UNITED STATES DEPARTMENT OF COMMERCE
National Bureau of Standards
Gaithersburg, Maryland 20899

May 15, 1988

Mr. Charles Peterson
Technical Review Branch
Division of High-Level Waste Management
Office of Nuclear Materials Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Draft Reviews

Dear Mr. Peterson:

Enclosed is Attachment B - Draft Reviews of the April 1988 monthly progress report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-7).

Sincerely,

Charles G. Interrante

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

Enclosure

Distribution:
NMSS PM (1)
Ofc of the Director
  NMSS (Attn: PMPDAS) (1)
HLWM Div. Director (1)
HLTR Branch Chief (1)
WM Docket Control Center (4)
Office of Research (1)
Office of Administration
  and Resource Mgmt.,
  Div. of Computer and
  Telecommunications
  Services (Attn: Director) (1)

A4171
NH14/7
WM-11

Attachment B

Draft Reviews (April 1988)

NBS Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory, Livermore, California.

(b) Author(s), Reference, Reference Availability

McCright, R. D., Halsey, W. G., and Van Konynenburg, R. A., "Progress Report on the Results of Testing Advanced Conceptual Design Metal Barrier Materials Under Relevant Environmental Conditions for a Tuff Repository," UCID-21044, December 1987.

DATE REVIEWED: 3/31/88; Revised 4/28/88.

PURPOSE

" This report discusses potential degradation modes for candidate materials being considered for use in waste package containers, and summarizes the results of metal barrier testing activities conducted at LLNL and subcontractor facilities."

CONTENT

This report consists of 105 pages which include an executive summary, an introduction, 16 Tables, 17 Figures, and the following number of pages covering each topic listed.

Topic	Pages
Candidate Materials	5
Degradation Modes of Austenitic Materials Under Repository Conditions	11
General Corrosion and Oxidation of Austenitic Materials	6
Intergranular Corrosion and Intergranular Stress Corrosion Cracking of Austenitic Materials	19
Pitting Corrosion, Crevice Corrosion, and Transgranular Corrosion Cracking of Austenitic Materials	21
Phase Stability and Embrittlement of Austenitic Materials	5
Projections of Containment Lifetimes Using Austenitic Materials	6
Copper Base Materials	17
Borehole Liner Materials	3
References	7

TYPE OF DATA

Experimental. This is a review of repository conditions, emplacement considerations and potential degradation modes of six materials and summarizes test data from LLNL and other DOE contractor facilities. Tables and figures of some referenced experimental test data are given.

MATERIALS/COMPONENTS

Six candidate materials for the waste package are discussed. The materials are AISI 304L and 316L stainless steels, high-nickel austenitic alloy 825, oxygen-free copper CDA 102, 7% aluminum bronze CDA 613, and 70-30 copper-nickel, CDA 715.

TEST CONDITIONS

Topopah Spring tuff repository conditions were used in the various tests. These conditions included J-13 well water with a near neutral pH and temperatures of 28°C to 150°C. Some tests were conducted in the presence of gamma radiation at approximately 10^4 to 10^5 rads per hour.

METHODS OF DATA COLLECTION/ANALYSIS

Weight loss measurements were used to determine corrosion rates and effects of composition and other variables. Slow-strain-rate tests were conducted to obtain data on changes in mechanical properties brought about by the environment. U-bend SCC tests were conducted. Metallographic techniques, using light microscopy, were used applied to analyze microstructures for cracking and other effects.

AMOUNT OF DATA

There are 16 tables and 17 figures.

UNCERTAINITIES IN DATA

It is not certain whether sensitization would occur during the tens to hundreds of years at 100°C to 250°C in the repository.

In the transgranular SCC determinations, tuff material was taken from surface outcroppings and was not representative of the tuff at the level of the repository. Total volumes of water were not kept constant throughout the test and resulted in varying chloride concentration. Temperatures were not controlled satisfactorily at 90°C for the first month which resulted in a more concentrated solution. A peak value of chloride was measured after seven months.

