A	-1.2346	1924.3	77.94
	W-Fe	-1.1434	1682.5	103.74

* The data were fit to the Fulcher Equation,

$$\log_{10} \eta \text{ or } \log_{10} \rho = A + \frac{B}{T - T_0}$$

to give these constants.

TABLE 38

**Interpolated Values of the Viscosity and Resistivity
of Frit 165 Glasses at Selected Temperatures***

T, °C	W-Al		TDS-3A		W-Fe	
	η , poise	ρ , ohm-cm	η , poise	ρ , ohm-cm	η , poise	ρ , ohm-cm
1200	120	2.7	46	3.0	26	2.5
1150	180	3.2	67	3.6	35	2.9
1100	290	3.9	100	4.4	50	3.5
1050	480	4.9	170	5.6	75	4.3
1000	850	6.1	290	7.1	120	5.4
950	1600	7.8	560	9.4	210	7.0
900	3300	10	1200	13	400	9.3
850	7500	14	3100	18	910	13
800	19000	19	9500	27	2500	19

* Calculated using the Fulcher parameters of Table 37.

TABLE 39

Calculated Specific Heats of Frit 165 Glasses, Cal/g-°C

Temp, °C	Mean Heat 165 TDS-3A	True Heat 165 TDS-3A	Mean Heat High Fe	True Heat High Fe	Mean Heat High Al	True Heat High Al
20	0.198	0.204	0.198	0.204	0.196	0.202
100	0.218	0.240	0.219	0.243	0.216	0.237
200	0.237	0.272	0.240	0.278	0.235	0.269
300	0.253	0.295	0.257	0.302	0.250	0.291
400	0.266	0.312	0.270	0.320	0.263	0.308
500	0.276	0.325	0.282	0.334	0.273	0.321
600	0.285	0.334	0.292	0.344	0.282	0.330
700	0.293	0.342	0.300	0.353	0.289	0.338
800	0.299	0.349	0.307	0.359	0.296	0.344
900	0.305	0.354	0.313	0.365	0.302	0.349
950	0.308	0.356	0.316	0.367	0.304	0.352
1000	0.310	0.358	0.318	0.370	0.307	0.354
1025	0.311	0.359	0.320	0.371	0.308	0.355
1050	0.313	0.360	0.321	0.372	0.309	0.355
1075	0.314	0.361	0.322	0.372	0.310	0.356
1100	0.315	0.362	0.323	0.373	0.311	0.357
1125	0.316	0.362	0.324	0.374	0.312	0.358
1150	0.317	0.363	0.325	0.375	0.313	0.359
1175	0.318	0.364	0.326	0.376	0.314	0.359
1200	0.319	0.365	0.328	0.377	0.315	0.360
1250	0.321	0.366	0.330	0.378	0.317	0.361
1300	0.322	0.367	0.331	0.379	0.319	0.363

Above the glass transition point there appears to be significant divergence between the linear extrapolation of the experimental data and calculated values. For critical applications experimental values should be determined, or the corresponding calculated values used with an assumed accuracy of $\pm 10\%$ below 700°C , and $\pm 25\%$ above 700°C .

Density

Low temperature densities and thermal expansion coefficients have been measured for Frit 131 glasses.¹³ In addition, high temperature Frit 131 densities have been measured in the range from 1000 to 1150°C . The slope of high temperature density with respect to temperature (which can be used to calculate buoyancy forces) is correct to $\pm 10\%$. The low temperature values are relatively insensitive to composition,¹³ and are compatible with those of similar sodium silicate and borosilicate glasses¹⁵ (see Figures 32 and 33).

Values for Frit 165 glasses are shown in Table 40. Low temperature densities (up to 400°C) were calculated using experimentally determined room temperature densities and the corresponding Frit 131 glass between 400°C and the upper dilatometric softening point. Densities between 1000 and 1200°C were calculated by multiplying the room temperature density ratios of the corresponding Frit 165 and Frit 131 types times the experimentally determined high temperature densities of Frit 131 glasses. Values between 500°C and 1000°C are interpolated, with assumed accuracies of $\pm 10\%$.

Thermal Diffusivities/Thermal Conductivities

Thermal diffusivities of Frit 131 glasses have been measured in the range of 950 to 1150°C ¹⁶ (see Figure 34). Thermal conductivities have been calculated using the measured thermal diffusivities and the heat capacities and densities discussed in this report (see Table 41). Calculated thermal conductivities are available as a function of iron content.

Typical measured thermal diffusivities are plotted against temperature using published data for several glasses in Figure 35. Although the measured values for the Frit 131 glasses and the synthetic tektite glass ($75\% \text{SiO}_2$, $11.5\% \text{Al}_2\text{O}_3$, $2\% \text{K}_2\text{O}$, $4.4\% \text{Fe}_2\text{O}_3$, $0.5\% \text{FeO}$) are similar, the temperature dependence is stronger for Frit 131 glasses. The temperature dependence is similar to what might be expected of a borosilicate glass or other complex glass, but the levels are not compatible. Since the extrapolated diffusivities go negative at approximately 1150°C , it might be assumed that the experimental method used has calibration difficulties for heavily doped glasses. The thermal conductivities calculated from

FIGURE 32. Comparison of Thermal Expansion of Frit 131 to Sodium Silicate Glass

FIGURE 33. Comparison of Thermal Expansion of Frit 131 to Borosilicate Glass

FIGURE 34. Measured Frit 131 Thermal Diffusivities

FIGURE 35. Thermal Diffusivity of Glasses

TABLE 40

Densities of SRP Waste Glasses

Temp, °C	131 TDS-3A, g/cm ³	131 High-Fe, g/cm ³	131 High-Al, g/cm ³	165 TDS-3A g/cm ³	165 High-Fe g/cm ³	165 High-Al g/cm ³
20	2.75	2.82	2.60	2.76	2.94	2.67
100	2.743	2.813	2.593	2.753	2.933	2.663
200	2.735	2.804	2.585	2.745	2.923	2.655
300	2.726	2.795	2.577	2.736	2.914	2.646
400	2.718	2.786	2.569	2.728	2.905	2.638
500	2.68	2.75	2.537	2.680	2.867	2.605
600	2.64	2.701	2.516	2.65	2.816	2.584
700	2.60	2.651	2.495	2.61	2.764	2.562
800	2.56	2.602	2.474	2.57	2.713	2.541
900	2.52	2.552	2.453	2.53	2.663	2.519
1000	2.48	2.503	2.432	2.49	2.610	2.497
1025	2.47	2.498	2.426	2.48	2.604	2.491
1050	2.465	2.493	2.42	2.47	2.599	2.485
1075	2.452	2.486	2.445	2.465	2.592	2.511
1100	2.45	2.48	2.407	2.46	2.586	2.472
1125	2.443	2.475	2.401	2.452	2.580	2.466
1150	2.436	2.47	2.395	2.445	2.575	2.459
1175	2.429	2.465	2.389	2.438	2.57	2.453
1200	2.422	2.46	2.383	2.431	2.71	2.447

TABLE 41

Elevated Temperature Thermal Diffusivity and Thermal Conductivity of Frit 131 Glasses

Temp, °C	Measured Thermal Diffusivity, cm ² /sec			Calculated Thermal Conductivity, cal/sec cm °C		
	131 TDS-3A	131 Hi/Fe	131 Hi/Al	131 TDS-3A	131 Hi/Fe	131 Hi/Al
950	0.0035	0.0028	0.0057	0.00316	0.00258	0.00451
1000	0.0063	0.0038	0.0095	0.00568	0.00348	0.00764
1050	0.0096	0.0086	0.0125	0.00864	0.00790	0.0100
1100	0.0129	0.0099	0.0156	0.0116	0.00909	0.0126
1150	0.0161	0.015	0.0175	0.0145	0.0138	0.0142

the measured thermal diffusivities do confirm the magnitude of the previously calculated conductivities in the operating range between 1000 to 1200°C (see Figures 36, 37, and 38). The measured diffusivities also confirm the dependence of diffusivity and conductivity on iron, which served as the basis for the previous calculations.

To help resolve these discrepancies, the measurement of low temperature thermal conductivities of Frit 165 glasses between the glass transition and melter operating temperatures have been requested. In view of the discrepancies between calculated and measured values, we recommend using the calculated Frit 131 numbers with an assumed accuracy of $\pm 50\%$ below 900°C, and $\pm 25\%$ from 1050 to 1200°C. Frit 131 and 165 values should be similar since they are controlled by iron content.

Experimental/Quality of Data

Viscosities were measured with a Brookfield rotating-spindle viscometer using a platinum spindle. Temperature measurements were made using a platinum/platinum-rhodium thermocouple at the base of the alumina crucible containing the glass.

The composition of Frit 165 is given in Table 33, and those of the three waste compositions in Table 34. The actual samples were prepared in 250 g (as glass) batches following Table 42. The final compositions, as measured by inductively coupled plasma emission spectroscopy (ICP), appear in Table 43. The chemical analyses are based on triplicate samples for the high-aluminum and TDS-3A glass, and on duplicate samples for the high-iron glass.

FIGURE 36. Effective Thermal Conductivity for Frit 131 Composite Waste.

FIGURE 37. Effective Thermal Conductivity for Frit 131 High-Fe Waste

FIGURE 38. Effective Thermal Conductivity for Frit 131 High-Al Waste

TABLE 42

Glass Composition — As Prepared

Component	Grams per 100 grams glass		
	W-Al	TDS-3A	W-Fe
Frit 165	71.30	70.20	70.20
Fe ₂ O ₃	3.96	14.10	17.61
MnO ₂	3.24	4.05	1.19
Zeolite	2.93	3.04	2.89
Al ₂ O ₃	14.15	2.83	0.42
NiO	0.58	1.73	3.01
SiO ₂	1.29	1.22	0.86
CaO	0.26	1.04	1.19
Na ₂ CO ₃ ·H ₂ O	2.87	1.37	3.52
Coal	0.66	0.68	0.62
Na ₂ SO ₄	0.20	0.18	0.24

TABLE 43

Glass Compositions — wt %

Oxide	W-Al			TDS-3A			W-Fe		
	Reported	Normalized	Nominal	Reported	Normalized	Nominal	Reported	Normalized	Nominal
Al ₂ O ₃	14.5	16.3	14.2	5.2	6.2	2.8	3.3	14.3	0.4
B ₂ O ₃	6.9	7.7	7.1	6.4	7.8	7.0	6.5	8.4	7.0
CaO	0.3	0.3	0.3	0.4	0.5	1.0	0.7	0.9	1.2
Fe ₂ O ₃	4.2	4.7	4.0	13.9	16.8	14.1	15.9	20.6	17.6
Li ₂ O	4.9	5.6	5.0	4.8	5.8	4.9	4.5	5.9	7.0
MgO	0.7	0.7	0.7	0.7	0.8	0.7	0.7	0.9	0.7
MnO ₂	3.2	3.6	3.2	4.0	4.8	4.1	1.1	1.4	1.2
Na ₂ O	10.4	11.7	10.8	9.4	11.4	9.9	10.1	13.1	11.0
NiO	0.5	0.6	0.6	1.5	1.9	1.7	2.1	2.7	3.0
TiO ₂	0.1	0.1	0.0	0.1	0.1	0.0	0.1	0.1	0.0
ZO ₂	0.7	0.8	0.7	0.7	0.8	0.7	0.7	0.9	0.7
SiO ₂	42.6	47.9	49.8	35.4	43.1	49.0	31.3	40.8	48.6
Σ	89.0	100.0	96.4	82.5	100.0	95.9	77.0	100.0	98.4

Reported = Converted to oxide from elemental analysis by inductively — coupled plasma emission spectroscopy (ICP).

Normalized = ICP results forced to add to 100%.

Nominal = Calculated from Tables 33 and 34, neglecting zeolite.

ICP is capable of measuring every element in the glass except oxygen. Therefore, when every metal measured by ICP is converted to its highest oxide, the sum should equal — or exceed — 100 wt %. This was never achieved in practice. Table 43 reports both the as-measured oxide values and the values when the oxide sum is forced to be 100%. The nominal oxide levels generally fall between these two numbers, except for silica (SiO_2) and alumina (Al_2O_3).

Measurement of silica by ICP requires a separate sample preparation. The consistently low measured values of silica are probably due to a systematic error in this preparation or in the ICP measurement of silicon. In spite of this, ICP remains the best way to analyze silicon currently available at SRL.

The excess alumina is probably from corrosion of the high (>99.8%) alumina crucibles in which the samples were prepared and the viscosity measurements taken. The small extra amount of alumina should not change either the resistivity or viscosity significantly.

The primary source of error in the viscosity measurements is in the calibration factors found by measuring a standard glass from the National Bureau of Standards. The analysis of error in this calibration is the subject of a separate report. The effect of any error in the calibration factor would be to shift all three curves of Figure 27 up or down a constant amount. The estimated error in the calibration factor is 20%.

It should be noted that the relative viscosities — whether between different glasses or between different temperatures with the same glass — are accurate to much better than 20%. The maximum such error is 14%, and a more typical value would be 3%. This error arises from the uncertainty in reading the viscometer, which is $\pm 1\%$ full scale regardless of where the reading lies on the scale (0 to 100).

After the viscosity data were collected the samples were broken out of their crucibles for use in resistivity measurements. These were measured using an electrode separation technique. The resistance of the glass melt is measured in a ceramic crucible as a function of electrode separation. The current is held constant. A plot of resistance versus separation gives a straight line of slope ρ/A where ρ is the resistivity and A is the cross-sectional area, assumed constant. The area is measured by planimeter, and the product of the area and the slope gives the resistivity.

The resistivity measurements were also made on triplicate samples. The standard errors ranged from 18.3% down to 0.44%, with an average of 4.9%. The standard errors in the slopes of the plots of resistance versus electrode separation range from 0 to 2.6% with

an average of 0.61%. The errors in measuring the cross-sectional area of the ceramic crucibles used to hold the glass are not large, but the variations in area (as the electrode separation increases) are. This error is mainly responsible for the 20% error in the final resistivities.

The method used to measure and calculate heat capacities, densities, and thermal conductivities are included in the material referenced in the earlier discussions. The accuracy of these measurements will be covered in a future report.

Chemical Oxygen Demand — A Technique for Evaluating Reducing Strength of Melter Feeds

Chemical oxygen demand (COD), a titrimetric technique of analyzing the reducing strength of melter feeds, has been applied to laboratory samples and to actual feeds for large-scale melters at TNX and PNL. The precision of the technique and agreement between measured and predicted COD values are excellent. COD values for the SRL and PNL feeds are equal, although different feed preparation methods are used. Thus, the metal formation observed at PNL, but not at SRL, is from differences in melter design and operation rather than differences in feed chemistry. Relating to $\text{Fe}^{2+}/\text{Fe}^{3+}$ ranges for glasses from the two melters to their common COD value shows that it is possible, for a given melter, to specify a COD value below which metal formation will not occur. Chemical oxygen demand is also a potentially useful quality control tool to test actual melter feeds, since it will "flag" any excessively reducing feeds before they reach the melter.

Analysis of Chemical Oxygen Demand

The chemical oxygen demand technique used for melter feed materials was developed by the FMC Corporation. It is a minor variation of the ASTM Standard Method for Chemical Oxygen Demand of Industrial Waste Water.

