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

August 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: Monthly Letter Status Report for July 1988 (FIN-A-4171-7)

Dear Mr. Peterson:

Enclosed is the July 1988 monthly progress report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-7). The financial information is attached to this letter.

Sincerely,

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

Enclosures

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

Wm-11

Monthly Letter Report for July 1988

Published August 1988

(FIN-A-4171-7)

Performing Organization: National Bureau of Standards (NBS)
Gaithersburg, MD 20899

Sponsor: Nuclear Regulatory Commission (NRC)
Office of Nuclear Materials Safety and Safeguards
Washington, DC 20555

TASK 1 -- REVIEW OF WASTE PACKAGE DATA BASE

During June and July, the database was converted from one management system, Revelation®, to another, Advanced Revelation® 1.0. This conversion required a complete reprocessing of all 15 files used to store information. New and improved input entry screens were developed. Work on a prototype search program will begin in August. Staff retraining has begun, and the capabilities of this advanced system are regarded by staff to be considerably improved.

Appended to this report are the following three Draft Reviews not previously submitted (see p. 10 to 28). Comments by NRC and its contractors are solicited.

1. UCRL-53631, "Reaction of Topopah Spring Tuff with J-13 Water: A Geochemical Modeling Approach Using the EQ3/6 Reaction Path Code," November 1985.

2. UCRL-21005, SANL 616-007, "Corrosion Testing of Type 304L Stainless Steel in Tuff Groundwater Environments," November 1987.

3. UCRL-94708, "Carbon-14 in Waste Packages for Spent Fuel in a Tuff Repository," October 1986.

Status of Database

986 Document citations in HLW database.
78 Completed reviews in HLW database (taken from Vols. 1 to 4).
15 Draft and completed reviews for Vol. 5.

Papers currently being reviewed (Category 1), review when time permits (Category 2) and file with cross reference(s) to related report(s) (Category 3).

Status of Recently Listed Reviewable Documents

NNWSI

8 NNWSI reports currently under review (Category 1).
14 NNWSI reports to review when time permits (Category 2).
0 NNWSI reports to file with cross reference(s) to other reports (Category 3).
3 NNWSI reports identified and not yet categorized.
39 NNWSI reports received and not yet categorized.

GLASS -- VITRIFIED WASTE FORM

1 Reports currently under review (Category 1).
4 Reports to review when time permits (Category 2).
0 Reports to file with cross reference(s) to other reports (Category 3).
0 Reports identified and not yet categorized.

Database searches for the month of July, 1988 include the DOE Energy Data Base, the NTIS, Metadex (American Society for Metals), and Compendex Plus (Engineering Information Inc.). Thirteen articles were selected (to order). Some of these are for use as library references and others will be reviewed by NBS. Examples of the search conducted for each of these databases are attached at page numbers 29 to 34. Two of these databases have been added just this month, so as to widen the NBS search for all pertinent HLW literature: Metadex and Compendex Plus. For searches of these two databases, additional work is required to optimize the method of the search and the targeting of specific articles pertinent to waste-package materials and problems.

STATUS OF REVIEWS OF NNWSI REPORTS

NNWSI -- Reports recently identified for review

Three publications were identified for review this month. They deal with the subjects of 1) glass leaching, 2) copper corrosion, and 4) site characterization.

Using glass samples doped with $_{99}$Tc, ^{237}Np, ^{238}U, and ^{239}Pu, to simulate waste glass, laboratory leaching tests were performed in J-13 water at 90° [Bazan 1987]. Glass samples were held in tuff containers with some samples held by stainless steel supports. This assembly was placed in a teflon dish. The results show varied leaching behavior for radionuclides, with some migrating through the tuff, others sorbing on the inner surfaces of the container.

The corrosion of copper in HCl solutions containing H_2O_2 has been studied, using a rotating ring-electrode technique, where the sample is exposed to various electrolyte flow rates (Smyrl 1986]. The authors indicate that events occurring on the copper surface are complex and strongly influenced by the concentration of copper ions in solution.

This report describes the waste package environment and the types of in-situ tests that might be carried out to characterize the flux and flow mechanisms of water in the tuff repository [Yow 1985]. The tests are intended to measure the movement of water through pores and fractures of tuff, as affected by the thermomechanical behavior of the site.

1. Bazan, F., Rego, K. H., and Aines, R. D., "Leaching of Actinide-Doped Nuclear Waste Glass in a Tuff-Dominated System," UCRL-94721, January 1987.

2. Smyrl, W. H., Bell, B. T., Atanososki, R. T., and Glass, R. S., "Copper Corrosion in Irradiated Environments. The Influence of H_2O_2 on the Electrochemistry of Copper Dissolution in HCl Electrolyte," UCRL-95961, December 1986.

3. Yow, J. L., Jr., "Concept of Waste Package Environment Tests in the Yucca Mountain Exploratory Shaft," UCID-20450, May 1985.

NNWSI --

Category 1 -- Reports currently being reviewed

1. HEDL-TME 85-22, "Results from Cycles 1 and 2 of NNWSI Series 2 Spent Fuel Dissolution Tests," May 1987.

2. UCRL-21019, SAN-662,-027, "Recent Results from NNWSI Spent Fuel Leaching/Dissolution Tests," April 1987.

3. UCID-21272, "Plan for Spent Fuel Waste Form Testing for NNWSI," February 1987.

4. UCRL-53795, "Reaction of Vitric Topopah Spring Tuff and J-13 Ground Water under Hydrothermal Conditions Using Dickson-Type, Gold-Bag Rocking Autoclaves," November 1986.

5. UCRL-21013, "Summary of Results from the Series 2 and Series 3 NNWSI Bare Fuel Dissolution Tests," November 1987.

6. SAND85-7117, "A First Survey of Disruption Scenarios for a High-Level Waste Repository at Yucca Mountain, Nevada," December 1985.

7. UCRL-94633, "Experimental Study of the Dissolution Spent Fuel at 85°C in Natural Groundwater," December 1986.

8. UCRL-53761, "Waste Package Performance Assessment: Deterministic System Model Program Scope and Specification," October 1986.

Category 1 (continued) - Status of Reviews not yet sent to NRC and WERB

Document No.	Assigned to Reviewer	First Draft Completed	Lead Worker	Program Manager
HEDL-TME 85-22	x			
UCRL-21019	x			
UCID-21272	x			
UCRL-53795		x		
UCRL-21013	x			
SAND85-7117		x		
UCRL-94633			x	
UCRL-53761	x			

Category 2 -- Review as time permits

1. UCRL-95962, "Hydrogen Speciation in Hydrated Layers on Nuclear Waste Glass," January 1987.

2. UCRL-94658, "Integrated Testing of the SRL-165 Glass Waste Form," December 1986.

3. UCRL-91258, "Leaching Savannah River Plant Nuclear Waste Glass in a Saturated Tuff Environment," November 1984.

4. ANL-84-81, "NNWSI Phase II Materials Interaction Test Procedures and Preliminary Results," January 1985.

5. HEDL-7540, "Technical Test Description of Activities to Determine the Potential for Spent Fuel Oxidation in a Tuff Repository," June 1985.

6. HEDL-SA-3627, "Predicting Spent Fuel Oxidation States in a Tuff Repository," April 1987.

7. UCRL-15976, SANL-522-006, "Microstructural Characteristics of PWR Spent Fuel Relative to its Leaching Behavior", April 1985.

8. UCRL-96703, "Geochemical Simulation of Dissolution of West Valley and DWPF Glasses in H-13 Water at 90°C," November 1987.

9. UCRL-96555, Rev. 1, "Thermodynamic Data Bases for Multivalent Elements: An Example for Ruthenium," November 1987.

10. UCRL-96702, "Geochemical Simulation of Reaction Between Spent Fuel Waste Form and J-13 Water at 25°C and 90°C," November 1987.

11. UCRL-53645, "Hydrothermal Interaction of Solid Wafers of Topopah Spring Tuff with J-13 Water and Distilled Water at 90, 150, and 250°C, Using Dickson-Type, Gold-Bag Rocking Autoclaves," September 1985.

12. UCRL-53702, "Spent Fuel Test - Climax: An Evaluation of the Technical Feasibility of Geologic Storage of Spent Nuclear Fuel in Granite," March 1986.