DEFICIENCIES/LIMITATIONS IN DATABASE

Partial sensitization might occur and needs to be quantified. Additional slow strain rate tests are being conducted. A model to predict sensitization is being developed. Detrimental effects of hydrogen on

Alloy 825 need to be addressed. Additional work is needed to study phase instability and embrittlement. More developmental work is needed with Alloy 825 to assure its good weldability. More research is needed to determine whether microorganisms can survive the hot, dry period and later revive. More information is needed on radiation effects on corrosion and oxidation of copper and copper alloys. Interaction of borehole materials and the container will need to be experimentally determined.

<u>KEY WORDS</u>

Experimental, literature review, supporting data, general corrosion, pitting, corrosion, electrochemical, irradiation-corrosion test, microscopy, slow-strain-rate test, wieght change, x-ray diffraction,laboratory, simulated field test, Yucca Mountain, air, J-13 water, tuff composition, concentrated (20X) J-13 water,tuff gamma radiation field,high temperature, carbon steel, copper base, nickel base, stainless steel, weld, 304L stainless steel, 316L stainless steel, Alloy 825, CDA 102 copper, CDA 613 7% Al-Cu bronze, CDA 715 70-30 Cu-Ni alloy,annealed, sensitized, welded, wrought, slow strain rate, C-ring SCC, J-13 steam, chloride (low ionic content), chloride (high ionic content), tuff, cracking, hydrogen embrittlement, sensitization.

<u>CONCLUSIONS</u>

There is not a conclusions section in this report. Some of the authors' summary statements are discussed in the general comments section of this review. Some assumptions and comments, mostly from the executive summary, follow.

1. Austenite in 304L and 316L stainless steels is metastable and might transform to martensite, ferrite, sigma or to other phases causing a reduction in fracture toughness and an increased tendency toward mechanical failure.

2. All welded joints except the final closure will be annealed to relieve stress. The final closure could be a point of limitation and this should be resolved.

3. The maximum temperature after closing the repository will be 250°C. The repository will cool eventually and allow condensation or flow of water. Water near the site is low in ionic content and has a neutral pH. The atmosphere will be warm air, steam and cool aqueous air. Gamma radiation will be present and will alter the local environment during the early period.

4. Candidate materials appear to be resistant to corrosive attack to meet performance requirements, but test indicate that concentration of ionic species could cause more severe pitting attack.

5. Tests on 304 and 304L stainless steel revealed transgranular stress corrosion cracking (TGSCC) under accelerated conditions of stress, gamma flux and water chemistry. Tests are being conducted to quantify critical values and to detemine when this corrosion mode becomes limiting.

6. Indications are that phase stability and embrittlement effects will not be performance limiting, but little testing has been done, and further study is being pursued.

7. Testing of the waste package materials is directed toward determining which potential degradation modes are performance-limiting and which degradation modes do not appear to be limiting.

GENERAL COMMENTS OF REVIEWER

This report contains much useful information and addresses many of the scientific questions associated with repository metal barrier systems. There are not sufficient data and the data matrices are not developed well enough to make valid conclusions regarding the long term durability of the metal barrier. The data presented serve as a beginning for needed investigations. Some of the points made in this report are discussed in the following paragraphs.

The authors' summary of general corrosion and oxidation testing for candidate stainless steels states that a conservative corrosion rate of 0.2 um/y can be extrapolated to show that a 1-cm-thick wall would not be penetrated by general corrosion for over 1000 years. These data are not complete enough in terms of times and temperatures tested to make this assumption. Also, it may not be appropriate to extrapolate one year data at 28°C to 1000 years at repository temperatures since changes in temperatures and the environment could cause the film to thicken and scale off, leaving conditions favorable for the higher corrosion rates exhibited after two months. The tests that ran for one year were conducted at 28°C, and the tests that ran for two months were conducted at 150°C.

Experimental weight-loss data in the paper show the corrosion rates to be of 0.242 to 0.285 um/y in non-irradiated J-13 water at 28°C after one year and rates of 0.31 to 0.55 um/y after two months at 150°C. Corrosion rates based on electrochemical techniques, using Tafel extrapolation and linear polarization in shorter laboratory tests, were stated to be higher but numbers were not given.

It is inappropriate to conclude there are no effects from differences in composition of the austenitic alloys based on weight loss measurements. These compositional effects would be evident using electrochemical measurements.

Stress corrosion cracking (SCC) of austenitic materials occurs intergranularly (IGSCC) in sensitized materials and transgranularly (TGSCC) in chloride environments, and often, failures can have both transgranular and intergranular components. Examples of both SCC failure modes have been observed in AISI 304 used in the cooling system of BWR reactors.