A known mass of feed is refluxed for two hours with sulfuric acid and potassium dichromate. Any oxidizable component in the feed will be oxidized under these conditions. The only two reactions which should occur for melter feeds are those of the dichromate ion with formic acid or with manganese (II):

The rest of the components in the feed, including all of the oxides in the frit, are already in their highest oxidation state.

The dichromate is added in excess, and this excess is titrated with iron (II):

Since the amount of dichromate initially added is known, as is the amount left after reflux, the amount consumed by the feed is found by difference. The results are expressed as weight percent carbon equivalent, i.e., the amount of carbon that would be consumed by that amount of dichromate:

The precision of this technique at SRL is excellent, as may be seen in Table 44. The samples are listed in chronological order.

Duplicate wet and dried feed samples were analyzed for chemical oxygen demand. This excellent precision was seen only for the wet samples. The dried samples were consistently lower than expected, but not by reproducible amounts. Presumably some of the formic acid was vaporized during drying, although the exact amount depends on the sample properties and drying conditions. It is recommended that only wet samples be analyzed in the future.

Prediction of Chemical Oxygen Demand

1. Chemical Oxygen Demand of Formic Acid

One mole of formic acid has half the reducing power of one mole of carbon. This can be seen by comparing the two half-reactions:

One mole of formic acid thus corresponds to 6 g of carbon. This was checked by measuring formate ion concentrations and chemical oxygen demands of two dilutions of concentrated formic acid. The results are shown in Table 45. The agreement between the predicted (6 g C/mole formic acid) and measured values is better at the lower acid concentration. This concentration is more typical of what will be used in actual sludge processing. However, even for more concentrated acid, the agreement is within 10%.

TABLE 44

Precision of SRL Analysis of Chemical Oxygen Demand

Source*	Number of Samples	Mean COD	Standard Deviation
LAB	5	2.50	0.19
	5	1.92	0.08
PSCM	6	1.22	0.01
	6	0.90	0.05

* PSCM = Pilot-Scale Ceramic Melter at PNL.

TABLE 45

Chemical Oxygen Demand of Formic Acid

Dilution	Concentration,* moles/liter	COD, wt % C	Density, g/cm ³	g C/mole HCOO ⁻ **
1:2	12.8	6.4	1.12	5.6
1:10	2.66	1.53	1.03	5.9

* Measured by ion chromatography.

$$** \frac{\text{g C}}{\text{g-mole HCOO}^-} = \frac{1}{\text{concentration}} \times \frac{\text{COD}}{100\%} \times \text{Density} \times \frac{1000 \text{ cm}^3}{\text{liter}}$$

2. Chemical Oxygen Demand of Melter Feeds

The COD's of both Frit 131 and Frit 165 were measured and found to be zero. The chemical oxygen demand of pure frit should be zero, since each constituent of the frit is already in its highest oxidation state. The waste (before formic acid treatment) is mildly oxidizing, and also has a negligibly small COD compared to formic acid. Unless carbon is added in the waste simulation, only formic acid will contribute to the COD of the feed.

As was shown above, 1 mole of formic acid corresponds to 6 g of carbon. Therefore, each mole of formic acid per pound of waste solids contributes

$$\frac{1 \text{ mole HCOOH}}{1 \text{ lb sludge solids}} \times \frac{6 \text{ g carbon}}{1 \text{ mole HCOOH}} \times \frac{1 \text{ lb}}{454 \text{ g}} \times 100\% \\ = 1.32 \text{ wt \% carbon}$$

In actual feed samples, frit and water will also be present. Since they add to the mass of the sample but contribute nothing to the COD, they reduce the measured chemical oxygen demand. Throughout this report all COD values are corrected for water and are therefore weight percents carbon relative to the total solids (sludge solids and frit) in the feed. Taking the frit into account gives the final predicted COD:

$$\text{predicted COD} = \frac{\text{moles HCOOH}}{\text{lb sludge solids}} \times 1.32 \text{ wt \% C} \\ \times \frac{\text{wt \% sludge of feed solids}}{100\%}$$

This equation was used to give the predicted COD values of Table 46 and Figure 39. The feed solids were assumed 35% sludge and 65% frit.

The measured COD values fall very close to the predicted ones, as may be seen from Table 46 and Figure 39. The predictions are at their best on actual feed samples from the large slurry-fed melter tests at SRL, where the feed preparation technique most closely follows the flowsheet for the actual DWPF. The differences between the predicted and measured values for the pilot-scale ceramic melter at PNL are most likely due to uncertainties in the predicted values, based as they are on calculated values of moles formic acid/pound hydroxide sludge solids. Since the measured COD values are so close to those predicted, the COD method could provide a valuable quality control test when used on actual feed in the DWPF.

FIGURE 39. COD as a Function of Formic Acid Level

TABLE 46

Predicted and Measured Values of Chemical Oxygen Demand

Source*	Run	<u>g-moles HCOOH**</u> <u>lb Sludge Solids</u>	<u>Predicted COD</u> <u>(wt % C/Feed Solids)</u>	<u>Measured COD†</u> <u>(wt % C/Feed Solids)</u>	<u>Formed Metal††</u> <u>Closed Crucible</u>
LAB	1	0	0	0	No
	2	4	1.85	1.76	Yes
	3	6	2.77	2.75	Yes
	4	8	3.70	3.24	Yes
	5	10	4.62	4.45	Yes
LSFM (SRL)	SME 5	2.6	1.20	1.26	Yes
	SME 6	3.1	1.43	1.49	Yes
PSCM (PNL)	5	2.75	1.27	1.12	NT
	6	2.86	1.29	1.19	NT
	7	2.86	1.29	1.21	Yes
	8	2.53	1.17	0.91	No

* LSFM = Large Slurry-Fed Melter
PSCM = Pilot-Scale Ceramic Melter

** LAB and LSFM values are as prepared. PSCM values are calculated.

† Corrected for water;

$$\text{Corr COD} = \text{COD} \times \frac{100\%}{\text{wt \% solids in slurry}}$$

†† NT = Not Tested (insufficient sample)

Comparison of Large-Scale Melter Feeds

Metal has been found in significant quantities on the floor of the pilot-scale ceramic melter (PSCM) at Battelle-Pacific Northwest Laboratory (PNL). This metal had been seen in floor scrapings, but became especially noticeable after the melter was drained. Floor scrapings of the large slurry-fed melter (LSFM) at SRL have not yet yielded any metal. This melter is still in active service and has never been drained. Metal formation is a serious problem in these joule-heated melters, since there is a potential to form a short circuit between the electrodes, completely disabling the melter. At the time the metal was found at PNL, it was unclear whether its formation in the PSCM — and not in the LSFM — was because of differences in melter design and operation, or to differences in feed preparation. Feed preparation for the LSFM closely follows the flowsheet for the actual DWPF. The simulated waste sludge is received as hydroxides and then refluxed during addition of formic acid. The formic acid reduces Mn(IV) in manganese dioxide to Mn(II) and forms formate salts of at least some of the sodium, calcium, manganese(II), iron(III), nickel, and strontium. It should be noted that measurements of soluble manganese after reflux indicated that only some 3/4 of the manganese(IV) is reduced. The moles of formic acid per pound of sludge solids listed in Tables 46 and 47 for the LSFM are moles of formic acid added during reflux, i.e., before any Mn(IV) reduction has occurred.

At PNL, feed preparation for the PSCM had to be done without reflux. They therefore started with Mn(II) formate, nickel formate, calcium formate, and sodium formate. This gives the feed composition one would expect if the Mn(IV) reduction and acid-base reactions in the LSFM feed preparation went to completion. Additional formic acid was added to bring the pH to 5.5. Calculation of the equivalent number of moles of formic acid per pound of hydroxide sludge solids for this feed preparation is somewhat involved.

The final result is given in Table 47. In spite of the differences in feed preparation, the chemical oxygen demand values are the same. Neither the LSFM nor the PSCM feed is inherently more reducing than the other. Both form metal in closed crucible tests (2 hr at 1150°C).

Reducing Conditions in Large-Scale Melters

Although the chemical oxygen demand values for feeds to the PSCM and LSFM are the same, there is no doubt that the PSCM is more reducing. This is shown not only by the formation of metal in the PSCM — and not in the LSFM — but also by the $\text{Fe}^{2+}/\text{Fe}^{3+}$ ratios in glasses from the two melters (see Table 47). The higher the ratio the more reducing the conditions under which the glass was formed.

TABLE 47

SRL Versus PNL Melter Feeds and Product Glasses

Source*	<u>g-moles HCOOH**</u> <u>lb Sludge Solids</u>	<u>COD,</u> <u>wt % C/Feed Solids</u>	<u>Fe²⁺/Fe³⁺</u> <u>in Glass</u>	<u>Metal in</u> <u>Melter**</u>
LSFM (SRL)	2.6	1.3	0.16-0.33	No
PSCM (PNL)	2.8	1.2	0.55-0.90	Yes

* LSFM = Large Slurry-Fed Melter
PSCM = Pilot-Scale Ceramic Melter

** The LSFM was checked for metal by scraping the bottom. The PSCM was examined after draining the melter, when detection of metal is much easier.

The highest Fe²⁺/Fe³⁺ ratio for LSFM glass is lower than even the lowest Fe²⁺/Fe³⁺ ratio for PSCM glass.

Since the reducing strengths of the feeds to the two melters are the same, the more reducing conditions in the PSCM must be due to differences in melter design and operation. Two differences which may be responsible are (1) the LSFM has a bubbler (0.05 scfm) the PSCM does not, and (2) the air leakage into the LSFM (80 to 160 scfm) is higher than that into the PSCM (10 to 40 scfm). Air inleakage can lower the concentrations of H₂ and CO both by reaction and dilution. The argon used in the bubbler can only lower these concentrations by dilution.

These results are summarized in Figure 40. The range marked "SRL" was found by taking the highest and lowest Fe²⁺/Fe³⁺ values for the LSFM from Table 47 at a COD of 1.2 wt % C/feed solids. A line from both of these points was drawn through the origin. The "PNL" area was found the same way. The line marked "crucible" was found to be linear up to 8 moles formic acid/pound sludge solids, although only the part of this line near the melter feed COD levels is shown. The region marked "onset of nickel metal formation" includes, somewhere in its range, the Fe²⁺/Fe³⁺ ratio at which nickel metal will first be formed. It will, of course, be formed at all higher levels of Fe²⁺/Fe³⁺. The vertical line at a COD of 1.2 wt %/feed solids show the present COD value of feeds to both melters.

FIGURE 40. Ratio of $\text{Fe}^{2+}/\text{Fe}^{3+}$ versus COD

The SRL glasses fall near or below the lowest $\text{Fe}^{2+}/\text{Fe}^{3+}$ ratio where nickel formation is possible. Both the PNL and crucible glasses fall well above even the highest $\text{Fe}^{2+}/\text{Fe}^{3+}$ ratio for the onset of nickel formation. It is therefore not surprising that metal formation has been observed in the PSCM at PNL, but not at the LSFM at SRL. Since the COD values of the feeds are the same, the differences in ease of nickel metal formation between the two sites must be due to melter design and operation, and not to differences in feed preparation.

Current studies are focusing on finding the lowest formic acid levels at which reduction of mercury and manganese(IV) will occur, and to pinpoint which parameters of melter operation most efficiently make the melt more oxidizing. This latter study is equivalent to finding conditions which lower the slopes of the $\text{Fe}^{2+}/\text{Fe}^{3+}$ versus COD boundaries in Figure 40. Knowledge of these boundaries for any melter will allow specification of the critical COD value to prevent nickel metal formation.

Radiation and Chemical Stabilities of Cesium Tetrphenylborate and Strontium-Plutonium Titanate

Introduction

The high-level liquid waste generated at SRP over the last 28 years is to be decontaminated by precipitating cesium-137 as cesium tetrphenylborate and by adsorbing strontium-90 and soluble plutonium on sodium titanate. This inherently simple process yields a highly decontaminated liquid waste, and, if carried out within a waste tank, can realize significant savings in capital investment since no additional canyon building is required.

For the in-tank decontamination to be successful, the precipitate must be stable during the decontamination. In the current multibatch concept, about three months will be required to complete a four-batch operation. The purpose of this report is to verify that the precipitate will be stable during the in-tank demonstration and that a high DF is maintained in the entire multibatch cycle.

Details

The chemical and radiation stabilities of the precipitate were determined in a HLC cell using the actual liquid waste from Tank 19F and 37H. Only about 300 mL were used in each test, because the radiation dose to the precipitate is mainly beta radiation.

The radioactive and chemical compositions of Tank 19F and 37H liquid wastes are given in Tables 48 and 49, respectively. The Tank 37H liquid waste was diluted to 4M sodium before the aging test. The radioactivity of the diluted Tank 37H sample was about 43 times higher than that of Tank 19F. This big difference in radioactivity enables us to determine whether chemical decomposition or radiolytic decomposition of the precipitate is the controlling factor.

The test results for Tank 19F sample are given in Table 50, where the DF values for cesium-137 are tabulated against time. Strontium-90 and plutonium were not included because sodium titanate was not available at that time and nonradioactive strontium was used as a carrier for strontium-90. The results indicated that good DF values are maintained over 389 days, more than four times longer than the anticipated duration of the multibatch decontamination.

The test results of Tank 37H liquid waste after 348 days are given in Table 51, where the DF values for cesium-137, strontium-90, and total plutonium are tabulated as a function of time. The initial DF for cesium-137 was very low because two hours of contact time was insufficient at 0.01M STPB excess. The DF values for cesium improved steadily through ion exchange between cesium ions and STPB precipitate and reached a steady state at 11 days of contact time (no mixing).

The test results indicate that the Tank 37H precipitate is stable over 348 days maintaining a cesium DF of approximately 10^6 . Since the radioactivity of Tank 37H liquid waste is about 43 times greater than the liquid waste in Tank 19F, we would expect that the Tank 19F sample would be stable for about 41 years. On this basis, chemical stability of the precipitate may be the controlling factor because it is unlikely that Tank 19F sample could be stable for so long. The DF values for strontium and plutonium for Tank 37H sample remained nearly constant during the test, indicating that strontium and plutonium titanate are also stable within this period.

In conclusion, the stability data for both samples demonstrate that the precipitate is stable on both radiation and chemicals, and that no appreciable decomposition will take place in a full cycle multibatch in-tank decontamination.