13. UCID-21274, "Plan for Integrated Testing for NNWSI Non EQ3/6 Data Base Portion," May 1987.

14. UCRL-96318, Ramirez, W. L. and Daily, W. D., "Electromagnetic Experiment to Map in Situ Water in Heated Welded Tuff: Preliminary Results," March 1987.

Category 3 -- File and cross reference

None this month.

VITRIFIED WASTE FORM --

Category 1 -- Reports currently being reviewed

The NBS review of PNL-5157, "Final Report of the Defense High-Level Waste Leaching Mechanisms Program" is continuing: Chapter 3, "Environmental Interaction," is expected to be available in late September. Chapter 4, "Dissolution of Specific Radionuclides," is expected to be completed shortly thereafter. Chapter 6, "Phenomenological Models of Nuclear Waste Glass Leaching" has been assigned and a first draft has been received. The other chapters of PNL-5157 have already been reviewed by NBS.

1. PNL-5157, "Final Report of the Defense High-Level Waste Leaching Mechanisms Program," August 1984.

Status of Reviews not yet sent to NRC and WERB

Document No.	Assigned to Reviewer	First Draft Completed	Lead Worker	Program Manager
PNL-5157 Chapter 3		x		
PNL-5157 Chapter 4	x			
PNL-5157 Chapter 6		x		

Category 2 -- Review as time permits

1. "Large Scale Leach Testing of DWPF Canister Sections," Proceedings of the Materials Research Society Symposium, "Scientific Basis for Nuclear Waste Management X," December 1986.

2. "Waste Glass Leaching: Chemistry and Kinetics," Proceedings of the Materials Research Society Symposium, "Scientific Basis for Nuclear Waste Management X," December 1986.

3. PNL-6353, "Comprehensive Data Base of High-Level Nuclear Waste Glasses: September 1987 Status Report: Volume 2, Additional Appendices," December 1987.

4. DOE/NE/44139--34, "Preliminary Results of Durability Testing with Borosilicate Glass Composition," January 1987.

Category 3 -- File and cross reference

None this month.

TASK 2 -- IDENTIFICATION OF ADDITIONAL DATA REQUIRED AND IDENTIFICATION OF TESTS TO GENERATE THE DATA

NBS lead workers are continuing their studies concerning the types of additional data and verification tests needed to demonstrate that the DOE waste package designs will meet the performance objectives of 10 CFR 60.

Work is continuing on consolidation of the various recommendations that have been made to date by the NBS.

TASK 3 -- LABORATORY TESTING

The work on each of the three projects reported below is on schedule with the work statements listed in their respective proposals. The work conducted in July 1988 is reported below. Work conducted in previous months was reported earlier. Summaries of two of the current laboratory testing studies were presented earlier and for the other two studies summaries will be given soon.

Title of Study: Evaluation of Methods for Detection of Stress Corrosion Crack Propagation in Fracture Mechanics Samples.
Principal Investigator: Charles Interrante

The test apparatus was rebuilt in June. Two tests were conducted in July and these resulted in the following advancements: (1) The production of the desired slow-crack extension in one test specimen, in the first test which used a hard (Rc 37) test specimen, and (2) A considerable improvement in the sensitivity of the measurement system to the detection of the acoustic energy from test specimens used in these tests; this was accomplished in the second test conducted in July. The second test used a softer (Rc 23) test specimen that has the advantage of being relatively less prone to the formation of branch cracks, which can be important to the interpretation and collection of the data.

In earlier tests, the specimen was held in a rubber stopper, which tends to greatly attenuate the acoustic signal. In the second test, the specimen was placed at the bottom of the test chamber on two glass rods. In addition, the transducer used to detect the acoustic signal was actually attached to the test specimen with set screws. In the past, the transducer was spring loaded onto the end of the test specimen. The set-spring method has the advantages of being both more firm and more reproducible. To date, the sensitivity of the acoustic detection system has been measured only qualitatively: A piano wire is snapped from about 1-1/2 inches above either the test chamber or the test specimen and the acoustic signal is observed using a digital oscilloscope.

In this second test, the acoustic transducer and the lead wires used for the measurements each had to be coated to retard degradation from the acidic environment of the test. The new configuration of the test apparatus worked very well and the acoustic and electric-resistance measurements functioned as expected. However, this second test specimen showed no measurable crack extension, and so collection of acoustic data was not possible. Usable data was not collected on the first test specimen, as during the period of crack extension, one of the leads, which are used to measure crack length by the electric-resistance method, disbonded from the attachment point on the end of the test specimen. No further crack extension was observed after this lead was rewelded to the specimen. The next test is planned for September.

Title of Study: Effect of Resistivity and Transport on Corrosion of Waste Package Materials.
Principal Investigator: Edward Escalante

A set of tests using the variables permeability (which is controlled using aqueous solutions, agar, and sand) and resistivity (which is controlled by the concentration of sodium chloride) were completed earlier. Evaluation of the data from this first experiment is continuing. From that work, measurements of permeation of oxygen through agar showed that the permeability decreased during the period of the experiment. This effect was so pronounced that the agar became less permeable to oxygen than a similar solution in sand. Part of this effect can be attributed to the growth of bacteria, which changed the character of the agar. Another important observation made concerning data taken in agar is that any loss of moisture in the agar caused the agar to draw away from the electrodes immersed in the media.

Title of Study: Pitting Corrosion of Steel Used for Nuclear Waste Storage
Principal Investigator: Anna C. Fraker

The status of this project was given in the May, 1988 monthly report. Work continues on data analysis and preparation of a paper for presentation. Data for A27 steel tested in media of varying salinity and ionic content show that the corrosion rate for A27 steel is 7.08 uA/cm^2 (3.35 mpy) in concentrated J-13 water at a temperature of 21.5°C and a pH of 8.6. This corrosion rate is less than that found in a higher pH media with increased ionic content (simulated Grande Ronde 4 water). The corrosion rate in the GR 4 water was 11.37 uA/cm^2 (5.25 mpy) in water at a temperature of 22°C and with a pH of 9.75. Tests of the A27 steel in 3.5 percent NaCl showed an initial corrosion rate of 25.32 uA/cm^2 which decreased dramatically to 2.64 uA/cm^2 after three days. A thick surface film develops, which is non-protective but limits diffusion and keeps the corrosion rate low. Any disturbance of this film, such as it falling off, would result in a large increase in the corrosion rate. These tests are being conducted at 95°C at present. Experimentally, there was a problem with the coating material used to isolate the specimen lead and expose only the selected specimen area to the solution at 95°C over an extended time. A more heat resistant coating material is being used for at present. Modifications are being made to teflon holders which also are used for specimen mounting.

Future plans include testing steels in the welded condition.

Title of study: Corrosion Behavior of Zircaloy Nuclear Fuel Cladding
Principal Investigator: Anna C. Fraker

The status of the work of this project was given in the May, 1988 monthly report. An update that lists work-in-progress follows:

Zircaloy-4 specimens are exposed to concentrated J-13 water at 22°C and 95°C for long-term testing. The initial electrode potential (-0.362 V vs. S.C.E.) shifts with time to a more positive potential (0.204 V vs. S.C.E. after one day). The corrosion rate is essentially nil in both cases. Contact will be made in the future with Babcock and Wilcox Co. or another appropriate supplier to prepare welded samples of the Zircaloy materials. Welded areas will be analyzed and tested for any corrosion-related effects on durability.

TASK 4 - GENERAL TECHNICAL ASSISTANCE

A response to your informal request for general technical assistance on the document "Preliminary Safety Analysis Report (PSAR), Vitrification System, Revision III West Valley Demonstration Project SAR Volume III," August 1985 was prepared by E. Plante and consultant Mr. B. Adams and transmitted to you on June 27, 1988.

A response to your informal request for general technical assistance on the documents "Waste Acceptance Preliminary Specifications for the Defense Waste Processing Facility High-Level Waste Form, (OGR/B-8, Defense Waste Processing Facility, December 1986 and OGR/B-9, West Valley Demonstration Project, April 1987) was prepared by Dr. E. Plante and consultant, Mr. B. Adams and transmitted to you on June 27, 1988.