Some studies on pitting, crevice corrosion and TGSCC were reported, and authors of this report indicated stainless steels to be sufficiently resistant to these modes of failure, but indicated that failure could occur under severe environmental conditions. Test data indicated that the stainless steels are not immune to pitting, crevice corrosion or SCC. Anodic polarization data showed hysteresis effects, which indicate a susceptibility to pitting. Additional tests are needed to determine if the protection potential moves in the direction of more negative voltage to make the material more subject to pitting. Alternately wet and dry SCC AISI 304 specimens that were tested for one year showed IGSCC, and some specimens are expected to show TGSCC. TGSCC occurs in more chloride concentrated electrolytes and nonsensitized materials. Conditions in the repository could become severe if ionic concentrations in the water increase due to evaporation.

Gamma radiation caused the corrosion potential to rise for the stainless steel and for both the copper and copper alloys, and removal of the radiation caused the potential to drop. Care in welding and specific welding compositions and parameters for given materials, stainless steel or Alloy 825, are needed. If materials are used in the "as welded" condition, the chemical and mechanical properties and the microstructural stability of the weldments should be investigated to determine the suitability of materials for canister use.

There has been limited testing of copper materials over a two year period, but test results and review information indicate that copper and copper alloys may be feasable candidate materials for use as the waste canister. TGSCC has been found in 1M nitrite solutions and nitrite could form from radiolysis of a cupric nitrate surface film on the copper. Studies involving thermodynamic analysis and assuming a stable $Cu(NO_3)_2.Cu(OH)_2$ film, indicated no change at room temperature due to radiation but other studies identified $Cu(OH)_3NO_3$ scales. Clarification is needed regarding the composition and structure of the surface film. Anodic polarization tests showed susceptibility to pitting in 0.1 N $NaNO_3$ at 95 C with pure copper being more resistant than the two alloys. A surface layer of nickel is found under the oxide of one copper alloy and a surface layer of aluminum is found under the oxide of another copper alloy. More data are needed on most aspects of corrosion of copper materials in the tuff repository environment and especially of the effects of oxygen, increased temperatures, radiation and in the area of SCC.

Corrosion test results on carbon steel indicate that a 1-inch-thick liner would last for fifty years. Maximum corrosion rates occur in high oxygen conditions and surface roughening occurs with the widths of selected areas exceeding the depths. More work is needed on this material in terms of effects of oxygen, temperature, pH and irradiation.

RECOMMENDATIONS

More work is needed in most of the major areas discussed in the report, and this need was indicated in the report for a number of cases. Some of the needed work is listed here.

1. Conduct corrosion tests for longer times in simulated repository environments and use electrochemical techniques to obtain data.

2. Conduct studies in the areas of effects of austenitic alloy composition, phase stability and embrittlement on corrosion and durability.

3. Conduct studies of effects of increased concentration of ionic species on pitting, crevice corrosion and stress corrosion cracking.

4. Determine the extent of the effects of gamma radiation of the corrosion and durability of austenitic alloys and copper materials.

5. Determine the structure, composition and stability of the surface film forming on copper materials in the repository environment and relate findings to localized and general corrosion.

6. Conduct corrosion and mechanical property studies on copper materials to show effects of oxygen, increased temperatures and radiation.

7. Conduct tests to determine susceptibility of copper materials to stress corrosion cracking in the repository environment, using methods including slow strain rate tests, U-bend tests, C-ring tests and others as needed.

8. If it is decided to use a borehole liner, galvanic effects and effects of corrosion or degradation products on the remaining waste package materials should be investigated.

APPLICABILITY OF DATA TO LICENSING

[Ranking: key data ( ), supporting (X)]

(a) Relationship to Waste Package Performance Issues Already Identified

Related to NNWSI ISTP issues

2.2.3	Mechanical failure modes for the waste package container
2.2.4	Potential corrosion failure modes for the waste package container
2.2.4.1	Corrosion rates as a function of time
2.2.4.2	Effects of radiation on corrosion failure modes and rates

2.2.4.2.2 Effects of generation of hydrogen, oxygen and other species on corrosion

2.3.1 Physical, chemical and mechanical properties of the waste package and how these properties will change with time

(b) New Licensing Issues

Effects of sensitization on corrosion behavior of austenitic materials

Effects of welding on microstructures, residual stress levels, chemical homogeniety and ultimately on mechanical integrity, corrosion and hydrogen-induced failures.