TABLE 48

Composition of Tank 19F Dissolved Salt Solution

A. Major Radioactive Isotopes

Cs-137	7×10^7 d/m/mL
Sr-90	1×10^5 d/m/mL
Ru-106	1×10^4 d/m/mL
Tc-99	3.2×10^4 d/m/mL
Pu	10^3 d/m/mL

B. Chemical Composition

Na ⁺	4.5M	SO ₄ ⁼	0.2M
NO ₃ ⁻	2.7M	NO ₂ ⁻	0.1M
AlO ₂ ⁻	0.4M	CO ₃ ⁼	0.1M
OH ⁻	1.1M	F ⁻	0.023M
K ⁺	0.015M	Cl ⁻	0.005M
C ₂ O ₄ ⁼	2×10^{-3} M	I ⁻	<7 ppm

TABLE 49

Composition of Tank 37H

A. Major Radioactive Isotopes

Cs-137	5.0×10^9 d/m/mL
Sr-90	2.10×10^7 d/m/mL
Ru-106	1.66×10^7 d/m/mL
Tc-99	1.90×10^6 d/m/mL
Pu	1.60×10^5 d/m/mL

B. Chemical Composition

Na ⁺	10.6M	NO ₂ ⁻	2.1M
NO ₃ ⁻	1.75M	CO ₃ ⁼	0.02M
AlO ₂ ⁻	0.1M	FeO ₂ ⁻	2×10^{-4} M
OH ⁻	4.75M	PO ₄ ⁼³	0.014M
K ⁺	0.06M	Cs ⁺	5.99×10^{-4}
SO ₄ ⁼	0.02M	sp. gr.	1.424

TABLE 50

Radiation and Chemical Stability of CsTPB in Tank 19F Supernate

Item	Days Since Precipitation	Aqueous Phase Cs-137, d/m/mL	Cs-137 DF
Feed	-	4.4×10^7	
A-2	12	ND	4.4×10^5
A-3	26	3.46×10^2	1.3×10^5
A-4	42	1.0×10^2	4.4×10^5
A-5	56	2.8×10^3	1.6×10^5
A-6	69	1.19×10^3	3.7×10^4
A-7	88	ND	4.4×10^5
A-8	120	ND	4.4×10^5
A-9	142	2.7×10^3	1.6×10^4
A-10	161	4.2×10^3	1.1×10^4
A-11	191	$< 3.7 \times 10^3$	$> 1.2 \times 10^4$
A-12	225	5.2×10^2	8.5×10^4
A-13	245	1.8×10^3	2.4×10^4
A-14	286	6.8×10^2	6.5×10^4
A-15	344	1.9×10^2	2.3×10^5
A-16	389	4.5×10^2	9.8×10^4

Initial Na-TPB excess was 0.013M.

TABLE 51

Radiation and Chemical Stability of CsTPB in Tank 37H Supernate

Item	Days Since Precipitation	Aqueous Phase, Cs-137*		Aqueous Phase, Sr-90		Aqueous Phase, Pu	
		d/m/mL	DF	d/m/mL	DF	d/m/mL	DF
Feed	-	1.9×10^9	-	6.3×10^6	-	2.7×10^4	-
V-1	**	2.1×10^7	9.0×10^1	1.0×10^5	63	1.3×10^3	21
V-2	4	1.2×10^5	1.6×10^4	1.2×10^4	525	1.9×10^2	142
V-3	11	ND	1.9×10^6	2.3×10^4	274	2.3×10^2	117
V-4	15	ND	1.9×10^6	NA	-	2.1×10^2	129
V-5	27	ND	1.9×10^6	1.2×10^4	525	2.4×10^2	113
V-6	42	1.8×10^5 †	-	1.1×10^4	371	8.6×10^2	31
V-7	53	ND	1.9×10^6	NA	-	2.3×10^3	12
V-8	88	ND	1.9×10^6	1.4×10^4	450	5.7×10^1	474
V-9	98	2.2×10^5 †	8.6×10^3	1.3×10^4	485	5.4×10^2	50
V-10	140	ND	1.9×10^6	1.2×10^4	525	6.8×10^2	40
V-12	240	ND	1.9×10^6	4.2×10^3	1500	NA	-
V-13	285	ND	1.9×10^6	NA	-	1.0×10^2	270
V-14	348	6.0×10^3	3.2×10^5	NA	-	1.4×10^2	192

* Na-TPB excess was 0.01M. Cs-137 is assumed to be 1×10^3 d/m/mL when not detectable (ND).

** 2 hr of mixing times.

† High Cs-137 radioactivity is probably due to contamination.

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LOW-LEVEL WASTE MANAGEMENT PROGRAM

Criteria and Site Selection of a Proposed New Burial Facility at the Savannah River Plant

Introduction

The long-term consequences of past and present SRP burial ground practices are being studied by SRL. The current view is that some remedial action, such as emplacement of additional soil cover, will be required to place the current burial ground in a satisfactory low-risk, low-maintenance "decommissioned" condition for the future. This conclusion is reached from dose-to-man studies which project long-term migration potential via vegetative uptake and from current experience which shows that significant maintenance is required to control erosion and deep-rooted vegetation. Wastes currently being buried contain <10 nCi/g TRU elements, but fission product elements such as Sr-90 and Cs-137 (both having approximately 30-year half-lives) are present and will present significant migration potential for hundreds of years.

Current estimates indicate that the present burial ground will be filled in 7 to 11 years. Employment of improved practices in the next-generation burial ground is desirable to achieve greater protection against vegetative uptake and erosion and to avoid the necessity for future remedial action to decommission the site.

Greater confinement disposal (GCD) is intended to satisfy these requirements by emplacing higher-activity waste below the root zone and farther from the erosive forces of wind and water. In addition, minimization of infiltration of surface water is considered part of GCD. The latter should virtually eliminate leaching and migration of radionuclides into groundwater. A desirable objective of GCD is to release no radioactivity outside the burial site (zero-release facility). This objective can be closely approached by the combination of the selection of a site with favorable natural conditions, such as a relatively deep water table and low-hydraulic conductivity, with modern engineering design concepts for packaging and emplacement of wastes. Lower-activity waste will continue to be disposed of using shallow land burial (SLB).

In the context of geologic time it is impossible to prevent the mobilization of any material exposed to the near surface environment of the earth. The rate of weathering decreases with

depth, but it does go on. On the scale of several hundreds of years, however, it should be possible to emplace waste in such a manner so as to delay these processes long enough for the radioactivity of the buried waste to decay to innocuous levels before it reaches the biosphere. This is the purpose of greater confinement disposal.

Three separate but related topics are addressed in this report. First, preliminary site-performance criteria are developed for greater confinement disposal. These limit the amount of radiation which a GCD facility may release to the environment. Secondly, site-selection guides are devised to assist in finding sites with a set of properties which would aid in meeting the performance criteria. These are essentially geologic and geographic characteristics which can be used to predict the suitability of potential GCD sites. The third part of this report uses the site-selection guides to evaluate 17 potential GCD sites.

Summary

Six characteristics have been identified as important in the site-selection guides. These are (1) greater distance from waste generators, (2) distance to the public, (3) distance to streams, (4) topography of the site, (5) available surface area, and (6) depth to the water table. Preliminary site evaluation using the site-selection guides on potential GCD and SLB sites at SRP has shown that at least four of the sites appear to be suitable.

Performance Criteria

Performance criteria have been identified to ensure that the objectives of the GCD site will be incorporated in the site selection and design. These criteria all relate directly or indirectly to occupational radiation dose or dose to the public and apply to both GCD and SLB. Radiation dose limits equal to or lower than those in DOE Order 5480.1 and 10CFR61 are set forth in the performance criteria. These limits are the ones presently used for all operations at the Savannah River Plant and are, therefore, considered achievable. As stated above, however, the conceptual goal of the facility is one of "near-zero release," so the actual operating and design parameters will be such that members of the work force and the general public will receive actual dose orders of magnitude less than the current limits.

1. RADIOACTIVITY IN GROUNDWATER — The concentration of radionuclides in groundwater at the disposal site boundary must not exceed those established in the National Primary Drinking Water Standards (40CFR Part 141).

2. RADIATION DOSE TO PERSONNEL — The radiation dose to any member of the public due to releases of radioactivity in the groundwater, surface water, air, soil, plants, etc., as given in Du Pont Technical Standard DPSTS-RH-W-0-1, which are more stringent than those in DOE Order 5480.1, must not be exceeded.

Body part	mrem/yr
-----------	---------

Whole body, gonads, bone marrow	10
------------------------------------	----

Other organs, gastrointestinal tract, bone, thyroid	30
---	----

3. MINIMUM DEPTH BETWEEN WASTE AND WATER TABLE — The minimum distance between the waste and the groundwater table shall be at least 10 ft. This is the interval currently maintained at the burial ground. Burial depth for GCD will be deeper, but the zone above the water table will be the same. Knowledge of variations in water table level should be based on water table data for the site.

The intent of this criterion is to limit migration of radionuclides into groundwater. Although this might be done adequately if waste is placed below the water table, by special means such as encapsulation of waste, it is considered desirable to emplace waste above the water table because of the possibility of unforeseen leaching mechanisms and the possibility of water encroachment into engineered structures. It is felt that this criterion can be met with less difficulty at SRP by placement of the waste above the water table than through engineering techniques.

4. DISTANCE BETWEEN ROOT ZONE AND WASTE — The waste should be emplaced below the root zones of plants which are indigenous to the area. At SRP it is presumed that this criteria can be met by placement of waste at a minimum depth of 16 ft. The intent of this criterion might also be achieved by other means (for example, root barriers), but such barriers would be subject to breaching at flaws. Emplacement of waste below root zones is considered feasible at prospective SRP GCD sites.
5. SURFACE WATER CONTROL — Surface topography should be such as to minimize erosion (i.e., minimum slope) and surface water should be routed so as to avoid erosion and infiltration. Precipitation should be drained away from the GCD site.

6. SUBSIDENCE — Subsidence of wastes and backfilled soil should be minimized to avoid undue maintenance of surface topography and to avoid water infiltration and potential unacceptable migration of radionuclides. To achieve this, wastes (or waste packages) must be physically stable and spaces between packages must be minimized or filled with compacted soil or other fill material.
7. POST-CLOSURE CONTROL — It is assumed for the purpose of projecting radionuclide movements that institutional control shall be maintained for 100 yr following site closure. During this period it is assumed that the site will be well maintained to prevent surface erosion, intruder entry, etc.

This post-closure criterion requires that the site be designed such that the other performance criteria will be met for an additional 200 yr. The total of 300 yr is approximately 10 half-lives of fission product elements of concern, Sr-90 and Cs-137. At the end of this period the radioactivity will have decreased sufficiently so that no significant hazard remains. The means of meeting this criterion will be (1) limitation on the inventory of radionuclides emplaced in the site, and (2) deeper burial and design of a burial unit which will prevent the waste from reaching the environment before radioactivity has decayed to innocuous levels.

Site-Selection Criteria

A. Introduction

The site-selection guides listed below are specific to the Savannah River Plant site. In general they could be applied to other prospective burial sites in humid climates. Several facts are implicit in the criteria given. The SRP site is located in the Atlantic Coastal Plain of the southeastern United States. This means that the upper 600 to 1000 ft of material consists of deeply weathered, loosely consolidated, or nonconsolidated material. The material itself is primarily sandy-clay to clayey-sand, and, in general, is quite homogeneous over the plantsite. Therefore, the emphasis given to the geologic materials at prospective burial sites is not a factor considered here. However, at other locations with more complex geology, for example in the Piedmont or the Blue Ridge, the stratigraphy of individual sites would be a major factor in determining site suitability. It is assumed that any site of sufficient size which is found to be suitable for use as a greater confinement disposal facility will also be suitable for shallow land burial, since the requirements are more stringent for GCD.

The following is a series of characteristics considered important in burial ground site selection at SRP. Each characteristic is given a weighting factor, ranging from 1 to 6, representing its relative importance in meeting the performance criteria set forth above values for each prospective burial site are assigned ratings for each characteristic which increase relative to their effect in meeting the performance criteria. The prospective sites are then evaluated by multiplying their ratings times the weighting factor for each characteristic and summing the results. The total scores should indicate the relative merit of each potential site. The characteristics are quantifiable for each site, but the break points for the ratings and the magnitude of the weighting factors are basically subjective at this time.

As an example of the procedure consider the depth to the water table. If a prospective site has a depth to the water table of 75 ft it then has a rating of 4. The weighting factor for this characteristic is 6, giving a score of 24. The scores for all characteristics are then added, giving the total points for a site.

B. Selection Criteria

Distance to Waste Generators (Weighting Factor = 6)

The volume of waste and the distance that it must be transported from each generator has an impact on both the dose to waste transport personnel and the economics of prospective burial sites. It is also desirable to avoid widespread dispersal of wastes within the plantsite. At SRP there are multiple waste generators which are widely dispersed, so a volume of waste weighted method was used to rate the sites. The distance from each potential site to each operating area was rated, then weighted by the percentage of waste produced by each operating area, and finally multiplied by the criterion weight (6). The source for each potential area is then the total of the individual ratings.

Distance, mile	Rating
<2	Excellent (4)
2.0-4.0	Very good (3)
4.0-6.0	Good (2)
6.0-8.0	Fair (1)
>8	Poor (0)

Distance to Stream (Weighting Factor = 5)

Since surface water in the humid eastern United States generally represents areas of groundwater discharge and since transport by surface water is much more rapid than groundwater

it is desirable to maximize the distance from the buried waste to surface water.

Distance, ft Rating

>2000	4
1500-2000	3
1000-1500	2
500-1000	1
<500	0

Surface Topography (Weighting Factor = 2)

The energy available for erosion by surface runoff is directly proportional to the slope of the land surface. Therefore, the lower the slope, the more slowly the land surface will erode. This factor has a low weighting because it can be altered during site engineering.

Maximum Slope, ft/ft Rating

0-0.012	4
0.012-0.025	3
0.025-0.037	2
0.037-0.050	1
>0.050	0

Available Area (Weighting Factor = 4)

The available surface area of a prospective burial ground site is important from an economic standpoint because of the effort involved in characterizing sites and the desire to limit the number of sites placed in burial ground service. The useful life of a site is determined by the area of ground available for burial.

Acres Rating

>200	4
100-200	3
50-100	2
25-50	1
<25	0

Depth to Water Table (Weighting Factor = 6)

The present reference process for greater confinement disposal calls for emplacing the waste in trenches to a depth of 15 ft while keeping the waste at least 10 ft above the water table. Therefore, for a 20-ft waste zone the depth from the surface to water must be at least 45 feet. Since transport and leaching are much slower in the unsaturated zone, depths greater than 50 ft are rated more highly.

Depth, ft	Rating
>80	5
70-80	4
60-70	3
45-60	2
<45	0

Distance to the Public (Weighting Factor = 6)

Placement of the GCD facility at greater distance from the public has two beneficial effects. The farther the facility is from public water supplies the longer time is available for radio-nuclides to decay before they have any possibility of interacting with the water supplies. If the site is remote from public access then the probability of accidental exposure to a member of the public during burial operations is minimized.

Distance, mile	Rating
>4	4
3-4	3
2-3	2
1-2	1
0-1	0

Preliminary Site Evaluation

The site-selection criteria were applied to 17 potential greater confinement disposal and shallow land burial sites. Figure 41 is a map showing the location of these sites. Table 52 is a summary of the rating process, listing the sites in order of their total score. This table shows that sites B and K are the two highest rated sites, followed by sites L and I, which have virtually the same rating. Therefore, these are the sites which should be examined in greater detail to select the location of the new burial facility.

FIGURE 41. Potential Intermediate Depth Burial Sites

TABLE 52**Sites Ranked by Ratings Score**

<u>Site</u>	<u>Score</u>
B	97
K	94
L	88
I	87
P	84
Q	83
E	80
D	75
C	72
F	71
J	66
A	60
H	55
M	55
N	45
O	40
G	37

The area, distance to surface water, and depth to the water table for each site are given in Table 53.

Volume and Curie Definition of Waste for a Proposed New Burial Facility at SRP

In addition to site-selection work described in the previous article, it is essential that the volume and types of waste to be emplaced in the new facility be defined. Currently, we are investigating how to provide greater confinement disposal (GCD) for the most significant (i.e., containing most of the radionuclides) fraction of SRP waste. The GCD fraction must be identified and characterized. The next step is to characterize the GCD waste physically, and then develop an improved burial concept based on greater confinement.