A response to your informal request for general technical assistance on the document entitled "Waste Compilance Plan for the West Valley Demonstration Project High-Level Waste Form," WVDP-055, West Valley, New York, 1986 was prepared by Dr. A. Fraker and Dr. E. Plante and transmitted to you on July 7, 1988.

Dr. Charles Interrante attended a ASTM C26 Committee meeting on Nuclear Fuel Cycles in Toronto, Canada on July 24-28, 1988. His trip report is attached (pages 35 to 38).

DRAFT

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

DATA SOURCE

(a) Organization Producing Report

Work performed under the auspices of the U. S. Department of Energy by the Lawrence Livermore National Laboratory under Contract No. W-7405-Eng-48.

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

Delany, J. M., "Reaction of Topopah Spring Tuff with J-13 Water: A Geochemical Modeling Approach Using the EQ3/6 Reaction Path Code," UCRL-53631, November 25, 1985.

DATE REVIEWED: 5/21/88; Revised 6/13/88; 7/17/88.

PURPOSE

To investigate the usefulness of the EQ3/6 geochemical modeling code as a tool to reproduce theoretically the physical/chemical environments of the NNWSI waste package.

CONTENTS

The report consists of 46 pages, containing four tables and six figures in the main body of the text and three appendixes. Two pages provide a general description of the EQ3/6 software, ten pages give the code input parameters, ten pages discuss the four different computer simulations reported and the results, and there is a two-page summary. The experimental data characterizing the J-13 well water and the bulk mineralogy of the Topopah Spring Tuff are discussed in five pages. The bibliography covers six pages and includes 54 references. The three appendixes cover eight pages and provide data and sample computer input files.

TYPE OF DATA

(1) Scope of the Report

Description of EQ3/6 Reaction Path Code; a comparison of specific experimental data (taken from UCRL-53630 and UCRL-53645) with the results of computer modeling of the reaction of Topopah Spring tuff with J-13 groundwater.

(2) Failure Mode or Phenomenon Studied

Dissolution of Topopah Spring Tuff in J-13 groundwater over time (computer simulation).

MATERIALS/COMPONENTS

Topopah Spring Tuff from Drill hole USW G-1 at 1232 feet.

J-13 groundwater from USGS Test Well 6 1963.

TEST CONDITIONS

(1) State of the Material being Tested

In the quoted experimental work:

Tuff was tested in both crushed and wafer forms.

J-13 groundwater from USGS Test Well 6 1963, stored in a plastic-lined drum filled directly at the well. Storage conditions do not significantly affect overall composition, but tests indicated that the storage tank became saturated with oxygen and the pH rose from 7.1 to 7.6 during storage.

(2) Specimen Preparation

In the quoted experimental work:

1. Crushed samples of tuff were dry-sieved to a size range of 75 to 150 μm. The samples were unwashed and probably contained fine particulate material.

2. Core-wafer samples were cut, polished, washed thoroughly in distilled water, and cleaned in an ultrasonic bath to remove all fine particles from the surfaces; effective porosity 6.5%, measured bulk density 2.335 g/cm^3, grain density 2.522 g/cm^3.

(3) Environment of the Material being Tested

In the experimental work quoted, the tests of tuff reacting with J-13 water were run at 150 and 250°C.

METHODS OF DATA COLLECTION/ANALYSIS

The laboratory data were gathered by other workers at Lawrence Livermore National Laboratory. Details are not given in this report except that the experiments were run at 150 and 250°C for 66 days on the samples described above.

Geochemical modeling was done using the EQ3/6 reaction path code to model the complex chemical interactions between groundwater and repository host rock. The EQ3/6 code consists of two large FORTRAN codes which are supported by a common thermodynamic database. One code is a speciation-solubility code that computes a model of the state of an aqueous solution. The program produces the distribution of aqueous species, their thermodynamic activities and saturation indices for various solids. The second program can be used to compute models of the evolution of aqueous geochemical systems (the reaction path). Changes in rock/water systems are calculated as the reactions proceed toward a state of overall chemical equilibrium. The thermodynamic data are processed through a code which checks for thermodynamic consistency, extrapolates heat capacity functions with temperature, and generates data for insertion into the master thermodynamic data file. The internal database contains free energies and enthalpies of formation, third law entropies, and heat capacities of reactions commonly used in geochemical calculations. The program also checks for mass and charge balance, fits all data to a predetermined temperature grid.

This report describes the application of the codes to help evaluate chemical reactions occurring in closed-system high-temperature experiments. The interaction between tuff and J-13 water is modeled as the reaction of an aqueous solution in contact with a specific mineral assemblage. The bulk-rock mineralogy was represented as an assemblage of cristobalite, alkali feldspar, and quartz with minor amounts of biotite, plagioclase, and montmorillonite. Analyses of natural J-13 water served as direct input for the composition of the initial fluid. The EQ6 code predicts the reactions of the rock and water compositions at the experimental temperatures for the designated time intervals. The reaction products produced by the EQ6 code were compared with experimental data.

AMOUNT OF DATA

There are four tables and six figures in the main body of the text and three appendixes.

Table 1.: "Average J-13 fluid analysis for LLNL laboratory supply," lists the pH, and the concentration in milligram/liter for 14 ions in the water.

Table 2.: "Composition of Topopah Spring tuff," lists six mineral phases by name, by mineral composition, and the volume percent of each mineral in the tuff.

Table 3a.: "EQ6 rock recipe for crushed Topopah Spring Tuff," lists for six mineral species, the weight percent, the moles of reactant in moles/gram, the initial number of moles of reactant, the specific surface area in square centimeters/gram, and the total surface area in square centimeters.

Table 3b.: "EQ6 rock recipe for G-1 Topopah Spring Tuff core wafer," lists for six mineral species, the weight percent, the moles of reactant in moles/gram, the initial number of moles of reactant, the specific surface area in square centimeters/gram, and the total surface area in square centimeters.

Table 4.: "Steady state J-13 composition (open system 150°C, 100 years; see text)," lists the pH and the concentration in milligrams/liter for 11 ionic components.

Figure 1.: "Log activity plot of potassium vs aqueous silica at 25°C (see text). The open squares represent J-13 compositions from various literature sources. The crosses represent concentrations of the LLNL laboratory supply." Log activity a_{K+}/a_{H+} (from 0 to 6) is plotted against log activity a_{SiO2} (from -6 to 0) with five mineral areas identified in the plot.

Figure 2.: "Log activity plot of sodium vs aqueous silica at 150°C. See Figure 1 for symbol explanation." Log activity a_{Na+}/a_{H+} (from 2 to 8) is plotted against log activity a_{SiO2} (from -6 to 0) with six mineral areas identified in the plot.

Figure 3a.: "Concentration of Si and Na vs time for 150°C Topopah Spring Tuff (Tpt) core-wafer simulation. Symbols represent analytical values from Knauss et al. (1985b)." The concentration in ppm (from 0 to 140) is plotted against the time in days (from 0 to 70).

Figure 3b.: " Concentration of Ca, K, Al, and Mg vs time for 150°C Tpt core-wafer simulation." The concentration in ppm (from 0 to 12) is plotted against the time in days (from 0 to 70).

Figure 4a.: "Concentration of Si and Na vs time for 250°C Tpt core-wafer simulation. Symbols represent analytical values from Knauss et al. (1985b)." The concentration in ppm (from 0 to 400) is plotted against the time in days (from 0 to 70).

Figure 4b.: " Concentration of Ca, K, Al, and Mg vs time for 250°C Tpt core-wafer simulation." The concentration in ppm (from 0 to 14) is plotted against the time in days (from 0 to 70).

Figure 5a.: "Concentration of Si and Na vs time for 150°C Tpt Crushed - tuff simulation. Symbols represent analytical values from Knauss et al. (1985a)." The concentration in ppm (from 20 to 160) is plotted against the time in days (from 0 to 70).

Figure 5b.: " Concentration of Ca, K, Al, and Mg vs time for 150°C Tpt crushed-tuff simulation." The concentration in ppm (from 0 to 14) is plotted against the time in days (from 0 to 70).

Figure 6.: "Concentration vs time plot for 150°C Tpt core-wafer 100-year simulation." Concentrations in ppm (from 0 to 200) of Si, Na, Al, and Ca are plotted against time in years (from 0 to 100).