Effects of microstructural instability

(c) Comments Relating to Licensing

This report addresses a number of pertinent degradation issues which need to be settled prior to licensing.

NBS Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Pacific Northwest Laboratory Operated for the Department of Energy by the Battelle Memorial Institute, Columbus, OH.

(b) Author (s), Reference, Reference Availability, Date

"Final Report of the Defense High Level Waste Leaching Mechanisms Program," Chapter 5, Radiation Effects, PNL-5157, August 1984.

DATE REVIEWED: 9/30/87; Revised 4/26/88.

TYPE OF DATA

Literature review includes data on the effects of alpha, beta and gamma radiation on structural damage to nuclear waste forms, and subsequent leaching performance of such irradiated waste form, are meager. Measurable structural damage begins at a cumulative dose of about 1×10^{23} alpha decays/m^3 and "saturates" at a dose of approximately 5×10^{24} alpha decays/m^3. These doses correspond to cumulative doses expected for commercial glasses within the first 10,000 y of disposal in a geologic repository. A rough correlation between degree of structural damage, as measured by percent increase in solid volume, and enhancement of initial leach rate has been established for irradiated nuclear waste forms. Ground waters exposed to radiation from nuclear waste forms will form free radicals, ionic, and molecular species that can significantly alter the geochemical environment in repositories. Gamma radiolysis effects, alpha radiolysis effects, reactions between radiolytically produced species and dissolved chemical species, presence of dissolved gases and interactive effects of solids including waste forms, metallic barriers, and host rocks. Study on the effect of radiation damage on potential repositories have focused on salt. Two aspects of radiation damage in NaCl were discussed: colloid sodium formation and stored energy. The status of the computer codes that have been developed, to predict concentration of radiolytically produced species in simple ground water system was discussed briefly.

MATERIALS/COMPONENTS

Borosilicate glass, NaCl, Na_2SO_4, salt brine, tuff, basalt, granite, groundwater, synroc-c

TEST CONDITIONS

None given.

METHODS OF DATA COLLECTION/ANALYSIS

Actinide-Doping technique, external heavy-ion and neutron irradiations

AMOUNT OF DATA

Tables

5.1 Primary radiolytic species G values for Gamma and 5MeV alpha radiation.

5.2 Gamma radiolysis of basalt ground water with methane.

5.3 Gamma radiolysis of basalt ground water without dissolved methane.

5.4 Results of alpha radiolysis on a cycle 4 synethetic permian brine using ^{244} Cm.

5.5 Summary of total Pu release in MCC DWRG at 40°C in DI water.

5.6 Summary of pH change in MCC DWRG at 40°C in DI water.

5.7 Summary of Si release in MCC DRWG 40°C in DI water.

5.8 Reaction scheme for irradiated salt.

Figures

5.1 Expected volume changes as a function of dose ánd correlated to the waste storage times of both defense and commercial high-level waste forms. Dose range: 10^{21}-10^{27}, swelling range: (-1)-(+6)%.

5.2 Increase in relative leaching L/L_0 versus volume change, for percent volume change from 0 to 40% and increasing leaching factor from 0 to 80.

5.3 Total gas pressures generated by salt solutions in a ^{60}Co irradiation field. Normalized dose range: 0-25 grad, pressure range: 0-100 ATM.

5.4 The effect of SO^{2-} on the total gas pressures generated by ^{60}Co irradiated field. Normalized dose range: 0-25 Grad, Pressure Range: 0-100 ATM.

5.5 Total gas pressures generated by brine and solid waste package components in a ^{60}Co irradiated field. Normalized dose range: 0-25 Grad, Pressure Range: 0-100 ATM.

5.6 Increase in elemental leach rates of MCC 76-68 waste glass as a function of gamma-dose rate at 70°C. Dose rate range: 0-2 MR/h elemental leach rates: 0-4 gr./m^2.h ($x10^3$).

5.7 Increase in pH due to dissolved irradiated salt as a function of integrated gamma dose to the salt. pH range: 0-14, pe range: (-12)-(+24), dose range: 5.7×10^4 -2.3×10^5R.