TABLE 53

Site Characteristics

Site	Approximate Area, acres	Distance to Closest Stream, ft	Distance to Closest Marsh or Standing Water, ft	Approximate Depth to Water Table, ft
A	135	2500	>2000	>40
B	300	2500 1000**	>2000	55-65*
C	90	500	>2000	55*
D	80	1500 200**	>2000	40-70*
E	185	2500†	>2000	45-65**
F	115	1500	on site	40-60*
G	85	1500†	>2000	40-50*
H	135	3500	1800	50-60
I	215	2000 500**	>2000	40-85*
J	220	500-2000 500**	>2000	50
K	220	2000 300**	>2000	50
L	100	1700†	>2000	40-80*
M	160	2000 1500**	>2000	-
N	210	1500 500**	>2000	-
O	225	1000 500**	>2000	-
P	240	5000 1000**	>2000	60-80*
Q	255	5000	>2000	45-90*

* Site-specific information available.

** Indicates distance to intermittent stream.

† At site boundary.

The fraction of the waste consigned to contained shallow land burials will contain very little radioactivity, and thus it can continue to be buried by the low-cost trench method. However, incineration of this fraction will concentrate the radioactivity and reduce the volume, perhaps making it practical to bury it in the GCD facility.

Discussion

The COBRA records provide a basis for defining the waste. Each waste receipt is accompanied at the burial ground with a burial slip in which the volume and curies (or grams) of radionuclides is given, along with whether it is combustible or non-combustible. Also specified is the "variety" of waste that is buried. The "variety" descriptor identifies the type of radionuclides present. There are ten main "varieties" of waste currently being buried in trenches at SRP. A radioisotope of interest has been identified for each "variety." In some cases the isotope of interest is the same as the name of the "variety" but in others it is not.

The curies of the isotopes of interest were determined by recalling curies for the "variety" in question from COBRA burial data, and then calculating the curies of the isotope of interest (if not the same). The calculations were completed for each waste receipt by accessing COBRA/ADABAS files. The concentration ($\mu\text{Ci/cc}$) of the radionuclide of interest was calculated, and the data were then sorted on the basis of concentration. The methods used for determining the curies of the "variety" (i.e., the input to COBRA) and the methods used for determining the curies of the isotopes of interest (see Table 54 below) were not exact but do serve as a reasonable basis for this study.

The study was based on a three year period, 1979 through 1981. This period was chosen because the data are the most recent available, and it is expected that future waste generation will be similar. Results from this study indicate that >99% of the activity in the waste is contained in about 5% of the volume. Table 55 shows the volumes for low-level waste. The curies or grams associated with the volumes of low-level waste are given in Table 56.

With the use of the beta-gamma incinerator, the SLB fraction of the waste could be reduced from 1,847,000 ft^2 to about 666,000 ft^2 , assuming ash would still be disposed of by SLB. If the expected 62,000 ft^2 of ash is disposed of by GCD, only the noncombustible fraction (604,000 ft^2) would be disposed of by SLB, and the GCD fraction would be increased by up to 62% (110,000 to 162,000 ft^2). Incineration can have little effect on the curie

TABLE 54

SRP Waste Categories

<u>"Variety"</u>	<u>Symbol Used in COBRA</u>	<u>Isotope of Interest and Method for Determining Quantity</u>
Cobalt-60	Co	^{60}Co . Taken directly from COBRA records as "Variety" 1 or "Variety" 2.
Cesium-137	Cs	^{137}Cs . Taken directly from COBRA records as "Variety" 1 or "Variety" 2.
Fission Products	FP	^{90}Sr . Assumed to be 2% of total fission product curies, based on estimate of average content of this waste type.
Induced Activity	IA	^{63}Ni . IA curies is assumed to be ^{60}Co . ^{63}Ni is assumed to be $0.15 \times ^{60}\text{Co}$ curies, based on DP-1519.
Strontium	Sr	^{90}Sr . Taken directly from COBRA records as "Variety" 1 or "Variety" 2.
Tritium	87	^3H . Taken directly from COBRA records as "Variety" 1 or "Variety" 2.
Enriched Uranium	20	Total alpha. Curies based on isotopic split of 95% ^{235}U , 4.3% ^{238}U , and 9.71% ^{234}U .
Natural Uranium	81	Total alpha. Curies based on isotopic split of 99.28% ^{238}U , 0.72% ^{235}U , and 0.0054% ^{234}U .
Plutonium-238	83	^{238}U . Taken directly from COBRA records as "Variety" 1 or "Variety" 2.
Plutonium-239	50	^{239}Pu . Taken directly from COBRA records as "Variety" 1 or "Variety" 2.

TABLE 55. Volumes of Low-Level Waste, ft³

	<u>Combustible</u>	<u>Noncombustible</u>	<u>Total</u>
SLB	1,243,000	604,000	1,847,000
GCD	22,000	78,000	100,000

TABLE 56

Curies or Grams Associated with Volumes of Low-Level Waste

	<u>Combustible</u>	<u>Noncombustible</u>	<u>Total</u>
SLB			
Curies	70	1,150	1,120
Grams ²³⁸ Pu	0	0	0
Grams ²³⁹ Pu	0.8	0.1	0.9
Grams natural uranium	0	5,180	5,180
Grams enriched uranium	305	26,195	26,500
GCD			
Curies	44,470	306,910	351,380
Grams ²³⁸ Pu	0	0.07	0.07
Grams ²³⁹ Pu	39	6	45
Grams natural uranium	0	7,000	7,000
Grams enriched uranium	46	136,324	136,370

split because very few curies are present in combustible waste assigned to SLB (Note: The GCB combustible waste is assumed to be too radioactive for incineration).

The COBRA files have also provided information on the source (generator) and type of waste included in the GCD fraction. This information is being used in discussion with the generators to study the possibility of segregating waste at the source to improve packaging for stabilization and containment.

Comparison of 10CFR61 Requirements with Current and Proposed SRP LLW Practices

Introduction

This report outlines the revised final 10CFR61 requirements for land disposal of LLW, published 10/23/82, and compares 10CFR61 requirements with current SRP practices and with the design and operation of a new greater confinement disposal (GCD) facility. There are seven subparts and 53 sections of 10CFR61, as listed below:

<u>Subpart</u>	<u>No. of Sections</u>
A. General Provisions	8
B. Licenses	19
C. Performance Objectives	5
D. Technical Requirements for Land Disposal	10
E. Financial Assurances	3
F. Participation by State Governments and Indian Tribes	4
G. Records, Reports, Tests, and Inspections	4

10CFR61 is concerned with licensing of commercial facilities designed for disposal of low-level waste in near-surface disposal facilities. The terms "waste" and "near-surface" disposal are defined below.

- The term waste, as used in 10CFR61, has the same meaning as low-level waste has in the Low-Level Policy Act — i.e., "radioactive waste not classified as high-level radioactive waste, trans-uranic waste, spent nuclear fuel, or by-product material (uranium or thorium tailings and waste)."

- Near-Surface disposal facility means a land disposal facility in which radioactive waste is disposed of within the upper 30 meters of the earth's surface. A geologic repository, as defined in Part 10CFR60, is not considered a land disposal facility.

Summary

10CFR61 contains regulations concerned with preoperational planning and site selection, design and operation, and postoperational practices. Present SRP practices comply with most of the rules, while proposed GCD practices would comply with all of them. Table 57 provides this comparison on an item-by-item basis.

In brief, there are seven rules in 10CFR61 that are significantly different from current SRP practices. These are:

1. No cardboard or fiberboard containers be used.
2. Waste be segregated into three classes — A, B, and C, according to the concentration of various radionuclides.
3. Class B and C wastes be stabilized.
4. Class C waste be provided with an intruder barrier, or be buried 5 meters deep.
5. No groundwater intrudes to the waste.
6. A site closure plan be prepared prior to operation.
7. Individual trenches be closed according to the closure plan immediately after filling.

Improved practices, such as use of the LLW incinerator, coupled with the new GCD facility will meet all of these rules.

Discussion

Table 57 lists the technical requirements of 10CFR61 and compares these with current burial ground practices and with proposed practices in a new SRP greater confinement disposal (GCD) facility. While the proposed facility would be controlled by DOE, the intent would be to make the operation compatible with all of the technical requirements of 10CFR61.

10CFR61 divides a low-level waste into three classes, according to the concentration of long- or short-lived radionuclides. These classes are compared with a proposed GCD classification in Table 58.

TABLE 57

10CFR61 Technical Requirements and Comparisons

<u>10CFR61 Section and Requirements</u>	<u>Do Present BG Practices Comply</u>	<u>Will GCD Comply</u>
<u>61.7 Concepts</u>		
- Site characteristics should be evaluated for at least a 500-yr time frame	No	Yes
- Class A waste must be segregated from Class B and C waste, or else stabilized	No	Yes
- Class B and C waste forms or containers must be stable for over 300 yr	No	Yes
- Future inadvertent intruders should be protected by institutional controls and intruder barriers	*	Yes
- Institutional control of access to site is required for up to 100-yr	*	*
- Intruder barriers (for Class C waste) must have an effective life of 500 yr	No	Yes

61.12 Technical Information

The following have to be described:

- Natural and demographic characteristics	Yes	Yes
- Design features of facility	Yes	Yes
- Design basis, natural events	Yes	Yes
- Description of codes and modeling	Yes	Yes
- Site closure plans	Yes	Yes
- Known natural resources	Yes	Yes
- Q/A programs	Yes	Yes
- Radiation safety	Yes	Yes
- Environmental monitoring plans	Yes	Yes
- Administration procedures	Yes	Yes

61.13 Technical Analysis

Must include:

- Release pathways for air, soil, groundwater, surface water, plant uptake, exhumation	Yes	Yes
- Analysis of protection against future inadvertent intrusion	No	Yes
- Assessment of exposure of operating personnel	Yes	Yes
- Analysis of long-term stability	No	Yes

TABLE 57, Contd

<u>10CFR61 Section and Requirements</u>	<u>Do Present BG Practices Comply</u>	<u>Will GCD Comply</u>
<u>61.41 Population Exposure</u>		
- Concentrations of radioactive material which may be released to the general environment must not result in an annual dose exceeding 25 mrem whole body, 75 mrem thyroid	Yes	Yes
<u>61.50 Technical Requirements</u>		
- Site must be capable of being modeled	Yes	Yes
- No effect from projected population growth	Yes	Yes
- Natural resources to be avoided	Yes	Yes
- Free from 100-yr flood	Yes	Yes
- Minimize surface runoff	Yes	Yes
- Groundwater intrusion to waste must not occur	No	Yes
- Groundwater must not be discharged to surface	Yes	Yes
- Avoid areas of tectonic processes	Yes	Yes
- Avoid areas of erosion, landsliding, etc.	Yes	Yes
<u>61.51 Technical Requirements</u>		
- Site design directed toward long term isolation and avoidance of active maintenance after closure	No	Yes
- Design covers to minimize water infiltration	No	Yes
<u>61.52 Facility Operation</u>		
- Class A wastes must be segregated from Class B and C, or stabilized	No	Yes
- Class C wastes must be disposed of so that top of waste is at minimum depth of 5 meters or with intruder barriers designed to last 500 yr	No	Yes
- All wastes must be disposed of to:		
- Maintain package integrity		
- Minimize voids in and between packages		
- Keep surface dose ALARA	No	Yes
- Boundaries of waste emplacement must be marked	Yes	Yes

TABLE 57, Contd

<u>10CFR61 Section and Requirements</u>	<u>Do Present BG Practices Comply</u>	<u>Will GCD Comply</u>
- Buffer zone must be maintained between waste and site boundary large enough to do environmental monitoring	Yes	Yes
- Closure and stabilization measures must be carried out as each disposal unit (e.g., trench) is filled	No	Yes
- Uncontaminated waste shall not be disposed of at the site	**	**
<u>61.53 Environmental Monitoring</u>		
- Preoperational monitoring program	No	Yes
- Operational and postclosure monitoring programs	Yes	Yes
<u>61.55 Waste Classification</u>		
- Class A waste must meet minimum requirements of Section 61.56	No	Yes
- Class B waste must meet both minimum and stability requirements	No	Yes
- Class C waste must meet minimum, stability, and inadvertent intruder requirements	No	Yes
- Radionuclide limits	(see Table 58)	
<u>61.56 Stability</u>		
- Minimum requirements are:		
- No cardboard or fiberboard boxes	No	Yes
- Liquid wastes in absorbent to absorb 2x volume of liquid	Yes	Yes
- Free liquid less than 1% of volume	Yes	Yes
- No explosive reactions	Yes	Yes
- No waste that generates toxic gases	Yes	Yes
- No pyrophoric waste	Yes	Yes
- Gaseous waste pressure to be less than 1.5 atm, less than 100 Ci per container	†	†
- No untreated hazardous, biological, pathogenic, or infectious materials	††	††
- Stability requirements are:		
- Waste must have structural stability	No	Yes
- Void spaces within the waste and between the waste and its package must be as low as possible	No	Yes

TABLE 57, Contd

<u>10CFR61 Section and Requirements</u>	<u>Do Present BG Practices Comply</u>	<u>Will GCD Comply</u>
<u>61.57 Labeling</u>		
- Each package to be labeled Class A, B, or C	No	Yes
<u>61.59 Institutional</u>		
- Land must be owned in fee by the Federal or State government	Yes	Yes
- Institutional controls may not be relied upon for more than 100 yr after closure	*	*

* Requires a DOE Order.

** Lack of a de minimus specification and an approved detection method means that we are now, and will continue to add uncontaminated waste to the burial ground.

† No gaseous wastes are permitted.

†† Not addressed at this time.

TABLE 58

Maximum Concentration (Ci/m³) of Radionuclides for
Classes A, B, and C Wastes

Radionuclide	Half-Life, yr	A	B	C*	Proposed GCD Classification
<u>Long-lived**</u>					
C-14	5730	10.8	-	8	†
C-14††	5730	8	-	80	†
Ni-59††	80,000	22	-	220	†
Nb-94††	20,000	0.02	-	0.2	†
Tc-99	210,000	0.3	-	3	†
I-129	16,000,000	0.008	-	0.08	†
Alpha	>5	10‡	-	100‡	100‡‡
Pu-241	14.3	350‡	-	3500‡	†
Cm-242	163 day	2000‡	-	20,000‡	†
<u>Short-lived**</u>					
Total <5 yr	<5	700	§	§	†
H-3	12.3	40	§	§	5
Co-60	5.3	700	§	§	150
Ni-63	92	3.5	70	700	0.3
Ni-63††	92	35	700	7000	0.3
Sr-90	29	0.04	150	7000	150
Cs-137	30	1	44	4600	40

* Waste exceeding these limits is not acceptable for near-surface disposal.

** Separate rules exist for mixtures of short-lived and long-lived radionuclides.

† Limits are to be determined, but will be equal to or less than Class C concentrations.

†† In activated metal.

‡ Activity in nanocuries per gram.

‡‡ Activity in nanocuries per gram. Waste with concentrations greater than 100 nanocuries per gram alpha would be classified as transuranic (TRU) waste.

§ There are no limits established for these radionuclides in Class B wastes. Wastes with these nuclides shall be Class B unless other radionuclides determine the waste to be Class C.