Appendix A.: "Constants used to construct theoretical rock recipes," lists the numerical values of the densities and molecular weights for six mineral species.

Appendix B.: "Sample EQ3NR Input File of J-13 Water," lists the parameters and the values used in the computer simulations.

Appendix C.: "Sample EQ6 Input File for Core-Water Experiment," lists the data used in the run at 150°C.

UNCERTAINTIES IN DATA

Uncertainties in the experimental data are not given in this report.

DEFICIENCIES/LIMITATIONS IN DATABASE

It is not possible to analyze the alteration products of the crushed-tuff experiments in the same detail possible in the core-wafer experiments. In the core-wafer experiments only a negligible amount of the solids appear to have reacted and so the concentrations of major cations in solution are a sensitive measure of the course of the reaction. Only if good agreement is found between the progressive changes in the theoretical and analytical solution compositions can the EQ6 results be interpreted to predict the secondary alteration that may form in these experiments.

The author states that there is a lack of complete thermodynamic data and also a lack of some of the specific data for species dissolution, for instance, data on the dissolution of smectite phases.

KEY WORDS

Data Anaylsis, computer modeling, solubility, EQ3/6, laboratory, J-13 water, tuff composition, tuff, dissolution.

CONCLUSIONS

The author states that the dissolution of Topopah Spring tuff in J-13 water may be reasonably approximated and that a closed-system environment at 150°C can be simulated with a minimum number of constraints in code parameters but the simulation does not work equally well at 250°C. Future work must include more data on various mineral compositions. The addition of several rate laws for precipitation kinetics will be made to the EQ3/6 software.

GENERAL COMMENTS

The validity of the original data being used in modeling must always be a weak point. For instance, the rate constants for dissolution of an individual mineral component, determined experimentally for that component by itself, may not be valid for the component in association with other mineral phases. Also, the assumptions concerning the identity of all mineral species in the tuff rock may not be inclusive enough. The uncertainty of the data is difficult to estimate. The dependence of the results of modeling on the validity of the assumptions made about any complex system is another weakness.

The author herself states that although the data for experiments at 150°C agree well with the modeling results, the results at 250°C indicate that the database cannot adequately model the system at the higher temperature. The modeling results validate the modeling procedures but the database needs expansion. Too much of the database must consist of estimated thermodynamic values and better data must be inserted in the database.

RELATED HLW REPORTS

1. Aines, R. D., "Application of EQ3/6 to Modeling of Nuclear Waste Glass Behavior in a Tuff Repository," UCID-20895, May 1986.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

This report relates to NNWSI ISTP issue 2.3.2.1.2, the rates of dissolution associated with the potential waste form dissolution mechanisms.

(b) New Licensing Issues

(c) General Comments

DRAFT

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, Pacific Northwest Laboratory, Richland, Washington 99352.

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

Westerman, R. E., Pitman, S. G. and Haberman, J. H., "Corrosion Testing of Type 304L Stainless Steel in Tuff Groundwater Environments," UCRL-21005, SANL 616-007, November 1987.

DATE REVIEWED: 9/26/88

PURPOSE

"The purpose of this study was to evaluate the stress corrosion cracking resistance of solution treated and sensitized 304 and 304L stainless steel at elevated temperatures in tuff rock and tuff groundwater under irradiated and unirradiated conditions by U-bend and slow strain rate tests."

CONTENTS

This report consists of 67 pages which include a summary, introduction, 27 Figures, and 9 Tables.

TYPE OF DATA

(1) Scope of the Report: This report covers the results of experiments designed to evaluate 304 and 304L stainless steel as a container alloy.

(2) Failure Mode Studied: Stress Corrosion Cracking, Corrosion

MATERIALS/COMPONENTS

This report focusses on the behavior of 304 and 304L stainless steel. However, samples from the following six different alloys or heats were used for these experiments:

Code No.	Material
1	Wrought 304L SS sheet, 1.52 mm thick
2	Wrought 304 SS sheet, 1.52 mm thick
3	Wrought 304L SS plate heat A, 6.35 mm thick
4	Wrought 304L SS plate heat B, 6.35 mm thick
5	Wrought 304 SS plate, 6.35 mm thick and
6	Wrought 316L SS plate, 6.35 mm thick.

The chemical compositions and the mechanical properties of these materials are included in this report as tables 3.1.I and 3.1.II respectively. To distinguish between these materials in this report, the authors use the material code numbers in this review.

TEST CONDITIONS

(1) The State of the Material Being Tested

The alloys were tested in two different conditions of heat treatment: (1) solution annealed and (2) solution annealed and sensitized. The intention was to represent the best and worst possible heat-treatment conditions of the container. However, various heat treatments were used to achieve these two conditions and the authors do not make a clear presentation of the specific heat treatments used for all samples. The following table is an attempt to assemble heat-treatment information taken from various parts of the report and may be incomplete or contain errors.

Code No.	Solution Anneal	Sensitized Anneal
1	1050°C, 15 m[1]	600°C, 24 h, air cool (ac)
2(a)	1050°C, 15 m[1]	550°C, 24 h, ac
2(b)	1050°C, 15 m[1]	700°C, 24 h, water quench (wq)
2(c)	1050°C, 15 m[1]	700°C, 8 h, wq
3(a)[2]	1050°C, 15 m[1]	600°C, 24 h[1]
3(b)[2]	1050°C, 15 m, wq	600°C, 10 h[1]
4(a)[2]	1050°C, 15 m[1]	600°C, 24 h[1]
4(b)[2]	1050°C, 15 m, wq	600°C, 10 h[1]
5	Mill annealed	600°C, 24 h[1]
6(a)	1000°C, 15 m, wq	250°C, 1 day, wq
6(b)	1000°C, 15 m, wq	250°C, 7 days, wq

Notes: 1. Cooling method not specified.
2. Authors specified 304L but did not specify the heat.

(2) Specimen Preparation

Two different types of samples were used for these experiments. First, U-bend samples were made from the flat sheet materials (Code No. 1 and 2) in the solution annealed and the solution annealed and sensitized conditions. These samples were made by cutting strips 12.7 mm wide and 102 mm long from the 1.5 mm thick sheets and bending them to form a U which is loaded by a bolt. The composition of the bolt, the isolation of the sample from the bolt and the radius of curvature of the bend are not discussed in the report. The U-bend samples were held in place with alumina (Al_2O_3) spacers and rods. The surface treatment, if any, is not specified by the authors. The second type of samples used were tensile samples of square cross section. These samples were cut from the plate materials (Code No. 3-6) and the gage sections of these samples were 25.4 mm long and 6.4 mm wide. The gage sections of these samples were ground (with 220 grit wet/dry paper to remove 0.08 to 0.13 mm) and tested in the as-ground condition.

(3) Environment of the Test

Three different types of tests were used: the irradiation-corrosion test, the boildown test and the slow-strain-rate test. The environmental conditions used with each of these are described below.

Irradiation Corrosion Tests: For the irradiation-corrosion tests, U-bend specimens were placed into alloy 600 autoclaves at 50°C and 90°C and exposed to gamma irradiation (^{60}Co) at intensities of 5×10^5 and 3×10^5 rd/h (maximum), respectively. Each autoclave was divided into three zones and duplicate 304 and 304L samples (in the solution-annealed and the solution-annealed-and-sensitized conditions) were mounted horizontally on alumina rods in each zone. The lower zone environment was crushed tuff rock (Topopah Spring tuff rock) and J-13 water. The middle zone environment was crushed tuff rock and air/vapor while the upper zone environment was air/vapor only. The autoclaves were sparged with ≈240 ml of air daily to insure the presence of atmospheric gases. The autoclaves were periodically opened and the rock and water replaced after examination of the samples (the rock was not replaced after the 10 months examination). For the 50°C test, these examinations were conducted at 3, 5, 7, 10, 16, and 24 month, and at the test conclusion (25 months). For the 90°C test, these examinations were conducted at 3, 5, 7, 10, and 14 months, and at the conclusion of the test (23 months).