5.8 Effect of gamma radiolysis in acidity (pH) and oxidation potential (pe) on salt brine.

5.9 Plot of the concentrations of radiolytically produced species versus time as predicted by Radiol at a dose rate of 3.5 MRads/hr and initial oxygen concentration of 1.5 ppm. Time scale: 10^{-2}-10^4, concentration scale: 10^{-6}-10^4 micromolar.

UNCERTAINTIES IN DATA

Uncertainties in many rate constants for radiation yields used in the computer modeling of radiolysis calculations and the lack of many appropriate rate constants are major limitations to the current computational efforts at modeling radiolysis of complex ground water.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

KEY WORDS

Literature review, brine, basalt composition, granite composition, tuff composition, salt, Cl, Br^-, SO_4^3, basalt, granite, tuff, alpha radiation field, gamma radiation field, redox condition, commercial high level waste (CHLW), defense high level waste (DHLW), ^{239}Pu, groundwater, corrosion (general), leaching (radiation enhancement), radiation effects.

GENERAL COMMENTS OF REVIEWER

The effect of radiation on the structural damage of nuclear waste forms is very important. Most studies on leaching of nuclear waste forms have been conducted with water in the absence of both a radiation field and other waste package barrier materials.

In this report there is information about the radiation damage of the waste form and the host rock and about the radiolysis of ground water solution. The section on the radiation damage reviews the current literature on these effects. The section on radiolysis presents and discusses new experimental data generated for ground water and ground water plus borosilicate glass.

Measurable structural damage begins at a cumulative dose of about 1 x 10^{23} alpha decays/m^3 and "saturates" at a dose of approximately 5 x 10^{24} alpha decays/m^3. These doses correspond to cumulative doses expected for commercial glasses within the first 10,000 years of disposal in a geologic repository. A rough correlation between degree of structural damage, as measured by percent increase in solid volume, and enhancement of initial leach rate has been established for irradiated nuclear waste forms. The studies on the effect of radiation damage on potential repositories have focused on salt. Two aspects of radiation damage in NaCl that have received the most attention are colloid Sodium formation and stored energy.

Computer codes have been developed, and are being continually improved, to predict concentration of radiolytical produced species in simple ground water systems.

This excellent work is only the beginning in the understanding of the actual repository condition. More work is needed to understand the effects of radiation on radionuclide release rate from the waste form in the present of radiation field and appropriate barrier materials.

APPLICABILITY OF DATA TO LICENSING
[Ranking: key data ( ), supporting data (X)]

(a) Relationship to Waste Package Performance Issues Already Identified

Related to ISTP issues, 2.2.4.2, concerning the effects of radiation on the corrosion failure modes and associated corrosion rates for the waste package container and 2.2.4.2.1., concerning the predicted rate of radiolytic generation of hydrogen, oxygen and other species due to gamma radiation in the vicinity of the waste package container.

(b) New Licensing Issues

(c) General Comments on Licensing

NBS Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

E. I. duPont de Nemours and Company, Savannah River Plant
Aiken, South Carolina 29808.

(b) Author(s), Reference, Reference Availability

Boersma, M. D. and Mahoney, J. L., "Glass Making Technology for High Level Nuclear Waste", Document Identifier DP-MS-86-75. Paper proposed for presentation at the American Institute of Chemical Engineers Meeting, Boston, MA, August 24-27, 1986.

DATE REVIEWED: 8/17/87; Revised 4/2/88.

TYPE OF DATA

(1) Scope of the Report:

Descriptive paper discusses principles and design of an air-tight DWPF melter and associated unit processes in detail. Unit operations of batching, melting, and forming involved in the DWPF are discussed, with emphasis (in decreasing order of priority) on batching and melting.

(2) Failure Mode or Phenomenon Studied:

Wet batch processing and production of vitrified nuclear waste materials is described with particular emphasis on details and advantages of wet vs. dry batch processing and melting.

MATERIALS/COMPONENTS

Materials described are high-level nuclear waste sludge raffinate derivatives which are treated to yield soluble nonradioactive salts, soluble radioactive species, mercury, gases, and high-level radioactive metal hydroxide sludge precipitates. Radwaste materials are combined with glass frit for melting and nuclide immobilization.