Solid Waste Incineration

Introduction

The solid waste incineration facility for testing (SWIFT) has operated for 400 hr and burned 25,000 lb of waste. After facility start-up, three week-long campaign runs were conducted. During these tests, the incinerator processed a wide variety of simulated (nonradioactive) waste at rates of 300 to 400 lb/hr. The resulting ash reduced the storage volume of the waste by a factor of 20. The off-gas treatment successfully cooled the exhaust from 1000°C to 180°C and filtered particulates from the gas stream. Cleaned off-gas was released to the atmosphere at 100°C. Monitoring of the exhaust gas determined that pollutant and particulate concentrations were below South Carolina and Federal limits.

Summary

The research program for the SWIFT was developed to verify the system's operation under typical conditions expected in a production incinerator. The waste components — rubber, polyethylene, polyvinyl chloride (PVC), and cellulose — were burned during campaign runs to determine incineration parameters. The off-gas treatment system was monitored during operation to ensure that the gases were cooled, filtered, and released to the atmosphere at a temperature above the dew point.

The process control strategy required development during initial operation. Feedback loops designed for the system were adjusted and modified to meet the control requirements of the process. The controlled parameters were the incinerator temperature and pressure, the spray quench outlet temperature, and the stack temperature. New components and modified control schemes were tested during the campaign runs.

Products of the incineration process were measured and analyzed. The ash from the incinerator and the fine particulates removed by the spray quench and the baghouse were quantified and tested for chemical content and particle size. Baghouse ash was also tested for explosibility and was inert. The filtered exhaust vented through the stack was analyzed. Concentrations of CO₂, O₂, N₂, HCl, CO, SO₂, NO_x, and particulates were determined.

Background

An incineration development program is in progress at the Savannah River Laboratory. Currently generated solid waste contaminated with low levels of beta-gamma emitters is boxed in 24-in.

and 15-in. cartons and buried in trenches in an onsite burial ground. Incineration has been proposed as the primary method of volume reduction, since 60% of the solid waste generated is combustible.

After a survey of the incineration processes being tested and demonstrated at nuclear installations, a full-scale facility based on a two-stage, controlled-air incinerator and a dry off-gas system was designed. The test unit, called the solid/solvent waste incineration facility for testing (SWIFT), is being used to evaluate uncertainties in design including the following areas:

- Flexibility to burn a wide variety of materials.
- Operability of a dry off-gas system.
- Effectiveness of process control for temperature and pressure.

During Phase I of the program, the SWIFT is being tested with nonradioactive materials of similar composition to waste sent to the burial ground. After the equipment has been successfully demonstrated, Phase II will begin and the process equipment will be relocated to H Area and tested with actual suspect and low-level radioactive wastes.

The SWIFT was constructed at SRL during FY-1981, and start-up was achieved in October 1981. About 2000 lb of computer paper were burned during this run (Solid Run 1).

Discussion

Facility Description and Operation

Processing steps in the SWIFT are waste feeding, incineration, ash packaging, and off-gas cleanup. Figure 42 shows a conceptual flowsheet for solid waste burning. The completed incinerator installation outside 774-A is shown in Figure 43.

Incineration

A campaign run is composed of periods of heatup, waste burning, idling, cooldown, and shutdown. During heatup, diesel fuel burners in the combustion chambers increase the temperature to 600°C at a staged rate of 50°C/hr. Automatic controllers turn the burners on and off to maintain the desired set points. Heatup is followed by two to three days of waste burning.

FIGURE 42. Conceptual Flowsheet of the SWIFT

FIGURE 43. Front View of the SWIFT Installed Outside of 774-A

During waste feeding a horizontal ram feeder pushes boxes of solid waste into the lower chamber of the incinerator. The ram pushes the box onto a fire step at the front of the chamber, where waste components begin to volatilize. Air flow is reduced to reduce the amount of ash entrained in the exhaust gas while the chamber contents are disturbed.

Subsequent boxes push partially burned material from the fire step into the hearth area. Underfire air is introduced through air ports on the side of the hearth. This air flow is low to avoid excessive ash entrainment, but flow is high enough to partially oxidize carbon and hydrogen in the waste. Ash is pushed through the hearth by an internal ram. The ash in the drop is periodically displaced for removal by another ram.

Products of partial combustion (CO , H_2 , hydrocarbon radicals) are completely oxidized in the secondary chamber. Online analyzers measure the oxygen concentration in each chamber to ensure that overall incinerator conditions are air-rich.

Waste feeding is continued for 10 to 12 hr during day shifts and is succeeded by an idling period over the midnight shift. Figure 44 shows an example of incineration during a campaign run. The burners modulate during the waste burning and idling periods, since the waste becomes a supplementary fuel. Figure 45 shows typical oil consumption during a campaign. The slope of the line indicates the average oil burning rate of 15 gal/hr. For isobaric combustion of fuel oil, the heat of reaction is 131,000 Btu/gal. Using this value, the total heat input of the burners is 1.96×10^6 Btu/hr.

At the conclusion of a campaign, the ash and partially burned waste in the chamber is allowed to "burn down" for two hours. The burners and blowers then deactivate, and the incinerator begins to cool. Heat transfer through the ceramic and metal walls is slow, reducing the temperature approximately 25°C/hr . Figure 46 shows a plot of temperature versus time for the shutdown period. When the incinerator cools to 150°C , the process utilities and the control equipment are shutdown.

Off-Gas Equipment Operation

During the initial startup, the baghouse was bypassed, and only one induced draft blower was operated. In the subsequent runs the off-gas treatment was refined to adequately cool and filter the exhaust.

FIGURE 44. Run 4 Box Burning Data

FIGURE 45. Run 4 Oil Consumption

FIGURE 46. Temperature Profile During Shutdown

The spray quench uses an air-atomized water spray to reduce the temperature of the incinerator exhaust from 900°C to 180°C. Original process calculations were made for cooling the gas to 150°C. When this was tested, water condensed at the bottom of the quench and drained through the discharge airlock. Increasing the outlet temperature to 180°C evaporated all the water, eliminating the drip.

The quench water is activated when the incinerator temperature reaches 300°C during heatup. A minimal amount of water is needed and is controlled manually until the temperature reached 500°C. At this point, the controller can be turned to automatic mode.

Figure 47 shows the increase in pressure drop across the bag-house as waste is burned and the decrease after a cleaning cycle. The particulates are removed from the bags with a 90 psi air pulse that vibrates the filters and shakes the dust into the hoppers below the structure.

The blower capacity is damped by a 24-in. butterfly valve located upstream in the process line. The valve is automatically controlled to maintain a 1-in. H₂O pressure in the primary.

The blowers discharge the exhaust through a 35-ft stack to the atmosphere. The flow through the stack varies in a range of 5000 to 7500 ft³/min. The stack exhaust is sampled and analyzed during campaign runs.

Materials Burned

The SWIFT has burned a variety of materials, including computer paper, cotton fiber, cardboard, polyethylene bags, PVC sheeting, and red rubber. The amount and type of material burned during each run is shown in Table 59. In addition to feed composition, the box size and packing density were also varied, as shown in Table 59.

Cellulosics, in the form of computer paper and cotton fiber, were burned during Run 2. This material was chosen for the initial and second burning campaign because it is easier to burn than plastic materials. While operating with cellulose, the burning rate was maintained at 375 to 400 lb/hr during two 9-hr periods. A total of 6670 lb of computer paper and cotton fiber was incinerated.

Waste mix was incinerated during Run 3. This mix contained red rubber and polyethylene, in addition to cardboard, cotton fiber, and computer paper. Table 60 shows the composition of the waste mix. During three days of burning, 10,520 lb of waste mix were incinerated at an average rate of 300 to 325 lb/hr.

FIGURE 47. Run 3 Baghouse ΔP Rise

TABLE 59

Summary of Solid Waste Burning Campaigns

Run No.	Date	Duration of Incineration, hr	Total Amount of Waste Burned		Waste Components	Box Size, in.	Packing, lb/cu ft
			Weight, lb	Volume, cu ft			
Solid 1	10/19-10/23/81	4	2140	335	Comp. paper	15,24	6
Solid 2	11/30-12/4/81	20	6670	987	Comp. paper Cotton fiber	15,24	5
Solid 3	1/25-1/29/82	38	10520	1531	Waste mix w/o PVC	15,24	3,5,8
Solid 4	4/4-4/8/82	21	6000	1072	Red rubber Waste mix, PE	15,24	4,6,10
Total		83	25330	3925			

TABLE 60

Composition of Waste Mix

• Without PVC (Run 3)

Component	Wt %
Polyethylene	23
Red rubber (SBR)	19
Cotton fiber	29
Computer paper	29

• With PVC (Run 4 and Long-Term Campaign)

Polyethylene	23
Red rubber (SRB)	19
Polyvinyl chloride	8
Cotton fiber	25
Brown reinforced paper	25

During Run 4, boxes containing only red rubber (SRB) were burned to test the capacity of the incinerator. SRB has the highest oxygen (and air) demand of the waste components. Both 15-in. and 24-in. boxes with packing densities of 4, 6, and 10 lb/ft³ were burned without reaching the limit of the incinerator air supply. Nearly 3,000 lb of red rubber were processed in 9 hr for an average burning rate of 330 lb/hr.

Test burns with waste mix that contained polyvinyl chloride were also conducted during Run 4. A total of 2,050 lb of waste mix containing 160 lb of PVC was incinerated. The HCl generated was partially neutralized by a solution of Na₂CO₃ pumped into the spray quench.

Run 4 concluded with burning a small amount (320 lb) of boxes containing only polyethylene. These tests were being conducted since PE has a high-heat release rate, but were not completed since ash removal problems had created a buildup of ash in the primary combustion chamber and waste feeding became impossible.

Temperature Control

The theory of controlled-air incineration is to use the amount and location of combustion air to maintain operating temperatures. In the primary chamber, the airflow is low to avoid excessive ash entrainment. The small amount of "underfire" air creates a sub-stoichiometric environment where the waste pyrolyzes and only partially oxidizes. To control temperature in this chamber, a modulator on the combustion blower increases the amount of air to cause more pyrolysis gas to oxidize and release the heat of combustion. The temperature in the primary can increase 100°C in less than 2 min when the combustion air is increased.

The secondary chamber operates under excess air conditions to ensure complete oxidation of pyrolytic gases. The blower is modulated to decrease temperature by increasing the amount of air, which acts as a heat sink.

Excessive air inleakage caused by inducing a vacuum in the combustion chamber decreases the effectiveness of the controlled-air strategy. The air leaks into the primary around the charging door and various flanges on the chamber. The additional air causes more pyrogenic gas to oxidize and this in turn increases the temperature from 800°C up to 1100°C. Excursions have reached 1200°C. Steam at 120°C was tested as a cooling medium, but addition of up to 100 lb/hr did not reduce the temperature.

The 400 lb/hr feed rate is maintained until the primary chamber temperature increases to 1100°C. Feed rates mentioned earlier are generally less than 400 lb/hr since waste loading is usually stopped or reduced to prevent excessively high temperatures. Figure 48 shows how the temperature increases as more waste is burned and does not decrease until waste feed is discontinued.

Ash Removal

Ash is moved through the incinerator by a series of rams and is eventually displaced into a 55-gal drum. The internal ram pushes the partially burned material along the hearth. The ash reaches the end of the hearth in 4 to 8 hr and falls through an opening in the primary chamber into the ash drop.

The residue cools in the drop area and is periodically pushed through a rectangular duct into the discharge hopper. After four or five strokes of the pusher, small amounts of ash move to the end of the duct, but successive cycles of the ram do not push the ash further. The ash pusher stroke is only half the length of the duct, and a large block develops at the end of the pusher's travel. The ram was modified by adding I-beams to extend its effective length by 24 in. Even with the extenders the ram travel was not long enough to push the ash through the duct. A new ram nearly 3-ft longer is being fabricated and will be tested during another campaign run.

PVC Incineration

Burning halogenated hydrocarbons, such as polyvinyl chloride, releases halogen into the incinerator exhaust. In the SWIFT, the chlorine released reacts with hydrogen to form HCl. The acid can cause corrosion problems in the spray quench, baghouse, and stack ductwork, since these items are constructed from carbon steel. To reduce the amount of HCl in the off-gas, a process of spray drying soda ash (Na_2CO_3) in the quench was tested. The HCl is expected to react according to the following equation:

The amount of carbonate needed for stoichiometric and 100% excess conditions was calculated, but only stoichiometric conditions were tested because of time restraints. Soda ash is added to the quench supply to make a 0.1M solution. At a quench solution rate of 4.5 gal/min, the amount of carbonate input into the quench is 27 lb/hr.

Efficiency of the neutralization process is a function of the reaction rate, the amount of surface area available for reaction,

FIGURE 48. Effect of Feed Rate on Primary Temperature

and the quality of mixing between the gas, liquid, and solid phases. Studies with spray dryers have shown efficiencies of approximately 90%.

During PVC incineration gas samples were taken before and after carbonate was added to the quench tank. Much of the chlorine is staying in the incinerator or baghouse. Before carbonate addition, 12% of the HCl generated reached the stack; after neutralization, the amount of HCl in the exhaust is 8% of that generated. The HCl is reacting with basic components in fly ash accumulated on the baghouse. Chlorine is present in the baghouse ash.

Incineration Products

Incinerator and Baghouse Ash. Table 61 shows the amounts of ash produced during the campaign runs. Volume and mass reduction ratios have varied for different waste compositions, ranging from 22 to 15 for volume and 12.5 to 3.4 for mass. The volume reduction numbers represent a decrease of storage space for the waste equivalent to packing the ash of nineteen 24-in. or seventy-six 15-in. boxes into one 55-gal drum. The mass reduction is low as compared to other incineration processes.¹⁷ Laboratory tests of ashing red rubber were burned. The mass reduction ratio for Run 4 is 3, as compared with ratios of 12 and 9 for Runs 2 and 3.

Samples of primary chamber ash were analyzed by x-ray fluorescence and neutron activation to determine the elemental composition. The components found in the ash include calcium, titanium, iron, and zinc as shown in Table 62. The metals exist mainly in the oxide form, since the ash stays in a high-temperature oxidizing environment.

The carbon content of the ash ranges from 29 to 16 wt %. Low carbon burnout is due in part to insufficient residence time in the hearth of the incinerator. High carbon content in the ash is also caused by air inleakage in the primary.

Two-stage combustion minimizes the amount of ash entrained in the off-gas. A small amount of ash is entrained in the gas leaving the primary. The heavier fraction of the fly ash falls out in the secondary, but fine particulates are carried through the off-gas ductwork and spray quench into the baghouse. Samples of baghouse ash contain zinc, calcium, silicon, and aluminum, with 4 to 19 wt % carbon (see Table 62). The particulates are a fine, homogeneous powder, with a mean particle diameter of 14 microns.