Boildown Tests: For the boildown tests, U-bend specimens were embedded in water-saturated tuff rock fragments and the concentrations of the various dissolved species were increased by periodically boiling off the water present in the autoclave. All of the samples used in these tests were sensitized (heat treatment codes 1 and 2(a-c) above). The autoclave was normally operated at 200°C and 1000 psig, and once every 7 days the back pressure was reduced and the water allowed to boil dry. After 24 h of operation in the "dry" condition, the autoclave was refilled with fresh air-sparged J-13 water. The autoclaves were opened and the samples examined at 3 and 6 months, and testing was halted after 12 months and 50 boildowns.

Slow-Strain Rate Tests: For the slow-strain rate tests, square-cross-section tensile samples were pulled at slow strain rates (1×10^{-4} and 2×10^{-7} m/m-s) in an autoclave environment containing crushed tuff rock through which J-13 well water passed at a slow rate (≈35ml/h) at 95°C and 150°C. The authors refer to an earlier report for a more complete description of this system (Westerman et al. 1982).

METHODS OF DATA COLLECTION AND ANALYSIS

For the irradiation corrosion tests, the autoclaves were periodically opened and the samples examined. Any "obvious" evidence of cracking was considered to be failure and an indication of susceptibility to stress corrosion cracking (SCC). After examining the samples, water samples were taken and analyzed for pH, conductivity and ionic content. Failed samples were examined metallographically. One unfailed specimen from each of the three environmental regions of each of the autoclaves was examined by X-ray photoelectron spectroscopy (XPS) to determine the surface chemistry of this alloy in these environments.

For the boildown tests, the autoclaves were periodically opened and the samples examined. Any "obvious" evidence of cracking was considered to be failure and an indication of susceptibility to SCC. Failed samples were examined metallographically. Water samples were taken under both hot and cold conditions and analyzed for ionic content.

For the slow-strain-rate tests, load and strain were continuously monitored during the test and the fracture surfaces were examined optically, metallographically and in the scanning electron microscope. The reduction in area at fracture, the elongation to fracture, the ultimate tensile strength, the yield strength and the failure mode (ductile, intergranular etc.) were determined for each test. The authors used three indicators to evaluate the relative susceptibility to SCC. These are: (1) a reduction in the ductility as compared to an inert environment, (2) a diminution of ductility at a particular strain rate as compared to other strain rates and (3) fractographic evidence of a brittle fracture.

To correlate microstructural changes with SCC susceptibility, metallurgical examinations and electrochemical experiments were performed on samples after various sensitizing heat treatments. The metallurgical examinations consisted of standard metallography and transmission electron microscopy (TEM). The electrochemical examinations consisted of electrochemical potentiokinetic reactivation (EPR) measurements. For this technique, the total charge per unit surface area required to repassivate the surface of a sample was measured. Since a sensitized structure will require more charge per unit surface area to repassivate, the magnitude of this measurement is a measure of the sensitization. The exact technique used for these measurements is not discussed in this report.

AMOUNT OF DATA

There are 27 figures and 9 tables.

UNCERTAINTIES IN DATA

The irradiation corrosion experiments at 90°C had a number of operational problems. The autoclave was removed four times from the irradiation access tube and dismantled because of blocking of the air refresh line, and each time the autoclave was opened it was dry due to local overheating and boiling. Repositioning of the control thermocouple corrected this problem, and the tests operated as intended thereafter. This moderate overheating should not alter the metallurgical conditions of the samples so the tests were resumed.

The water analyses showed a great deal of scatter and some relatively high ionic strengths. The solutions showing high ionic strengths of key ingredients were reanalyzed with the results confirming the earlier analyses. This effect was attributed to the "fresh" rock which was added after each examination and so the procedure was altered so that the rock was not replaced when the autoclave was opened for the 10-month and later examinations.

The purpose of the XPS experiments was to evaluate the possibility of surface contamination of the U-bend samples located in the rock/vapor and the vapor only regions of the autoclaves (especially for the 90°C tests). That is, the authors were concerned that wicking of water with aggressive ions or transport of water droplets during boiling and/or air sparging could be responsible for the high failure rate in the "dry" regions.

In the sixth month of the boildown experiment, the autoclave boiled dry and the temperature rose to 290°C for a 41 h period.

During the sensitization studies, annealing a 304L sample at 600°C for 10 h resulted in a sensitized microstructure while annealing a sample of the same alloy at the same temperature for 24 h resulted in an unsensitized structure. TEM examination showed that a continuous layer of grain boundary precipitate was present in the 10 h sample while no grain boundary precipitates were observed in the sample annealed for 24 h. The authors attributed this result to the fact that the samples used for this study were from different heats (code no. 3 and 4) and that some irregularity in the processing of the steel must be responsible for this difference.

DEFICIENCIES/LIMITATIONS IN DATABASE

The causes and the mechanisms of stress corrosion cracking in the vapor regions of the irradiation corrosion autoclave are unknown. It is likely that the chloride ion caused cracking and that the gamma flux accelerated this process as no SCC of 304L was observed in the absence of gamma irradiation.

The authors concluded that the test conditions differed from the repository conditions because (1) the temperature was not well controlled at the beginning of the 90°C test and boildown occurred, (2) the rock used was surface outcropping containing soluble salts, (3) the method of air sparging might have caused transfer and concentration of chlorides on the specimen surfaces, (4) the gamma irradiation levels were higher than expected in the repository, and (5) the stresses were quite high.

KEYWORDS

Experimental data, corrosion, electrochemical, irradiation-corrosion test, slow-strain-rate test (SSR), spectroscopy, visual examination, U-bend, air plus water vapor, J-13 water, tuff composition, Cl, tuff, cobalt 60, gamma radiation field, ambient pressure, dynamic (flow rate given), high pressure, high temperature, stainless steel, 304 stainless steel, 304L stainless steel, 316L stainless steel, sensitized, solution-treated, slow strain rate, J-13 steam, chloride (low ionic content), corrosion (stress cracking) SCC, cracking, cracking (environmentally assisted).

CONCLUSIONS

(1) "It was found that solution treated 304L can exhibit transgranular stress corrosion cracking."

(2) "It is likely that the cracking was chloride induced and accelerated by additional oxidizing power resulting from the gamma irradiation."

(3) "Most of the failures observed in the gamma flux test occurred in the vapor phase region of the 90°C autoclave" and "test conditions differ from the anticipated repository conditions..."

(4) "Type 304 stainless steel is more susceptible to stress corrosion cracking than 304L..."

(5) "When gamma flux was not present, the sensitized 304 exhibited intergranular stress corrosion cracking while the sensitized 304L did not."

(6) "The 304 stainless steel was found to be susceptible to intergranular SCC in SSR tests performed in 150°C J-13 well water after sensitization at 600°C for 24 h."

(7) "The susceptibility of 304 SS to SCC in SSR tests was correlated with the formation of grain boundary precipitates. Cracking was intergranular in all cases."

(8) "Neither 304L nor 316L stainless steel was found to be susceptible to SCC in SSR tests using J-13 well water test environment."

GENERAL COMMENTS OF REVIEWER

(1) Irradiation Corrosion Experiments

General (uniform) corrosion damage was greatest in the rock/vapor and vapor only regions of the autoclave. Specimens in the rock/water region of the autoclave showed no obvious evidence of attack. No evidence of pitting was found on any of the samples. Fifteen of the 48 U-bend samples in these tests failed as a result of stress corrosion cracking. In the 50°C test, five of the six solution annealed and sensitized 304 samples failed by intergranular cracking while no other failures occurred. These failures occurred in each of the three regions of the autoclave. In the 90°C tests, 10 of the 24 samples failed by stress corrosion cracking: 6 in the vapor-only region, 2 in the rock/vapor region and 2 in the rock/water region. Both of the sensitized 304 samples failed in the rock/water region while one each of the sensitized 304 and 304L samples failed in the rock/vapor region. In the vapor only region, two sensitized 304 samples failed by intergranular SCC while one of the sensitized 304L samples failed by transgranular SCC and one failed by mixed intergranular and transgranular SCC. Also, in the vapor only region one solution annealed 304 sample and one solution annealed 304L sample failed by transgranular SCC.