TEST CONDITIONS

(1) State of the Material being Tested:

Sludge precipitates containing nonradioactive salts and radioactive metal hydroxide precipitates.

(2) Specimen Preparation:

Soluble, nonradioactive salts are separated from raffinate and disposed in solid low-level waste forms. The remaining raffinate sludge is further processed for incorporation in vitrified borosilicate glass waste forms. Glass matrix formed from carefully sized, premelted, crushed or ground frit mixed with treated sludge. Melting combines slurried glass matrix frit with slurried high level waste sludge. Gaseous melting byproducts and melting particulates are concentrated in the off-gas system and recirculated back through the melting process or filtered and discharged to the atmosphere following dilution.

(3) Environment of the Material being Tested:

Sludge resulting from raffinate neutralization is stored in an aqeous environment in selected DWPF steel underground liquid waste-storage tanks. Immobilizing glass is melted in a special air-tight melter in the DWPF facility. The melter is designed to channel off-gases either back into melting process or to post-melting treatment, incinerate residual waste stream organics, pyrolize nonvolatile organics in waste stream and other melting remnants. Molten glass package is poured into and contained within metal canisters in DWPF for long-term repository storage.

METHODS OF DATA COLLECTION/ANALYSIS

Prior testing of melter models and research scale melters used to accumulate operating and design data for present operational DWPF melter - (see present article reference 5.), K. R. Routt, "Modeling Principles Applied to the Simulation of a Joule-Heated Glass Melter", DP-1540, E. I. duPont, Savannah River Laboratory, 1980.

AMOUNT OF DATA

Figures

1. DWPF Slurry Processor (cutaway schematic)

2. Steam Stripping Efficiency vs. Mercury Concentration
   y-axis Water/Mercury Mass Boilup Ratio, 250 - 4250 (linear)
   x-axis Hg Concentration in Sludge, Wt% Dry Basis, 0 - 2.4% (linear)

3. DWPF Melter Feed System (cutaway schematic)

4. Mercury Displacement Pump (schematic)

5. DWPF Melter (cutaway schematic)

6. Riser and Pour Spout Heaters, Conceptual Cutaway View
   (cutaway schematic)

UNCERTAINTIES IN DATA

Uncertainties in existing data and performance parameters will, in the words of the authors on pages 15-16 of the reviewed report, "provide challenges to chemical engineers for years to come as performance is optimized and design improvements are made in replacement melters".

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

KEY WORDS

Radioactive waste fixation, vitrification, glass melter, wet slurry batching, slurry feeding, borosilicate glass frit, fused-cast chrome-alumina refractory.

GENERAL COMMENTS OF REVIEWER

Concepts in article detailed and well thought-out. Rationale for various unit operations within overall immobilization process consistent with good glass melting practice. Operating details draw on existing waste disposal and glass technology, or rely on data and operational practices derived from modeling and pilot scale studies.

APPLICABILITY OF DATA TO LICENSING
Ranking: key data ( ), supporting data (X)

(a) Relationship to Waste Package Performance Issues Already Identified:

Related to ISTP issues 2.7.2, how will the waste package design ensure that the radioactive wastes will be in solid form in a sealed container and 2.7.3, how will the waste packagé design ensure that particulate waste forms will be consolidated (for example, by incorporation into an encapsulating matrix) to limit the availability and generation of particulates?

(b) New Licensing Issues

(c) General Comments on Licensing

NBS Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

West Valley Nuclear Services Company, Inc., West Valley, NY.

(b) Author (s), Reference, Reference Availability, Date

Eisenstatt, L. R., Chapman, C. C., and Bogart, R. L., "A Method for Showing Compliance with High-Level Waste Acceptance Specifications," Waste Management '86, University of Arizona, Tucson, Arizona, 1986.

DATE REVIEWED: 5/12/87; Revised 9/23/87; 4/20/88.

TYPE OF DATA

Discussion of the methods, including process analysis, that the WVDP will be using to construct the waste form, and to show that the waste form will meet the preliminary specification of D.O.E.

MATERIALS/COMPONENTS

WV-205, Defense Waste Reference Glass (DWRG), Stainless steel ASTM A240, UNS Designation S30400/ Canister, Purex C-Sampler.

TEST CONDITIONS

None given.

METHODS OF DATA COLLECTION/ANALYSIS

None given.