TABLE 61

Summary of Ash Collected from Incinerator Runs

Run No.	Incinerator Ash Produced Weight, Volume		Baghouse Ash Produced Weight, Volume		Ash Production Rate, lb ash/lb waste	Ash Fraction Reaching Baghouse	Reduction Ratio	
	lb	cu ft	lb	cu ft			Weight	Volume
Solid-1	91	7	-	-	0.04	-	22	45
Solid-2	496	37	37	7	0.07	0.07	12	22
Solid-3	1114	66	126	22	0.11	0.11	9	17
Solid-4	1561	58	175	15	0.26	0.10	3	15
Total	3263	268	338	44				
Average					0.13	0.09	7	18

TABLE 62

Composition of Incinerator and Baghouse Ash Samples

Component	Incinerator Ash Run 3, wt %	Baghouse Ash* Run 3, wt %	Baghouse Ash** Run 4, wt %
Al ₂ O ₃	30	23	5
BaO	12	-	-
CaO	5	5	1
C	16	6	3
FeCl ₃	0.1	-	-
Fe ₂ O ₃	2.9	4	2
NaCl	1	7	74
SiO ₂	29	55	7
TiO ₂	2	-	-

* No neutralization in use.

** Neutralization with Na₂CO₃ in use.

Off-Gas Analysis. Incinerator exhaust is released to the atmosphere through a 35-ft stack. A sampler is installed at the 26-ft level, which is 16-ft downstream of bends and elbows (8 pipe diameters) and 10-ft (>2 pipe diameters) upstream of the stack exit. Sample gas is pulled through a filter and a series of bubbler-impingers, as shown in Figure 49.

Flow Rate. A reverse-type Pitot tube is used to measure the velocity pressure differential in the gas stream. The pressure is measured at several points across the stack diameter (ΔP), and is converted to a gas velocity by the following equation:

$$v = K_p C_p (\Delta P)^{1/2} \sqrt{\frac{T_s}{M_s P_s}}$$

$$K_p = 85.49 \frac{\text{ft}}{\text{sec}} \left[\frac{(\text{lb/lb mole})(\text{in. Hg})}{(^{\circ}\text{R})(\text{in. H}_2\text{O})} \right]^{1/2}$$

(units conversion factor)

$C_p = 0.84$ (Pitot tube coefficient)

$T_s = 672^{\circ}\text{R}$ (stack temperature)

$M_s = \text{Appx. } 29 \text{ lb/lb mole}$ (stack molecular weight)

$P_s = 29.93 \text{ in. Hg}$ (stack pressure)

$v = \text{ft/sec}$ (stack velocity)

Figure 50 shows the velocity profile as a function of normalized distance from the stack center. The flow is in the turbulent range ($N_{Re} = 2.27 \times 10^5$), with variations of up to 20% in the velocity across the diameter. Using the cross-sectional area ($A = 3.141 \text{ ft}^2$), the volumetric flow rate is calculated by:

$$V = v \cdot A \quad V - \text{ft}^3/\text{min} \text{ (volumetric flow)}$$

V is then averaged over the stack diameter and multiplied by the gas density ($\rho_{\text{gas}} = 0.055 \text{ lb/ft}^3$) to determine a mass flow rate. Table 63 gives the velocity, volumetric flow, and mass release rate for typical operation.

Chemical Composition. Major constituents in the stack gas are N_2 , O_2 , H_2O , and CO_2 . The component concentrations are converted to a wt %, which is then used to determine a release rate (lb/hr) in the stack gas. Table 64 shows the component release rate for

FIGURE 49. Stack Sampling System

FIGURE 50. Velocity Profile of SWIFT Stack

TABLE 63

Typical Exhaust Gas Flow Rates

Traverse Point	Distance from Stack Wall, in.	Normalized Distance D-12/12	Pressure Drop, in. H ₂ O	Velocity, ft/sec
1	2.5	-0.21	0.16	25.5
2	4.7	-0.39	0.15	24.7
3	7.8	-0.65	0.20	28.5
4	16.2	+0.35	0.18	27.0
5	19.3	+0.61	0.27	33.1
6	21.5	+0.79	0.17	26.3
7	23.3	+0.94	0.17	26.3

Average velocity = 27.8 ft/sec
 Average volumetric flow = 5240 ft³/min
 Average mass flow = 17,290 lb/hr

TABLE 64

Calculated and Measured Stack Gas Composition
Major Components

	Calculated Data	Measured Data
Temperature	90°C	100°C
Waste feed rate	380 lb/hr	400 lb/hr
Volumetric flow rate	7060 ACFM	6660 ACFM
Mass flow rate	25,740 lb/hr	20,890 lb/hr
Components		
O ₂	4066 lb/hr	3610 lb/hr
N ₂	16,830 lb/hr	14,320 lb/hr
H ₂ O	3813 lb/hr	2130 lb/hr
CO ₂	1032 lb/hr	835 lb/hr
HCl	4 lb/hr	1 lb/hr

waste mix operation and compares the rates with flows calculated during the design phase of the project. The calculated numbers are about 20% higher than measured data. This is due to the following differences in the basis of each set of values:

1. Original estimates of the carbon content in the waste were higher than the actual value.
2. Heat loss through equipment walls was not taken into consideration for the calculated values. The amount of water needed for cooling is, therefore, higher.
3. Air dilution was originally calculated for a stack temperature of 90°C. The calculated amount of air needed was higher since the stack gases were exhausted at 100°C.
4. Calculated data use a basis of 380 lb/hr continuous feed rate, but measured data are taken while running at an average rate of 400 lb/hr (eight 50-lb boxes loaded each hour).

The material balance for the entire incineration system is shown in Table 65, and the numbered streams are shown in the flow-sheet (Figure 51). Using the effluent release rates and data provided by the online oxygen analyzers, a component balance on O_2 is calculated. This is used for determining the amounts of dilution air and combustion air supplied to the incinerator. The remainder of the material balance is developed from these flows, based on component and energy balances.

Several pollutants are present in the exhaust gas at levels below State and Federal limits. The bubbler solutions for the stack sampler are analyzed by ion chromatograph for SO_4^{2-} and NO_3^- , and by neutron activation for Cl^- . The GC method described earlier is used to determine CO in the gas.

Table 66 shows the release rate of the pollutants for different burning conditions. SO_2 concentration is highest when red rubber is burned because of the sulfur used in polymer processing. Sulfur in the fuel oil accounts for 10% of the SO_2 produced. Production of NO_x increases with higher temperatures and percent excess air. CO is generated in the primary chamber, but is oxidized to CO_2 in the secondary. Concentrations of CO are below the detection limits for gas chromatograph (<0.1%).

To compare the pollutant concentrations with EPA limits for air quality, the release rates are multiplied by a dilution factor derived from experimental data at SRL. The result is the expected concentration of the pollutant in ambient air one kilometer from the stack. The air quality is also given in Table 66 along with the State standard.

FIGURE 51. SWIFT Flowsheet for Overall Material Balance

TABLE 65

Process Material Balance for the SWIFT

Stream No.	1	2	3	4	5	6	7	8	9
Location	Waste Feed	Fuel	Comb. Air	Ash	Incin. Exhaust	Quench Solution	Cooled Exhaust	Dilut. Air	Stack Gas
Temp, °C	17	17	17	400	800-900	17	180	17	100
Total, lb/hr	400	108	11,400	104	11,655	2087	13,715	7180	20,890
Volume	67 ft ³ /hr 200 cellulose	15 GPH	2660 ACFM Pri. burner	4 ft ³ /hr C, Si, S	10,460 ACFM	4.2 GPM	4220 ACFM	1670 ACFM	6660 ACFM
General Composition	92 PE 76 rubber 32 PVC		Under. air Sec. burner Sec. air	K, Ca, Ti Fe, Zn, Cl					
Components									
C	190	94		17					
O ₂			2630		1960		1960	1650	3610
N ₂			8790		8790		8790	5530	14,320
H ₂ O					70	2060	2130		2130
CO ₂					835		835		835
HCl									1
Na ₂ CO ₃	16 (est)								

TABLE 66

10 (10)

Pollutant Release Rate and Corresponding Air Quality

Component	Red Rubber		Waste Mix*		Waste Mix Burndown**		State Standard Air Quality, $\mu\text{g}/\text{m}^3$
	Release Rate, g/hr	Air Quality, $\mu\text{g}/\text{m}^3$	Release Rate, g/hr	Air Quality, $\mu\text{g}/\text{m}^3$	Release Rate, g/hr	Air Quality, $\mu\text{g}/\text{m}^3$	
NO_x	160	5.5	12	0.2	28	0.4	100
SO_2	1830	3.3	1200	2.1	2450	4.4	1300
HCl	390	1.3	590	1.6	900	2.4	Not specified
CO		<160		<160		<160	40,000
Particulate	13.5	0.02	13.5	0.02	13.5	0.02	60

* Data taken while neutralizing stack gas with Na_2CO_3 .

** No neutralization in use.

Particulate

A filter used to capture particulates in the gas stream is the first stage of the stack sampler. During the standard sample period (24 min), the particulates collected on the filter is slightly visible but too small to accurately weigh. A special test was conducted by pulling a sample for 24 hr while the incinerator was idling. The burners were cycling on and off during the test, but no waste was burned. The fly ash trapped on the filter is generally <1 micron in diameter. The release rate of solids in the exhaust gas calculated from this information is 0.02 lb/hr, which is equivalent to an air quality of $0.02 \mu\text{g}/\text{m}^3$ at a point one kilometer from the stack (State standard = $60 \mu\text{g}/\text{m}^3$).

A decontamination factor (DF) of 480 was calculated by the following equation:

$$\frac{M_{\text{ash}}}{Q_{\text{par}} * T}$$

$M_{\text{ash}} = 3600 \text{ lb}$ (total amount of ash removed during campaign runs)

$Q_{\text{par}} = 0.02 \text{ lb/hr}$ (particulate release rate)

$T = 370 \text{ hr}$ (hours of facility operation)

Equipment Performance

Facility Inspection. A thorough inspection of the process equipment was made after Run 3. In general, the equipment condition was good, but the SWIFT had only operated 300 hr at the time of the inspection.

The areas around the charging door and the burner are discolored with layers of soot that vented from the incinerator during periods of positive pressure.

The condition of the ceramic liner in the incinerator and part of the exhaust duct was checked. Cracks exist in the refractory, both circumferentially and longitudinally. The cracks vary from 1/4-in. to 1-in. wide but are expected to close at operating temperatures. The bricks in the hearth and fire step are in good shape.

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DEFENSE WASTE PROCESSING FACILITY

An extensive effort to update the CAB cost estimate for the DWPF sludge facilities was completed early in the quarter. This estimate was completed after agreements were reached regarding space requirements for crane decontamination, crane maintenance, weld test cell, etc. The estimated project cost was determined to be 880 million dollars assuming that funding is provided as required.

Several design changes were made during the quarter. Some of the more significant changes involving the vitrification building were as follows:

- The process equipment repair shop was deleted. The manipulator repair shop was relocated to the area previously occupied by the equipment repair shop.
- Decontamination solution make-up and feed tankage were relocated to achieve better efficiency.
- Elevation differences between crane maintenance and operating corridors were minimized.

As a result of the addition of cell covers to all process cells in the vitrification building (to better control contamination, reevaluation of heat loads, etc.), the design air flow for sand filter, fan house, and stack was reduced to 115,000 cfm.

A revised design basis for mercury removal from the waste sludge by distillation during feed adjustment in the slurry receipt and adjustment tank was transmitted to the Du Pont Engineering Department. About 75% of the mercury will be received by steam distillation. Of the remainder, about one-half will be transferred to the melter with the feed; the remainder will be retained as an amalgam with noble metals.

The design basis for the canister documentation process was revised at the end of the quarter. SRL tests indicated that an air-injected, slurry-blasting process was more efficient than direct pumping of the frit slurry. The change in the canister decontamination process is not expected to impact the space allocations.

After filling, the glass canister must be temporarily sealed during decontamination. Based on tests by SRL and the Du Pont

Equipment Engineering Department, the design basis was changed from a Welch plug to a shrink-fit plug.

After cost analysis by the Du Pont Engineering Department, a 115 kV primary substation was deleted from the scope of the DWPF project. Instead, electrical power will be obtained by installing two 13.8 kV lines from an upgraded substation in H Area. This upgrading is expected to be completed in 1985.

Preliminary design efforts continued for the supernate processing. Cost estimates are being developed for various facilities. These facilities include:

- Upgrading the existing experimental in-tank Cs and Sr precipitation equipment to a production facility status.
- A ^{99}Tc removal facility for decontaminating supernate.
- A facility for the disposal of decontaminated supernate onsite as saltcrete.

It is expected that these estimates will be completed during 1983.

Quality Assurance assessments of the DWPF were initiated during the year. For any item or service whose failure may result in exceeding an assessment criteria, an action plan will be developed to determine the requirements for preventing or mitigating the failure.

SR INTERIM WASTE OPERATIONS

LIQUID WASTE STORAGE OPERATIONS AND RELATED ACTIVITIES

Surveillance and Maintenance

Tank 11H Flooding

Because heavy rains flooded Tank 11H on September 19, a program to prevent future flooding was initiated. The storm sewer outfall and channel were cleared of vegetation and dredged. During October tests were made by shutting Tank 9-12H sewer gates at the 907-1H diversion gate box and then filling the gate box and sewers with water to simulate a rainfall. Not enough water could be added to match the unusually heavy rain experienced on September 19. However, the test came after the creek at the outfall had been dredged and did wash a great amount of sand and debris out of the pipe. Later television inspection by a contractor showed that both the 30-in. line from the storm grating at Tank 11H to the 907-1H diversion gate box and the 54-in. line from the gate box to the outfall at the creek were essentially clear of any debris. Inspection of F-Area storm sewer lines with the television camera revealed sand up to 50% of pipe depth in three lines from the Tank 17-20F area. In November an outside contractor removed the debris from the lines.

Calculations indicated that the H-Area sewer system clean capacity is only marginally greater than Tank 9-12H rainfall runoff for an unusually heavy rain such as that on September 19. Since the system had sand and debris in it, and since the outfall was overgrown with vegetation, the system capacity was less than the runoff. Investigation into other contributing factors to the Tank 11H flooding is continuing.

Radioactivity in the Annulus Air from Tanks 29H and 31H

Beginning in August, radioactivity up to 50 times background has been detected intermittently in the air exhaust from the annuli of Tanks 29H and 31H. Investigation of the problem is not complete, but the probable source of the activity is contaminated vapor from the HCTS pump tank cell. Tests conducted in October have shown that the sleeves around the concentrate lines penetrating the tank annuli leak air past the asbestos rope caulking which seals the telescoping sleeve joint. Since the concentrate line

jackets drain to the HCTS cell, there is a path for airborne activity between the cell and the two annuli. Air samples from the sleeves have also shown activity and the annuli are maintained at negative pressures with respect to the sleeves.

In other tests, DOP added to the interiors of Tanks 29H and 31H was not detected in the annuli. This indicates no communication via the vapor space. Smears of the annulus walls and floors and of the test ports on the concentrate line sleeves all showed no activity. Photographic and periscopic inspections of the annuli have revealed no salt nodules, indicating that there is no waste leak.

During November, there were no incidents of activity in the annulus exhausts of Tanks 29H and 31H. However, activity continues to be found in the telescoping jackets around the CTS supply and return where these pipes penetrate the tank annuli. The leak detection box (LDB) No. 3 drain from the CTS jackets to the CTS pipe sump remains the suspected path for contamination to migrate from the pit to the annuli. During most of November, the drain was submerged by several inches of water in the sump while new CTS tank ventilation equipment was tied in. This probably accounts for the halt in annulus air incidents. However, in one November test the sump was emptied and DOP smoke was added to the CTS cell. No DOP smoke was detected in Tank 29H annulus. This test will be repeated.