It is important to note that these results indicate that neither 304 nor 304L is completely resistant to SCC in the irradiated tuff/groundwater environment of this test. That is, failed samples were produced for each alloy in each heat treatment studied. In fact, one of the two candidate container alloy (304L) samples in its most corrosion resistant condition (solution annealed) failed by transgranular SCC in the vapor only region of this test. While the authors point out five differences between the irradiation corrosion test environment and the expected repository conditions, they do not point out that the environmental conditions of this experiment are closer to the expected environmental conditions of the repository than those of any other experiment reported to date. This reviewer does not feel that the five factors given by these authors are unrepresentative of the repository conditions for the following reasons:

(1) Temperature: The authors reported that the temperature in the experiment was not properly maintained and boildown resulted. The temperature of a waste container in a repository will not be constant and it will vary from point to point on the container. As a result, local boiling may occur and this condition is not unrepresentative of a possible repository condition.

(2) Rock: The authors report that use of fresh outcropping rock resulted in higher concentrations of soluble species than representative of repository conditions. This reviewer does not agree with this point because while the container temperature is above the boiling point, soluble species will concentrate in the rock around the container. Then, when the container temperature drops to the boiling point, water will migrate through these regions redissolving these species. As a result, the first groundwater that reaches the container will have anomalously high concentrations of soluble species. Therefore, the experimental conditions of this experiment are not unrepresentative of a possible repository condition.

(3) Air-sparging: The method of air sparging might have caused transfer and concentration of chlorides on the specimen surfaces. However, wetting of the container in the repository will not always be uniform and permeation of air thorough the rock to the container should maintain aerated conditions. Therefore, this is not unrepresentative of a possible repository condition.

(4) Irradiation Levels: The irradiation levels were far above those expected for the repository but this was intended to be a severe short-term test. While this is a severe test, it is logical for a conservatively "safe" experiment to use higher than expected radiation levels.

(5) Stress: The applied stresses were well above those expected for the repository. However, this experiment did not evaluate the minimum stress required for SCC, and the extensive secondary cracking or branching shown in the micrographs (e.g. fig. 5.3, 5.5 and 5.6) indicates that mass transport and not stress was limiting propagation. In fact, it appears from the figures that relatively low stresses may be sufficient for crack propagation.

(2) Boildown Tests

All ten of the 304 U-bend samples used in this test failed while none of the 304L samples failed.

(3) Slow-Strain-Rate Experiments

The sensitized 304 samples exhibited intergranular SCC while no SCC was observed for the other alloys and heat treatments.

(4) Sensitization Studies

The sensitization study is rather preliminary. The TEM diffraction conditions are not given for the micrographs and the reviewer cannot evaluate the micrographs from xerographic copies. The authors attribute the observed difference in the sensitization behavior of the 304 to precipitate morphology and for the 304L to chemical differences in the samples. Further studies will be required to verify these assumptions.

RELATED HLW REPORTS

Westerman, R. E., Pitman, S. G., and Nelson, J. L., "General Corrosion, Irradiation-Corrosion, and Environmental-Mechanical Evaluation of Nuclear Waste Package Structural Barrier Materials," PNL-4364, Pacific Northwest Laboratory, Richland, Washington, 1982.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

This report relates to NNWSI ISTP issue 2.2.4, the potential failure modes for the waste package container form dissolution mechanisms.

(b) New Licensing Issues

(c) General Comments

DRAFT

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

DATA SOURCE

(a) Organization Producing Report

Lawrence Livermore National Laboratory, Livermore, CA.

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

Van Konynenburg, R. A., Smith, C. F., Culham, H. W., and Smith, H. D., "Carbon-14 in Waste Packages for Spent Fuel in a Tuff Repository," UCRL-94708, October 1986.

DATE REVIEWED: 7/28/88

PURPOSE

To review available data in order to assess the behavior of the ^{14}C isotope in a tuff repository environment and whether possible ^{14}C release will be within limits set by the EPA and NRC.

CONTENTS

There is one table in the twelve-page report. Two of these pages list 56 references. The report reviews the following topics in about one page each: (1) the occurrence of ^{14}C naturally, in reactor releases, and health effects, (2) production of ^{14}C in light-water reactors, (3)physical distribution and chemical forms of ^{14}C in spent fuel, (4) the release of ^{14}C from spent fuel in aqueous solutions, in air, in nitrogen, and helium, (5) projected release behavior of ^{14}C from spent fuel waste packages in a tuff repository.

TYPE OF DATA

(1) Scope of the Report

The report is a review and discussion of measurements and calculations of the ^{14}C inventory in spent fuel, the physical distribution and chemical forms of the ^{14}C, and a description of projected ^{14}C behavior in the unsaturated conditions of the tuff repository.

(2) Failure Mode or Phenomenon Studied

The release of the ^{14}C isotope from spent fuel into moist air.

MATERIALS/COMPONENTS

^{14}C isotope; tuff; Zircaloy-clad fuel.

TEST CONDITIONS

(1) State of the Material being Tested

Details of the testing procedures for which the data are summarized are not given.

(2) Specimen Preparation

Details of the testing procedures for which the data are summarized are not given.

(3) Environment of the Material being Tested

The review summarizes data for ^{14}C releases from spent fuel in aqueous solutions, and in air, nitrogen, and helium gases.

METHODS OF DATA COLLECTION/ANALYSIS

Details of the testing procedures for which the data are summarized are not given.

AMOUNT OF DATA

There is one table.

Table 1: " Calculated and Measured Carbon-14 in U. S. Commercial Spent Fuel."

UNCERTAINTIES IN DATA

Not applicable.

DEFICIENCIES/LIMITATIONS IN DATABASE

The deficiencies in the data reviewed are given by the authors in their "Summary and Conclusions" section which is summarized under CONCLUSIONS below.

KEY WORDS

Literature review, air, nitrogen, helium, J-13 water, deionized, zircaloy, spent fuel (PWR), ^{14}C, ^{14}C release.

CONCLUSIONS

The authors give the following set of conclusions drawn from the review of available data:

1. Published measurements of ^{14}C in U.S. spent fuel are inadequate and deal mostly with Zircaloy-clad fuel (Westinghouse PWR fuel).

2. In Zircaloy-clad fuel, the uranium oxide contains more ^{14}C than the clad, and the fuel-rod gas contains only a small amount, probably as CO or CH_4. The chemical form of the isotope in the fuel is not known. The carbon probably exists as interstitial carbon or zirconium carbide in the cladding.

3. A negligible amount of ^{14}C is released from heated, intact spent fuel in N_2 or He.

4. In heating an intact PWR fuel assembly, ^{14}C on the surface was oxidized to CO_2 and released. The isotope may have originated with N ^{i}n the cladding, or been adsorbed from the reactor cooling water.

5. In less than one year, $>10^{-5}$ of the ^{14}C inventory can be release from spent fuel.

6. ^{14}C released by pressurized gas escaping when fuel-rod cladding ruptures, may be about 10^{-4} of the calculated total rod inventory. Cladding lifetimes will vary in time.

7. Determination of satisfactory ^{14}C release (the 10CFR60 limit) in an unsaturated tuff repository will depend on the results of measurements of oxidation and release from the spent-fuel rod surface and interior in moist air at temperatures above and below the boiling point of water in the repository.

8. Detailed transport modeling is needed to determine whether the release limit (according to 40CFR191) for ^{14}C release from an unsaturated tuff repository. The modeling must include factors for dilution of the naturally-occurring carbon which is less abundant relative to the earth's atmosphere.

9. Measurements should be made of the ^{14}C content in (a) the structural components of Westinghouse Zircaloy-clad PWR fuel which do not include Zircaloy, (b) all fuel components from other sources, especially BWR fuel, and (c) fuel clad with stainless steel.

The authors state, "Our current estimates of the ^{14}C inventory are based largely on calculations, which need to be tested against measurements.

GENERAL COMMENTS OF REVIEWER

This report apparently covers the data known at the time of its publication. The authors comment about the need for experimental data, in order to test their calculations, is important. As many specimens of spent fuel as possible should be analyzed to provide accurate inventory estimates in order to understand and estimate the potential for release of ^{14}C from a repository.

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, when, how, and at what rate will radionuclides be released from the waste form.