AMOUNT OF DATA

Tables

1. Composition of WV-205 (the standard glass that was recently selected.)
2. Variations on WV-205 that will be studied for radionuclide release rate.

Figures

1. West Valley vitrification flow diagram. Feed sample locations during nonradioactive process model testing are designated (in the flow sheet.)
2. West Valley glass variability approach. (The acceptable glass area and the test boundary are shown within a triangular diagram.)
3. Purex C-Sampler. (The sampling system that is now being tested.
4. Leach rates versus residence time results from the partial exchange interactive flow test for WV-205 and DWRG glasses. The normalized leach rate and the annual fractional loss per year is shown. Time scale: 0.001 - 1 (y), Leach rate: 10^{-3} - 1 ($g.m^{-2}d^{-1}$).
5. West Valley canister. (Schematic of the preliminary design for the West valley canister is shown.)
6. West Valley canister grapple. (Conceptual design of the grapple being tested at West Valley is shown.)

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

KEY WORDS

Planned work, process analysis, simulated field, stainless steel, defense high-level waste (DHLW).

GENERAL COMMENTS OF REVIEWER

This paper deals with the approaches that will be used by the West Valley Demonstration Project (WVDP) to show that the West Valley HLW glass waste form product will meet the preliminary specifications of D.O.E. for waste disposal.

The glass that will be generated will be characterized for chemical composition, crystallinity, and radionuclide release. During the process the characteristics of the glass will be monitored by measurements of the viscosity and conductivity at the melting temperature. Initial testing will take place with nonradioactive simulated waste glass. The WVDP will attempt to prove that it is not necessary to sample the radioactive end-product (i.e., the actual waste glass) and that the composition of the product can be obtained from sampling the Concentration Feed Makeup Tank (CFMUT).

The most important issue in deciding whether such testing is acceptable is the degree of homogenization of the sludge in the melter. One of the important parameters in our understanding of the homogenization process is the Residence Time Distribution (RTD) in the melter. This function depends on the viscosity, temperature, density, and feed rate. It is possible to measure the RTD as a function of the process parameters using a radioactive tracer technique[1].

These measurements, together with other measurements discussed in this report, and, most important, occasional sampling of the glass form, would be acceptable. It is our understanding that WVDP has now agreed to undertake spot sampling of the glass waste form.

RELATED HLW REPORTS

1. Wolf, D. and White, D., "Experimental Study of the Residence Time Distribution in Plasticating Screw Extruder," AIChE Journal 22, 122-131, (1976).

APPLICABILITY OF DATA TO LICENSING

[Ranking: key data ( ), supporting data (X)]

(a) Relationship to Waste Package Performance Issues Already Identified

Related to ISTP issue 2.3.1., what are the physical, chemical and mechanical properties of the waste form?

(b) New Licensing Issues

(c) General Comments

AUTHOR'S ABSTRACT

The West Valley Demonstration Project is in the process of showing that the West Valley high-level waste product will be acceptable for disposal. The methods that are being considered emphasize testing nonradioactive components and relating them to the radioactive production product. Glass and canisters processed at the Component Test Stand at West Valley will be studied to provide the basis for showing that the tested components are similar to those that will be produced during production. This testing will include defining and testing glass compositions that may be generated, process model development and verification, and canister design and testing. Administrative controls will need to be instituted to ensure that restricted materials are not included in the canistered waste form and to ensure that the proper materials are procured. During production accurate records will need to be kept.

NBS Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

E.I. duPont de Nemours & Co., Savannah River Laboratory, Aiken, South Carolina 29808.

(b) Author(s), Reference, Reference Availability, Date

Boersma, M. D., "Process Technology for Vitrification of Defense High-Level Waste at the Savannah River Plant", paper prepared for presentation and publication in the proceedings of the American Nuclear Society Meeting, Fuel Reprocessing and Waste Management, Jackson Hole, Wyoming, DP-MS-83-135, August 26-29, 1984.

DATE REVIEWED: 8/17/87; Revised 4/4/88.

TYPE OF DATA

(1) Scope of the Report:
The descriptive paper presents a process overview, a review of the characteristics of the high-level radiaoactive waste to be processed, descriptions of unit process operations, and process parameters used to vitrify high-level radioactive waste.