The coincidence of annulus activity and other waste farm and waste tank events will continue to be followed. Means to isolate the sleeves from the HCTS cell are being considered.

Cathodic Protection

Cathodic protection (CP) field surveys were recently performed by the Ebasco Corporation in F Area and by Harco Corporation in H Area to determine the feasibility of applying CP to underground tank-farm piping. In reports submitted to SRP in July, both vendors indicated the installation of CP is feasible. Additionally, both vendors indicated that corrosive soil conditions and a history of underground pipe failures make CP desirable as well.

After study, EED has recommended the installation of two turn-key CP systems to protect underground tank-farm piping in both F and H Area. CP systems in this application are expected to arrest corrosion and increase the life of protected buried piping by a factor of 100. CP systems in the tank farms can be designed to minimize any potential stray current damage.

The estimated cost for applying CP to the buried tank-farm piping in F and H Area, in FY-1982 dollars, is \$3.1 million. Waste Management Technology is currently studying the recommendation.

Waste Concentration

Repair of HCTS Loop to Tanks 35H-37H

Examination of the previously removed 2-in. sections of core pipe including the field welds on the west side of Cleanout Port 3 showed cracks extending up to 80% through the pipe. Four 12-in. sections of core pipe were removed from the 510 and 201 lines around Cleanout Port 3. The sections were examined by dye penetrant test and microscopic inspection. Stress corrosion cracks in the longitudinal and radial directions were found in three of the four sections. The cracks extended up to 30% through the pipe and were neither close to a field weld nor concentrated around the pipe seam.

In October, eddy current probes were used to test for cracks 20-ft up the core pipes both east and west of Cleanout Port 3. The probes were not sensitive enough to indicate cracks in any of the four 12-in. excised sections. However, abnormal currents indicated possible cracks in three of the four core pipes.

The abnormal current indications at 18 ft into all four pipes are probably welds. The indications on the 201 east line at 16 ft and on the 510 east line at 6 ft may be due to pipe spacers. There are no known explanations for the other abnormalities:

- at 2 ft in 510 line - east
- at 8 ft in 201 line - east
- at 11 ft in 510 line - west

The field tests were compared with eddy current tests of similar clean pipe at central shops. Tests of the clean pipe did not show any abnormalities.

A hydrostatic test of the 251 line at Cleanout Port 9 was also conducted in October to determine if a leak existed. The line was pressurized to 150 psig after which a pressure drop of 12 psi was observed in 30 min. Maintenance conducted a second hydrostatic test on a capped pipe with a leak rate similar to that observed at the spool piece gasket on the Tank 37H end of the 251 line. The test showed a comparable pressure drop. Further attempts to test Cleanout Port 9 and the 251 line by hydrostatic pressure have been discontinued. The probability of a substantial leak in line 251 is low, but the possibility of a small leak remains.

Sections of both the 510 and 201 lines (12-ft east and 8-ft west of Cleanout Port 3) were removed in November and inspected by eddy current test. Four 11-in. samples were removed from each of the 12-ft sections (one sample from each end, and two samples from areas where eddy current testing indicated possible cracking). Two

11-in. samples were removed from each of the 8-ft sections (one sample from each end). All the 11-in. samples were decontaminated at Building 299-H and inspected by dye penetrant tests. Results of the dye penetrant test are shown in Table 67.

TABLE 67

Dye Penetrant Test Results

	<u>510-E</u>	<u>201-E</u>	<u>510-W</u>	<u>201-W</u>
End nearest to CP 3	X	X	X	0
Suspect area nearest to CP 3	X	X	NA	NA
Suspect area farthest from CP 3	0	0	NA	NA
End farthest from CP 3	0	0	0	0

X - cracks and/or pits found

0 - no defects found

NA - not applicable

The areas where pitting and/or cracking were found are the areas where a 150 psig underground steam header had failed. Leaking steam probably heated the pipe sections corresponding to the crack locations above 40°C, the minimum temperature for significant propagation of stress corrosion cracks in the core pipes.

In November, EED and Waste Management concluded that the corrosion phenomenon observed at Cleanout Port 3 was localized and all defective piping was removed.

Replacement of the failed cleanout port in the HCTS loop to Tanks 35-37H began on December 12. Twelve feet of piping east and 8 ft of piping west of the cleanout port are also being replaced. All welds will be x-rayed, and the cores will be hydrostatically tested before being returned to service. The system should be ready for operation by February 1.

Low-Level Effluent Waste

Contaminated Water Control

F- and H-Area canyon cooling water samples and tank farm runoff water samples were analyzed by the 400-D Area Laboratory in October. The results are given in Table 68 and will be incorporated into the ion exchange and reverse osmosis development work.

The development of reverse osmosis for the contaminated water control project has been delayed until March 1983. Tests in the

TABLE 68

Water Analysis

Analysis	Canyon Cooling Water		Tank Farm Runoff Water		Units
	F Area	H Area	F Area	H Area	
pH	7.1	7.7	6.1	6.0	pH
Alkalinity	16.7	45.4	16.1	15.5	mg/L CaCO ₃
Turbidity	3.6	2.4	39.9	81.2	mg/L SiO ₂
Hardness	5	8	12	22	mg/L CaCO ₃
Conductivity	96	145	70	58	µmhos
Nonfilterable silica	3.2	3.9	12.3	15.0	mg/L Si
Filterable silica	<0.1	<0.1	1.2	<0.1	mg/L Si
Suspended solids	2	2	61	35	mg/L
Total dissolved solids	64	86	54	89	mg/L
Calcium	1	3	4	7	mg/L Ca
Magnesium	0.3	0.4	0.5	0.7	mg/L Mg
Manganese	<0.01	<0.01	0.04	0.08	mg/L Mn
Potassium	<1	<1	<1	<1	mg/L K
Sodium	13	9	3	5	mg/L Na
Iron	0.7	0.4	2.4	3.6	mg/L Fe
Bicarbonate	20.3	55.4	19.6	18.9	mg/L HCO ₃
Carbonate	<1	<1	<1	<1	mg/L CO ₃
Chloride	4.0	4.0	3.0	1.0	mg/L Cl
Sulfate	14.2	14.0	11.1	7.0	mg/L SO ₄
Fluoride	<1	<1	<1	<1	mg/L F
Residual chlorine	0.5	0.3	NA	NA	mg/L Cl ₂

high-level caves with a pilot-scale, reverse-osmosis device were to begin in January, when the reverse osmosis system was ready for operation. However, Waste Management has requested that the reverse osmosis pilot-scale facility be used initially to gather data on the removal of mercury, chromium, and cesium from waste farm evaporator overheads and RBOF wastes. Tests for this project will be conducted in January and February 1983. Testing for the contaminated water control project will begin in early March 1983. The tests will provide data for process selection by June so the project can remain a line item in the FY-1986 budget.

As part of the interim program, Waste Management and Separations Technology are developing basic data for modifications to the F- and H-Area canyon cooling water delaying basins to reduce the volume of contaminated water from a radioactive release. An REA to Project Department for the work will be issued in January.

Reverse Osmosis Demonstration

Waste Management has allocated funds to install and test a 10-gpm reverse-osmosis (RO) system at the 242-H evaporator. The tests are to determine the effectiveness of RO at removing cesium, mercury, and chromium from the combined stream of Tank 23H RBOF waste and evaporator overheads that is discharged from the Tank 24H CRC. The RO system will be tied into the CRC return line to process low-radioactive waste during the demonstration.

Permeate will be sent to the evaporator overhead tanks for release to the seepage basins. Concentrate from the RO system is to be discharged to waste Tank 22H.

A Basic Data Report was written for the RO system in November and an REA is being issued to the Project Department to provide the necessary site preparation. Waste Management is working with the Du Pont Engineering Department to complete the system design. A purchase requisition is to be issued for the RO system upon completion of that design. Startup of the 10 gpm system is tentatively scheduled for April 1983.

Releases to F-Area Seepage Basins

Ruthenium-103 exceeded the seepage basin monthly guide in August and September. Zirconium-95 exceeded the annual release guide in August and the seepage basin monthly guide in September and October. Releases of ^{95}Nb exceeded the October monthly guide. The ^{103}Ru , ^{95}Zr , and ^{95}Nb releases were attributed to the acid recovery unit (ARU) overheads.

Total recycle of the ARU overheads was to have been achieved in November. However, because of difficulties in piping modifications, 100% recycle is not expected until the first quarter of 1983.

The releases over guides for August, September, and October and year-to-date (YTD) are listed below.

TABLE 69

Releases to F-Area Seepage Basins, curies

<u>Nuclide</u>	<u>August Release</u>	<u>September Release</u>	<u>October Release</u>	<u>Monthly Guide</u>	<u>YTD Release</u>	<u>Annual Guide</u>
⁹⁵ Zr	0.223	0.559	0.690	0.333	5.357	4.0
¹⁰³ Ru	0.274	0.196	0.129	0.166	1.986	2.0
⁹⁵ Nb	0.134	0.322	0.459	0.333	3.239	4.0

Releases to H-Area Seepage Basins

The releases of 0.190 Ci ⁹⁰Sr to the seepage basins in August exceeded the monthly guide of 0.083 Ci. In early August the Tank 13H liquid level dropped to only about 5 in. above the sludge level. The agitation of that 5-in. layer by a partial recycle of liquid drawn off by the evaporator feed pump probably caused greater than normal ⁹⁰Sr in the evaporator feed with subsequent release to seepage via evaporator overheads. The ⁹⁰Sr releases for 1982 remain below the annual and prorated annual guides. All other releases to the seepage basins through October were below the prorated monthly and annual guide limits.

Seepage Basins F and H Areas

A three-part program is being developed to investigate the F- and H-Area seepage basins with respect to their organic and inorganic content, influx volume and sources, evaporation and seepage rates, and the controlling seepage mechanism. From this investigation a procedure will be developed to operate the basins within the prescribed release guides for radioactive nuclides and nonradioactive hazardous materials.

In Part I of the program all previous studies of the seepage basins and the actual operating performance of the seepage basins will be reviewed and summarized. In Part II further tests will be conducted as necessary to define the operating characteristics of the seepage basins. Part III will develop and implement control procedures for operating the seepage basins.

Tank Replacement/Waste Transfer

Mechanical Agitation Salt Removal

Tank 24H salt removal remains suspended until January 1983 when dissolved salt solution will be transferred to Tank 48H for the in-tank salt decontamination demonstration. By the end of December, approximately 822,000 gal of dissolved salt solution containing 296,000 gal of salt (52% of the total) were transferred from Tank 24H.

About 12,200 gal 25 wt % caustic were added to Tank 19F in October after samples showed marginal corrosion inhibitor concentrations. A surface and peristaltic pump sample were obtained one week after the caustic was added to confirm adequate mixing. Sample results confirmed inhibitors were well within Technical Standard limits.

Density-Driven Salt Removal

Tank 20F density-driven salt removal has been completed. A total of 714,000 gal of salt (66% of the total) has been removed from Tank 20F.

Recent sample analysis indicates Tank 20F supernate is above the 5.5M nitrate Technical Standard limit but within limits originally authorized for salt removal. Slow strain rate tests conducted by SRL at Tank 20F conditions showed low risk of crack initiation. Wedge open-loaded tests, after 2000 hr of exposure, show no sign of crack propagation; the tests will continue to 4000 hr exposure. A Test Authorization is being written to authorize continued storage in Tank 20F under present conditions based on the SRL test results.

Recent sample analysis indicated Tank 10H supernate was outside Technical Standard corrosion inhibitor limits. To bring Tank 10H within Technical Standard limits, approximately 175,000 gal of Tank 10H supernate containing 78,000 gal of salt (15.5% of the original) were transferred to Tank 13H in December. About 72.5% of the original 500,000 gal of salt in Tank 10H was removed in 844,000 gal of solution. Following the transfer, high caustic supernate or 25 wt % caustic may be added to Tank 10H to increase inhibitor concentrations. The Tank 10H interior was inspected periodically during transfers to Tank 13H. No unsupported salt was observed on the Tank 10H cooling coils or walls.

In-Tank Sludge Processing Demonstration

An additional 126,000 gal of well water were added to Tank 42H in October to complete water addition for the second water wash (total volume 1.15 million gal). Liquid surface agitation and cooling coil movement with pumps operating at 1800 rpm were recorded on videotape at the full-wash volume (328 in.). An edited videotape of cooling coil movement was reviewed with the Du Pont Engineering Department. The Du Pont Engineering Department will determine if coil movement will damage the cooling coils over an extended period of time.

Slurry samples from the second wash contained 1.22 wt % dissolved salt (5.3 wt % salt in 19 wt % sludge on a dry basis).

The second wash slurry started settling on September 27. Settling rates (Table 70) were measured with the ultrasonic interface detector. Tank 42H sludge settled at about the same rate as SRL lab tests predicted. Settling rates and interface heights are summarized in Table 71.

TABLE 70

Settling Rates

<u>Settling Time, hours</u>	<u>Settling Rate,* inches/hour</u>
0	0
8	5.4
30	3.7
78	1.75
102	0.25
126	0.42
150	0.125
174	0.08
204	0.03
270	0.15
444	0.12
516	0.03

* Settling rates calculated from previous reading.

After the sludge/supernate interface had settled to below 66 in., Tank 42H was sampled to confirm settling. A peristaltic pump sample from 66 in. contained 26 ppm solids compared to 20 ppm solids in a surface sample. About 971,000 gal of wash solution were transferred to Tank 43H between October 6 and 18. Following the decant the sludge was slurried and sampled. Well water addition for the third wash began October 22 after 2600 gal of 30 wt % sodium nitrite were added to maintain the nitrite inhibitor concentration at 1000 ppm.

The well water addition for the third water wash was completed October 30. A total of 540,000 gal of water was added. Sludge settling began October 30 after the slurry was sampled. Through November 19, the sludge level in Tank 42H had settled to 30 in. above the tank bottom. Settling rates and sludge interface heights for the third wash are as follows:

TABLE 71

Settling Rates and Interface Heights

<u>Settling Time, hr</u>	<u>Interface Height, in.</u>	<u>Settling Rate, in./hr</u>
0	220	-
52	100	2.3
100	80	0.4
148	73	0.15
220	61	0.17
287	48	0.2
407	33	0.13
479	30	0.04

The dissolved salt concentration in the third water wash slurry was 0.49 wt % which is within the experimental error of the goal concentration (0.46 wt %). The 0.49 wt % dissolved salt concentration corresponds to 2.1 wt % salt in sludge on a dry basis (goal for feed to DWPF of 2 wt % in 19 wt % sludge on a dry basis).

During December the processed sludge in Tank 42H settled to about 26 in. This sludge level corresponds to a 14 wt % total solids sludge. No significant settling was detected during December. SRL is evaluating the impact of feeding 13 wt % sludge to the DWPF sludge plant.

About 278,000 gal of the third water wash supernate were decanted to Tank 43H in December. An additional 320,000 gal will be decanted when space is available in Tank 43H.

A 25-L sample pulled from Tank 42H on December 7 contained only 1 L of sludge and 24 L of supernate. Another 25-L sample will be pulled after the decant of wash supernate has been completed and the sludge has been slurried.

An RTA was issued to EED in December to design and construct a floating corrosion coupon holder to be used in Tank 42H to study corrosion at the liquid-air interface. The preliminary design was completed in December. Delivery is expected by mid-January.