(b) New Licensing Issues

(c) General Comments

CURRENT OVHD RATES
CTR 35.50%
LAB 09.40%
BUR 45.70%

PREPARED FOR: DIVISION 450

NATIONAL BUREAU OF STANDARDS
DIVISION ONE-LINE
DOLLAR AMOUNTS ARE CURRENT YEAR OBLIGATIONS
PAY PERIOD 13 ENDING 07/02/88
PAY PERIOD 14 ENDING 07/16/88
PAY PERIOD 15 ENDING 07/30/88

	(---------C U R R E N T	P E R I O D	--------)				(---------------F I S C A L	Y E A R	T O	D A T E	--------------)			
PPD	COST CNTR	PROG TASK	LABOR HOURS	* LABOR+ APPLS $	OTHER OBJ $	TOTAL OBLG $	LABOR HOURS	LABOR + LV & BN $	TOTAL OVHDS $	OTHER OBJ $	TOTAL OBLG $	AUTHOR $	CURRENT BALANCE $	Percentage of FY87 Funds Obligated to Date
13	0480	14200	323 ~~21~~	18452 ~~-6728~~	196	18648 ~~-6532~~	8734	232017	214664	24340	471021	659975	188954	71.4
14	0480	14200	319	17918	1087	19005	9053	241418	223181	25427	490026	668675	178649	72.9
15	0480	14200	468	29068	13343	42411	9521	256741	236926	38770	532437	668675	136238	79.4

The labor hours and their costs shown as 21 h and -$6728 represent a correction made in PPD13, due to an earlier error in bookkeeping, actual hours and cost during PPD13 are 323 and $18,452. The balance of the information is correct as shown. The authorized amount is shown to increase by $8,700. This is due to cancellation of an $8,700 contract.

SDI006, UD 8816, SER. DD016

File(s) searched:

File 6:NTIS - 64-88/ISS16
(COPR. 1988 NTIS)

Sets selected:

Set	Items	Description
1	2	WASTE(W)PACKAGE?
2	2	CANISTER?
3	43	CORROSION
4	19	LEACHING
5	54	GLASS
6	9	VITRIFICATION
7	111	S3-S6/OR
8	6	HIGH(W)LEVEL(W)WASTE?
9	109	RADIOACTIVE(W)WASTE?
10	15	NUCLEAR(W)WASTE?
11	2	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

Prints requested ('*' indicates user print cancellation) :

Date	Time	Description
25jul	04:41EST	PR 11/5/1-25 (items 1-2)

Total items to be printed: 2

SDI103, UD 8812, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-88/JUNE(ISS12)

Sets selected:

Set	Items	Description
1	16	WASTE(W)PACKAGE?
2	5	CANISTER?
3	120	CORROSION (1974 DEC)
4	32	LEACHING (1974 DEC)
5	105	GLASS (1974 DEC)
6	13	VITRIFICATION (1974 DEC)
7	237	S3-S6/OR
8	15	HIGH(W)LEVEL(W)WASTE?
9	295	RADIOACTIVE(W)WASTE?
10	57	NUCLEAR(W)WASTE?
11	4	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

30

Prints requested ('*' indicates user print cancellation) :

Date	Time	Description
07jul	07:04EST	PR 11/5/1-25 (items 1-4)

Total items to be printed: 4

SDI103, UD 8813, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-88/JULY(ISS13)

Sets selected:

Set	Items	Description
1	13	WASTE(W)PACKAGE?
2	14	CANISTER?
3	320	CORROSION (1974 DEC)
4	61	LEACHING (1974 DEC)
5	169	GLASS (1974 DEC)
6	19	VITRIFICATION (1974 DEC)
7	540	S3-S6/OR
8	41	HIGH(W)LEVEL(W)WASTE?
9	342	RADIOACTIVE(W)WASTE?
10	62	NUCLEAR(W)WASTE?
11	10	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

Prints requested ('*' indicates user print cancellation) :

Date	Time	Description
25jul	05:46EST	PR 11/5/1-25 (items 1-10)

Total items to be printed: 10

SDI103, UD 8814, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-88/JULY(ISS14)

Sets selected:

Set	Items	Description
1	7	WASTE(W)PACKAGE?
2	7	CANISTER?
3	177	CORROSION (1974 DEC)
4	52	LEACHING (1974 DEC)
5	97	GLASS (1974 DEC)
6	17	VITRIFICATION (1974 DEC)
7	319	S3-S6/OR
8	19	HIGH(W)LEVEL(W)WASTE?
9	259	RADIOACTIVE(W)WASTE?
10	50	NUCLEAR(W)WASTE?
11	6	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

32

Prints requested ('*' indicates user print cancellation) :

Date	Time	Description
30jul	03:21EST	PR 11/5/1-25 (items 1-6)

Total items to be printed: 6

SDI008, UD 8807, SER. DD001

File(s) searched:

File 8:COMPENDEX PLUS - 70-88/JULY COPR. ENGINEERING INFO INC. 1988)

Sets selected:

Set	Items	Description
1	3	WASTE()PACKAGE?
2	7	CANISTER?
3	285	CORROSION
4	52	LEACHING
5	283	GLASS
6	2	VITRIFICATION
7	611	CORROSION OR LEACHING OR GLASS OR VITRIFICATION
8	1505	METAL?
9	770	STEEL?
10	1	ZIRCALOY
11	134	TITANIUM
12	234	COPPER
13	2105	METAL? OR STEEL? OR ZIRCALOY OR TITANIUM OR COPPER
14	1	ZIRCALOY?
15	2105	S13-S14/OR
16	14	HIGH()LEVEL()WASTE?
17	189	RADIOACTIVE()WASTE?
18	20	NUCLEAR()WASTE?
19	0	(1OR2) AND (S7 AND S15) AND (S16 OR S17 OR S18)
20	1	(S1 OR S2) AND (S7 AND S15)
21	1	S20 AND (S16 OR S17 OR S18)
22	1	S21/1984-1988
23	1	S22/1986-1988
24	0	ANNA FRAKER BLDG. 223 RM. B244 X6009

Prints requested ('*' indicates user print cancellation) :

Date	Time	Description
28jul	08:15EST	PR S21/5/ALL (items 1-1)

Total items to be printed: 1

SDI032, UD 8808, SER. DD003

File(s) searched:

File 32:METADEX 66-88/AUG
(Copr. 1988 ASM International)

Sets selected:

Set	Items	Description
1	0	WASTE()PACKAGE?
2	0	CANISTER?
3	441	CORROSION
4	20	LEACHING
5	49	GLASS
6	1	VITRIFICATION
7	501	CORROSION OR LEACHING OR GLASS OR VITRIFICATION
8	1476	METAL?
9	1504	STEEL?
10	3	ZIRCALOY
11	194	TITANIUM
12	299	COPPER
13	2619	METAL? OR STEEL? OR ZIRCALOY OR TITANIUM OR COPPER
14	3	ZIRCALOY?
15	2619	S13-S14/OR
16	0	HIGH()LEVEL()WASTE?
17	3	RADIOACTIVE()WASTE?
18	2	NUCLEAR()WASTE?
19	0	(1OR2) AND (S7 AND S15) AND (S16 OR S17 OR S18)
20	0	(S1 OR S2) AND (S7 AND S15)
21	0	S20 AND (S16 OR S17 OR S18)
22	0	S21/1984-1988
23	0	S22/1986-1988
24	0	ANNA FRAKER BLDG. 223 RM. B244 X6009

Prints requested ('*' indicates user print cancellation) :

Date	Time	Description
25jul	20:03EST	PR 21/5/ALL (no items to PRINT)

Total items to be printed: 0

Trip Report - Meetings of Committee C26 on Nuclear Fuel Cycle

July 25, 1988

Attendees

Bates, Barkatt, Interrante, Jantzen, Macedo, Peterson, and Chairman Thornton.

A pre-meeting of the Task Group or Accelerated Testing, C26.13, was begun today at about 10:30 am. At this meeting, the results of a June 21, 1988 (pre-meeting) meeting at Catholic University were discussed. C. Peterson's notes on the 6-21-88 meeting were used to lead the discussion. Problems associated with development of mechanistic understanding of and models for behaviors of components of the engineered barrier were discussed. In particular, glass and spent fuel were discussed and metal barriers and backfill were mentioned. The "morning" session extended to 1:00 pm and was devoted to refinement of a Logic Diagram for Extrapolation of Short-Term Data.