(2) Failure Mode or Phenomenon Studied:

Description of DWPF high-level radwaste vitrificatión process.

MATERIALS/COMPONENTS

Detailed in paper are high-level, liquid sludge and slurry radioactive waste forms and volumes, chemical compositions of the radwaste sludge and salts resulting from initial waste treatment, and calculated inventories of wastes on a "design basis" (Table IV).

TEST CONDITIONS

(1) State of the Material being Tested:

Physical, chemical, and radiological characteristics of anticipated alkaline liquid sludge, saltcake, and saturated salt solution waste are detailed for the Savannah River Plant (SRP).

(2) Specimen Preparation:

Processing of SRP waste to remove salts, precipitates, and sludge is described. Decontamination procedures of the bulk of the salt solution for disposal as low-level chemical waste disposal are detailed. Concentration processing of the remaining radionuclides for vitrification is also described, as are preparation of intermediate process by-products.

(3) Environment of the Material being Tested:

Radwaste processing conducted mainly in a liquid and radioactive environment. Salt separation and solids pretreatment uses specially-designed equipment within existing, sealed waste tanks outside the DWPF building. Decontaminated salt solutions will be blended with Portland cement in a separate facility for earthen trench subsurface burial. Radionuclide separation and processing will occur in the DWPF building. Radwaste sludge will be further processed within the DWPF building to adjust physical slurry properties, and to remove gases and mercury compounds to prepare slurry feed stock for vitrification. Vitrification, canister filling, and canister decontamination will also be carried out within the DWPF.

METHODS OF DATA COLLECTION/ANALYSIS

Parameters shaping the scope of the radwaste disposal program are based both on existing calculated radwaste inventories and DWPF goal to reduce inventories in next 10-15 years. Composition of material to be processed and vitrified in that time period is based on anticipated "design basis" of radionuclide inventory.

AMOUNT OF DATA

Figures

1. Defense High-Level Waste Treatment at the Savannah River Plant (process flow chart)

2. Sludge Receipt and Adjustment Tank (cutaway schematic)

3. DWPF Melter (cutaway schematic)

4. Off-Gas Film Cooler with Brush Reamer (cutaway schematic)

5. Welding Process (cutaway schematic)

Tables

I. SRP High Level Waste Volume, lists Column 1 - volumes (cubic meters) of sludge, salt cake, and salt solution in current inventory and Column 2 - respective rates of increase (cubic meters per year).

II. Chemical Ingredients of SRP Waste Sludge (Dry Basis), Column 1 - 12 sludge ingredients, Column 2 - as received weight percentage range, and Column 3 - expected DWPF feed range weight percentage.

III. Composition of SRP Salt Waste, Column 1 - 12 SRP salt waste compounds, and Column 2 - weight percentage of each on a dry basis.

IV. Aged SRP Waste Radionuclides, Column 1 - 22 isotopes in ages SRP waste, Column 2 - half-life in years, Cloumn 3 - Ci/L in 5 year old sludge, Column 4 - mg/L of 5 year old sludge, Column 5 - Ci/L of 15 year old supernate, and Column 6 - mg/L of 15 year old supernate.

V. Typical DWPF Glass Frit, lists cloumn 1 - 5 frit compounds, and Column 2 - weight percentage of each.

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

KEY WORDS

Design, simulated field, Savannah River Plant, air, ambient temperature, basic solution, defense high level waste (DHLW), vitrification, borosilicate glass, high-level radioactive waste, low-power alkaline waste, glass melting, off-gas treatment, slurry, sludge, salt solution, frit, melter, stainless steel canister, canister decontamination, upset resistance weld.

GENERAL COMMENTS OF REVIEWER

Paper is a good, detailed overview of the vitrification processing of defense high-level waste at the Savannah River Plant. This NBS review contains only a summary of the contents of the technical report. It contains neither critical commentary nor analyses by NBS staff and it will be included in the "Database for Reviews and Evaluations on High-Level Waste (HLW) Data."

APPLICABILITY OF DATA TO LICENSING
Ranking: key data ( ), supporting data (x)

(a) Relationship to Waste Package Performance Issues Already Identified

Related to ISTP issue 2.3.2, concerning the physical, chemical and mechanical properties of the waste form.

(b) New Licensing Issues

(c) General Comments on Licensing