Tank 16H Annulus Cleaning Demonstration

Construction of the facility to demonstrate annulus cleaning (Project S-1334) was completed in December. Three circulation jet assemblies were installed in October in the east, west, and south annulus risers. Steam piping from gang valves to the jet assemblies was completed in November. During October, the chemical feed line to the annulus was installed. The control valve and flowmeter on the feed line were added in November, and thermocouple installation in the tank annulus was also completed. Instrument checkout is now in progress.

Waste Transfer Program Status

Installation of inhibited water, spray wash, and instrument air piping on Tank 17F was completed in October. During November, the lines were heat traced, insulated, and the cable trays to Tank 17F were completed. The Tank 18F inlet from FDB-2 was removed in October. Work on the F- and H-Area waste removal control rooms is about 55% complete. The Tank 9H telescoping transfer jet and transfer lines were both completed in October. Salt removal Project 95-1959 was closed out with the exception of funds for the lead shielded slurry pump spacers. Other waste removal construction remains suspended. Service piping on Tank 3F and pipe insulation on Tank 2F were not completed under the original project.

Salt Decontamination Demonstration

Construction of the salt decontamination demonstration facility is about 90% complete. During October, Construction installed the slurry pumps on Tank 48H and set the slurry pump impeller clearances. Construction also poured equipment foundations and installed the chemical addition tanks. During November, Construction installed the telescoping transfer pump (TFP) and filters. In

December, Construction installed the diaphragm pump and ran in the service pumps. The service pumps include the sealwater pump, caustic pump, chemical feed pump, and chemical cleaning pump. Cable tray and piping installation are in progress. Heat tracing is nearly complete, and insulation is 30% complete. The final tank interior inspection was completed December 14.

In November, two surface cracks were discovered in one end of the lead shielded process line core pipe between the diaphragm pump adapter and the valve box. The cracks were over 4-in. long. They started about 90° apart at one end of the pipe and gradually converged along the pipe.

Poor fabrication caused the cracks. During December, EED inspected the 8-ft section of pipe which was replaced. Minute dents were noted in part of the pipe interior, and some minute crevices were found on the pipe exterior by dye penetrant inspection. Dye penetrant checking of the core pipe in the valve box showed no cracking. Other parts of the core pipe were not checked because of inaccessibility. However, EED and Waste Management are satisfied that the cracking was limited to the 8-ft section of pipe.

Plans for a cold run-in of the demonstration equipment installed in Tank 48H were finalized in November. The run-in will allow thorough procedure checkout, operation training, and equipment checkout. Repairs or modifications, if necessary, can then be performed on nonradioactive equipment. The cold run-in will be scheduled as part of the final equipment checkout.

For the cold run-in about 3200 lb of sodium tetraphenylborate (STPB), 200 lb of sodium titanate (ST), and 300 lb of potassium nitrate will be mixed with 50,000 gal of a 5.5M Na⁺ solution (3.7M NaOH and 1.8M NaNO₂) to form a solution similar to the actual hot solution as of December 22, 1982, about 25,000 lb of STBP for the demonstration has been received at SRP. The remainder of that required for the demonstration (~9000 lb) has been shipped.

The following is a summary of these receipts:

<u>Vendor</u>	<u>Quantity Ordered, lb</u>	<u>Quantity Received, lb</u>
Boulder	20,000	16,315
Raylo	15,000	5,650
AFF	6,000	1,700
Preiser	<u>1,000</u>	<u>1,000</u>
	42,000	24,665

Quality Control tests to confirm STPB quality are being performed on each drum of STPB received.

The average purity of the STPB tested is about 98%, well above the required 95%.

During October, flash point tests were performed on simulated Tank 48H solutions. Two conditions were tested; (1) after ST addition only (conservative case) and (2) after both ST and STBP addition (real case). Results showed that after sodium titanate is added to Tank 48H, the 15 vol % alcohol will be diluted outside the flammable and combustible regions (flash point $>200^{\circ}\text{F}$). Initial test of the ST and STPB solutions showed flash points under 200°F . Two additional tests using ST and STPB have shown no flash point ($>200^{\circ}\text{F}$). Impurities in the test apparatus during the first test may have caused the anomalous result. In addition to the flash point tests, the Experimental Station (Polymer Products Department) at Wilmington was requested to evaluate the thermal stability of STPB.

Several tests were also performed in October to evaluate foam film conductance in relation to tank liquid and foam level detection. Tests indicated that foam is conductive and will be detected by the tank reel tape and conductivity probe. The reel tape also detected the foam even after a foam residue had dried on the reel tape. An interface detector, like the one currently used in Tank 42H to detect the sludge/supernate interface, was tested in synthetic salt processing solutions in October. It was capable of detecting the supernate level below the foam layer. Plans are to install an interface detector in Tank 48H for liquid level measurement.

Dip tubes will also be available to measure liquid level during the demonstration. Additional tests conducted during November confirmed that solution foaming will not be a hazard during the demonstration. Tests to determine the extent of foaming have shown that a 6-in. foam layer forms under severe agitation. The foam stabilizes at 1 to 2 in. after agitation ceases. Other tests showed that both wet and dry foam are not flammable.

Plans are to provide continuous monitoring of benzene vapor in the Tank 48H interior ventilation exhaust. The benzene concentration in the tank exhaust has been calculated to be about 0.1 ppm. [Threshold Limit Value (TLV) is 10 ppm.] The monitor will be set to alarm at 5 ppm. A continuous monitor has been ordered and delivery is expected in January 1983. A portable monitor on loan from HP will be used as a backup.

In November two sodium titanate formulations were evaluated for the demonstration. Isopropyl alcohol (IPA) and methanol (MeOH) were used in the original formula. The other formula uses methanol only for sodium titanate preparation. Tests indicate that the methanol formula yields higher strontium decontamination factors

(250 versus 100). The higher decontamination factors reduce the required quantity of sodium titanate by half. Less ST in the process solutions will increase filtrate flow rates. Flash points for the IPA and MeOH sodium titanate solutions are 97°F and 98°F, respectively.

During December SRL conducted tests for decontamination factors and filterability of the methanol-based sodium titanate slurry. Decontamination factors and filterability were confirmed to be similar to previously tested sodium titanate slurries (isopropyl alcohol-based formula). The sodium titanate required for the demonstration will be prepared with the methanol-based formula since it yields high strontium decontamination factors. AFF in Atlanta, GA, has begun production of the sodium titanate slurry required for the demonstration.

During November the Barrett-Haentjens telescoping transfer pump was successfully run in at TNX. The telescoping transfer pump will serve as a backup to a diaphragm pump for feeding the filter assembly.

Run-in of the diaphragm pump at the TNX full-tank test facility was completed in December. The diaphragm pump will be used as the primary filter feed pump for the demonstration.

The crossflow filter assembly was also successfully run in at TNX in November. Five leaking valves on the filter assembly were repaired and leak-checked at a pressure of 90 to 115 psi for about 2 hr. The filters were run in with precipitate for 2 days. Filtrate rates of 50 gpm were measured. The filters were then cleaned with oxalic acid, rinsed with water, and shipped to Tank 48H. Initial tests with one filter bundle (31 tubes) obtained filtrate rates of 0.43 gpm/ft² at a 30 psi pressure differential and 90 gpm (3 ft/sec tube velocity). Testing with the four filter bundles in series yielded a lower filtrate rate (0.28 gpm/ft²) at similar head conditions. This is equivalent to an overall filtrate flow of 56 gpm. The lower rate achieved with all four bundles in series is due to a decrease in pressure differential and tube velocity in succeeding filter bundles. These tests suggest that filtrate rates obtained during the demonstration may be higher than rates seen previously in one-quarter scale tests.

Tests conducted in December by SRL to determine if filtration of the process solution is affected by iron oxide (from the tank, primary floor and walls) indicated that filtrate rates are reduced by about 20% after rust addition to the synthetic process solution. This reduction in filtrate rates does not significantly impact the objectives of the demonstration. The filters will be cleaned and the cleaning solution analyzed for iron oxide to determine if the iron oxide is concentrated in the filter pores.

The Quality Assurance Assessment Report (QAAR) was approved by Waste Management in November. Final approval is expected by January 3, 1983. The Test Authorization was completed in December and is now circulating for final approval.

A sample schedule for the demonstration was also developed in November. About 46 samples of Tank 48H contents will be required during the demonstration (three 10-gal and 43 variable-depth samples) to monitor test progress and corrosion inhibitors. Over 600 filtrate samples (taken by automatic samplers) are necessary to confirm that adequate decontamination has been achieved. A new vial design for sampling concentrated precipitate was successfully tested in simulated solution. The vial is compatible with the variable-depth sampler to be used in Tank 48H during the demonstration. Ten vials will be supplied by EED by January 10, 1983.

In December a test plan for the demonstration was issued to Waste Management Operations. The plan presents a detailed outline for operation of the Tank 48H facilities. Transfer of supernate from Tank 24H to Tank 48H, the first step of the demonstration, is scheduled to begin January 20, 1983.

Saltcrete Demonstration

The SRL saltcrete demonstration scheduled for March 1983 will use 11,000 gal of decontaminated supernate (filtrate in Tank 50H) from the January in-tank salt decontamination demonstration. Basic design data for the Tank 50H filtrate loading station was issued in September and was revised in October to incorporate the following:

- The project will provide temporary spill control dikes.
- The transfer line welds will be dye penetrant tested and hydrostatically tested upon completion rather than 10% radiographed.
- A filtrate flow totalizer will be installed on the filtrate transfer line.

A 7.5 hp submersible pump has been selected to transfer decontaminated supernate from Tank 50H into the tank trucks. The purchase requisition for the pump was approved in November. An existing evaporator feed pump was rejected due to the costs of installation (required ammeter, high temperature alarm, and leak detection).

The saltcrete test authorization was approved in December. The current plan for saltcrete demonstration is to process for six hours during the day with a truck discharge rate of 10 gpm. This will reduce the number of trucks actually needed to one truck per day. A spare truck will be available.

Final Tank 50H unloading station prints were routed for approval in December. Construction will start as soon as the approved prints are issued.

SOLID WASTE STORAGE AND RELATED ACTIVITIES

WIPP Certification Program

The FY-1983 Current Year Work Plan was completed and forwarded to DOE in October. For this program the DOE/Albuquerque Operations Office has allotted SRP \$125,000 to be used for developing an inventory work-off plan for retrievably stored contact-handled TRU waste at SRP. After retrieval this waste will be certified to WIPP requirements. Major activities for this year are:

- Characterize retrievably stored TRU waste, including isotopic content, volume, mass, and packaging description.
- Develop a retrievably stored waste work-off strategy and schedule.
- Identify resource needs for implementing the retrievably stored TRU waste certification plan.
- Identify potential problem areas in recovering and certifying this waste.

During October comments were received from the WIPP Review Committee on the draft of the "WIPP Certification Plan for Newly Generated Contact-Handled TRU Waste." These comments were incorporated into the final draft of the document.

In accordance with the National Environment Policy Act (NEPA), SRL was requested in October to perform an Environmental Evaluation (EE) of the proposed TRU waste certification building and storage facility.

During November, a suitable location in the burial ground for the TRU waste assay system was determined. The basic design data package for the TRU waste assay system was completed in December. This project will provide an assay system, using the LANL design, to be used for determining the variety and quantity of TRU isotopes contained in solid waste packaged in 55-gal drums. The overall sensitivity of this instrument will be below the present 10 nCi/g TRU level.

The assay chamber will be located in Building 724-5G, which has adequate storage space for the estimated TRU waste generated in one week of normal plant operations. Required instrumentation and

computers will be located in Building 724-6G. This building is already equipped with the H & V system which the equipment will require. After turnover, the assay system will be operated by Waste Management Operations.

During November plant records were compiled to prepare an inventory of stored TRU waste packages located on the retrievable pads in the burial ground. This information will be incorporated into the TRU Waste Inventory Work-Off Plan which will describe the SRP program for certifying this retrievably stored waste to WIPP requirements.

Nuclear safety assessments of proposed storage methods for TRU waste packages in the certification building and storage facility were begun in October. Preliminary analysis indicated these storage methods would not present any potential hazards or risks with the exception of stacking drums four high and then only if all contain greater than 100 g ²³⁹Pu.

A meeting was held in Albuquerque, NM, in December with DOE and U.S. Defense Program, TRU waste generators, to review available data on gas generation drums containing TRU waste. Studies have shown that gases are generated in TRU waste drums primarily from the radiolysis of organic matter and bacterial degradation. Of particular concern is the generation of hydrogen, which when combined with oxygen present in the drum vapor space, has a potential for producing explosive mixtures inside the waste container. This situation could hamper waste certification efforts, particularly those sites which produce significant quantities of high specific activity isotopes (i.e., Pu-238 and Cf-252), including SRP. Although no firm decisions were made, additional meetings will be forthcoming as more information becomes available, to discuss resolution of this situation.

Beta-Gamma Incinerator Demonstration

Disassembly of the beta-gamma incinerator at SRL was completed in November. The primary and secondary chambers, spray quencher, and most of the duct work were relocated to the H-Area demonstration site in November and erected in December. Also during November the feed lag storage area was backfilled and the solvent storage pad and dike were poured. Pouring of footings and floor slabs was completed in December. The receiving ramp wall and the ash-out enclosure were installed in December.

Baumco Incorporated was selected as the baghouse vendor. The baghouse, costing \$32,800, will be shipped between February 4 and March 17.

Excavation of the floor drain piping and installation of the fuel oil tank pad are both in progress. Cutting and grading of access roads continue. Erection of the prefabricated building is scheduled to start December 22.

²³⁸Pu Waste Incinerator

Conceptual design studies for an incinerator to volume reduce an estimated 5000 ft³/yr of noncertifiable (WIPP) ²³⁸Pu contaminated waste were reviewed in October. Tentative plans are to locate the incinerator in the 643-G burial ground as the first stage of a TRU waste management complex which will include facilities for alpha D&D and WIPP certification.

The final draft of the PWI Technical Data Summary was issued in December summarizing test results obtained since 1978 from the PWI prototype at TNX. In 170 test runs, more than 8300 lb of simulated waste was successfully burned by SRL. SRL, in cooperation with Waste Management, will also be requested to develop the design for feed preparation (i.e., x-ray, assay, shredding, etc.) and ash disposal.

A draft Basic Design Data package specifying design requirements for a PWI facility was issued on 12/2/82 with a final version to be issued by 1/14/83. To better define design requirements for the demonstration facility, a task force has been organized by Waste Management. Project timing will be set to allow facility construction to begin in FY-1983.

Alpha Disassembly and Decontamination Facility

The alpha disassembly and decontamination facility, a FY-1986 project, is currently estimated at \$36,000,000 for construction. Due to a lack of support for this level of funding, a task force composed of SRL, Waste Management, and HB-Line personnel was organized in October to research the possibilities of constructing the facility within the abandoned HB-Line, or adjacent to the 299-H Waste Management Maintenance Facility, or as a stand-alone demonstration facility centrally located within SRP (i.e., burial ground).

Initial studies indicate HB-Line has neither sufficient floor space nor adequate ventilation. Building 299-H appears feasible, but modifications to the waste collection system would be required. A stand-alone facility also appears feasible; however, the impact of cost is not known.

Scopes of work will be prepared by the task force to resolve which location is the most cost effective in reducing the project estimate.

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