The suitability of MCC-1, as proposed in the WAPS, for prediction of repository behavior was brought into question. The MCC-1 test, which is a static leach test, was originally thought of as a screening test for leachability of various proposed glass compositions.

July 26, 1988

Subcommittee C26.07 - Waste Materials

Chairman Dick Blauvelt opened the meeting explaining that the development of a guide for accelerated testing would be the topic of this meeting, and that this activity would be continued at the C26.13 meeting this afternoon. Then Tom Thornton took charge of the meeting.

Agenda --

Definitions -- seek preliminary agreement on how they should read.

Logic diagram -- develop it for use as an insert for Fig. 2.

Review the middle part (Sect. 12-19) of the document.

Lay plans to bring the document to C26.07 and C26.13 SC ballots.

In the morning session, definitions were discussed for various terms, including degradation mode, degradation mechanism, and accelerated test.

July 26, 1988 -- Subcommittee C26.13 on Mined Geologic Repository Waste Package Materials Testing.

This meeting was a continuation of the morning meeting and it was chaired by T. Thornton. The focus was development of the logic diagram. Plans were made for distribution of the accelerated test with the revised diagrams definitions (new and revised) to all members of this subcommittee.

The next meeting was set for September 20 and 21 (Tue. and Wed.) at Catholic University. Mr. Peterson will be the convener, working through ASTM Headquarters for the mailing of meeting notices if needed.

July 27, 1988 -- Subcommittee C26.07 - Waste Materials

The Draft Test Method for Splitting Tensile Strength for Brittle Nuclear Waste Forms was reviewed and prepared for concurrent ballot at the Committee (C26) and Subcommittee (C26.07) levels.

The MCC-1 static leach test was discussed. Peterson briefly discussed some concerns expressed by the NRC commentors. Subcommittee ballot will likely be made by the next meeting of C26. Changes would be made to deal with negative votes submitted by T. Johnson (NRC) on low-level waste and by Breckner on statistical considerations. Leaching is done in this test using brine, and it was noted that it would be a major effort to eliminate the specificity to brine, so as to attempt to make the test more broadly useful. In the U. S., only the WIPP site would have and interest in this test method.

Suggested modifications to the leach test ANS 16.1 were discussed. The notion is that with suitable modifications this test could be made to satisfy requirements of both NRC and EPA.

The next meeting of this committee will be made in New Orleans, Jan. 1988, at the Hilton Hotel.

There are two ISO Standards that ANSI would have ASTM subcommittees consider, so as to develop an U.S. position for the U.S.TAG for ISO/TC 85: "Measurement of Steady State Concentration of Radionuclides Released from Simulated Geologic Repository Conditions," and "Calibration of Leach Testing Devices for Solidified Waste." Contact chairman Blauvelt if you have interest in either of these activities.

July 27, 1988

D34.1 on Waste Minimization (Chairman John Frick) is a new subcommittee dealing with hazardous waste, whether low- or high-level. Mr. Frick is located at the Defense Logistics Station, Cameron Station, Alexandria, VA.

A draft for review of a document titled "Needs in Low-Level Radioactive Waste Standards" was reviewed at this meeting. The task group D34.07.01 was formed as a technical advising group with responsibility for development of a list of problems and standards needs in LLW management.

July 28, 1988 (Thursday morning) -- C26.01 - Editorial subcommittee

At the C26 meetings during ASTM Committee week in Albequerque, C. Interrante was designated as the person responsible for development of the terminology for the document on accelerated testing. In keeping with discharge of that responsibility this terminology was taken to this meering of C26.01.

After reviewing various items that had been generated by other subcommittees for inclusion in Terminology C857, the work of C26.07 and C26.13 on terminology for prediction of long-term performance of waste-package materials for disposal of high-level nuclear waste was discussed in detail. Numerous suggestions were made by those present at this meeting. These will be incorporated in the version of this terminology to be sent to Chairman Thornton for inclusion in the modified document that he is preparing for subcommittee ballot. The copy of this terminology that is attached to this trip report also incorporates modifications made after this meeting with C26.01, and it is the version that Chairman Thornton will receive.

July 28, 1988 (Thursday afternoon) -- C26 - Main Committee on Nuclear Fuel Cycle

Within the subcommittee reports, Dick Blauvelt (Chairman of SC C26.07) and Charles Interrante (in the absence of C26.13 Chairman Thornton) presented a status report on C26.07 and C26.13, the development by a guide for prediction of long-term performance of waste package materials for disposal of high-level nuclear waste. Our intention to meet in September was announced and the change in the focus of the method (away from a practice for accelerated testing and towards a guide for prediction of performance) was stressed, with emphasis on an anticipated change in the title and an expectation that the guide will outline necessary steps that must be addressed in the predictive process but it will not focus on accelerated testing and its interpretation.

The next meeting of C26 will be held in Orlando, Jan. 22-26 at Walt Disney World where hotel rooms will be about $95. After the close of this meeting, a former chairman of the ASTM Committee on Terminology, N. Trahey, reviewed the terminology for predictive testing and offered some suggestions, on various delimiters for use with these definitions.

DRAFT

ASTM Subcommittee C26.13 on Mined Geologic Repository Waste Package Materials Testing

DRAFT Terminology - Guide for Prediction of Long-Term Performance of Waste Package Materials for Disposal of High-Level Nuclear Waste

4. Definitions

4.1 accelerated test -- a short-term test that seeks to simulate long-term behavior

Discussion -- In this method, consideration must be given as to whether or not the mechanism of degradation for the accelerated test and the long-term behavior are the same. Examples of parameters that might be varied to accelerate processes might include temperature, electrical potential, mechanical stress, radiation level, and surface to volume ratio.

4.2 bounding condition -- in this practice, a value that might exceed those credibly expected for a variable affecting degradation of a component

4.3 characterization test -- in high-level radioactive waste, any test conducted principally to furnish a mechanistic understanding of degradation. Examples include polarization tests, potential-pH (Pourbaix) diagrams, solubility analyses, XRD of corrosion layers, etc. A characterization test might be conducted under conditions that exceed those expected for a repository setting

4.4 disposal -- in nuclear waste storage, the isolation of radioactive waste from the accessible environment and the public

4.5 degradation -- in nuclear waste storage, deleterious performance related change in the physical or chemical properties of a material in a waste package, such as glass, weldments, base metal, and even the waste form itself

4.5.1 degradation mode -- in nuclear waste storage, a particular form of degradation <general corrosion, pitting, crevice corrosion, stress rupture, dissolution, devitrification, selective leaching, etc.>

4.5.2 degradation mechanism -- in nuclear waste storage, the fundamental chemical or physical processes by which degradation occurs <the steps involved in galvanic corrosion of a container material>

4.6 engineered barrier system -- in high-level waste storage, the waste packages and the underground facilities [10 CFR 60.2] used to increase the time lapsed prior to release of radionuclides to the accessible environment

4.7 geologic repository -- a system for the disposal of radioactive waste in (excavated) geologic media

4.8 high-level radioactive waste -- 1: irradiated reactor fuel 2: liquid wastes from reprocessing of irradiated reactor fuel 3: solid waste formed from liquid high-level radioactive waste <glass>

4.9 mechanism -- the fundamental physical or chemical processes involved in or responsible for an action, reaction, or other natural phenomenon [Webster]

4.10 model -- a mathematical representation of a process

4.10.1 mechanistic model -- a model based on one or more mechanisms

4.10.2 empirical model -- a model based directly on observations or data from experiments, without regard to mechanism or theory

4.10.3 bounding model -- a model that uses extreme values for parameters that affect a process

4.11 performance confirmation test -- a test designed to give new assurance of the validity of expected performance, such as an in-situ test to validate model predictions

4.12 service-condition test -- a test for evaluation of behavior under conditions expected in service

4.13 short-term test -- in high-level radioactive waste management, a test conducted over a period less than five years, such as a characterization test, accelerated test, and a service condition test

4.14 waste form -- in nuclear waste storage, radioactive waste materials and any encapsulating or stabilizing matrix [10 CFR 60.2]

<u>revised Aug. 10, 1988</u>