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WM-16 (2)

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

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July 15, 1987

'87 JUL 16 P4:48

Mr. Everett A. Wick
Division of Waste Management
Office of Nuclear Materials Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Monthly Letter Status Report for June 1987 (FIN-A-4171-6)

Dear Mr. Wick:

Enclosed is the June 1987 monthly progress report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-6). In response to your request, the first section, Task 1, has been expanded to include both a brief overview of the repository project reports under NBS review and a section titled "Waste Form Degradation". The section on Waste Form Degradation includes a brief technical summary of the reports under review in the areas of glass, Zircaloy, and spent fuel. The financial information is reported separately.

Sincerely,

Charles G. Interrante

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

Enclosures

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Monthly Letter Report for June 1987

Published July 1987

(FIN-A-4171-6)

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

Sponsor: Nuclear Regulatory Commission (NRC)
Office of Nuclear Materials Safety and Safeguards
Silver Spring, MD 20910

Task 1 -- Review of Waste Package Data Base

STATUS OF REVIEWS

Appended to this report are the following three Draft Reviews not previously submitted. Comments by NRC and its contractors are solicited.

1. UCRL-15825, "The Effect of Gamma Radiation on Groundwater Chemistry and Glass Leaching as Related to the NNWSI Repository Site," May 1986

2. ANL-85-41, "One-Year Results of the NNWSI Unsaturated Test Procedure: SRL 165 Glass Application," August 1984

3. UCRL-53719, "Radiation Chemical Effects in Experiments to Study the Reaction of Glass in an Environment of Gamma-Irradiated Groundwater and Tuff," May 1986

BWIP -- BASALT WASTE ISOLATION PROJECT

The major deficiencies in the available data pertaining to a waste repository in basalt fall into three areas: (1) packing material breakdown (2) groundwater chemistry (Are there complexing agents which can increase the solubility of metals?) and (3) corrosion of the metallic waste container. Leaching from the glass waste form in a basalt environment is also an area currently under NBS review.

Three reports concerning the testing of AISI 1020 steel are under review. The first by R. P. Anantatmula, "Effect of Grande Ronde Basalt Groundwater Composition and Temperature on the Corrosion of Low-Carbon Steel in the Presence of Basalt-Bentonite Packing" [RHO-BW-SA-391P, 1985], deals with the effect of Grande Ronde Basalt groundwater composition and temperature on the corrosion rate of AISI 1020 steel in basalt-bentonite packing. The tests reported were conducted as part of the BWIP Containment Materials Testing Program. A second report, "Electromechanical Testing of AISI 1020 Steel in Hanford Grande Ronde Groundwater" [1985], by S. G. Pitman is also under review. During this month, review has been initiated on the third report "Environmental Testing of AISI 1020 Steel in Hanford Grande Ronde Groundwater" [B023959, 1983].

Further testing to gather strength and ductility data is being conducted on ASTM A387 and A27 steel. Work in this area is detailed in "Slow-Strain-Rate Testing of 9% Cr, 1% Mo Wrought Steel and ASTM A27 Cast Steel in Hanford Grande Ronde Groundwater" [SD-BWI-TS-008, 1984]. Tests continue on low-alloy cast steels.

The corrosion behavior of several iron-base and titanium-base alloys was studied and reported in "Irradiation-Corrosion Evaluation of Metals for Nuclear Waste Package Applications in Grande Ronde Basalt Groundwater" [RHO-BW-SA-316P, 1983]. Since low-alloy cast steels have been found to corrode when immersed in Grande Ronde basalt groundwater, review of this type of report is necessary. In NUREG-4737, Vol. 2, we have indicated that further work needs to be performed on these materials to determine whether they are susceptible to environmentally-assisted cracking.

In basalt groundwater, pitting has been studied in two carbon steels, ASTM A27 and A36, as well as in two low-alloy cast steels tested. Review of "Short-Term Stress-Corrosion-Cracking Tests for A36 and A387-9 Steels in Simulated Hanford Groundwater" [SD-BWI-TS-012, 1985] will monitor the results of relatively short-term tests on precracked self-loaded fracture-mechanics specimens of ASTM A36 and A387-9 steels. Environmentally assisted cracking studies for BWIP are reported in "Status of Environmentally Assisted Cracking Studies by the Basalt Waste Isolation Project" [RHO-BW-SA-560P, 1986], which is also under review.

Thermal calculations on the basalt repository continue to be updated. One report, "Thermal Analysis of Waste Package Preliminary Reliability Assessment" [RHO-BW-SA-0509P, 1986], is under review in this area.

Review has been initiated on NUREG/CR-4309 [1986], "Valence Effects of Solubility and Sorption: The Solubility of Tc(IV) Oxides." Solubilities of Tc(IV) oxides will be useful for calculation of transport rates of technetium in the case of solubility-limited transport in the studies on the leach rate of borosilicate waste glass.

In the selection of a repository for burying nuclear waste, the leach rate of the vitrified waste in groundwater is of concern. Basaltic groundwaters have inherently low redox potentials, which may affect the waste form leach rate. Review has been initiated on two reports, "Methods of Simulating Low Redox

Potential (Eh) for a Basalt Repository," by C. M. Jantzen [1984], and "Control of Oxidation Potential for Basalt Repository Simulation tests," by C. M. Jantzen and G. G. Wicks [1985]. Both reports identify studies on the leach rate of borosilicate waste glass.

Review is continuing on several other reports. "Technical Progress Report on BWIP Canister Materials Crack Growth Study for FY 1983" [SD-BWI-TI-165, 1984] describes the canister crack-growth study at Westinghouse Hanford Company. Review is continuing on "REPREL Computer Code: Users Guide" [SD-BWI-TI-165, 1985]. The REPREL computer code provides a tool for estimating the radionuclide mass release from a single container which failed at specified time. "BWIP General Corrosion Studies, Summary Report of Activities in FY-1984" [1984] and "Sorption Behavior of Selected Radionuclides on Columbia River Basalts" [RHO-BW-LD-48, 1986] continue to be reviewed.

BWIP -- Review is continuing on the following reports.

1. RHO-BW-SA-391P, "Effect of Grande Ronde Basalt Groundwater Composition and Temperature on the Corrosion of Low-Carbon Steel in the Presence of Basalt-Packing," August 1985

2. RHO-BW-SA-316P, "Irradiation-Corrosion Evaluation of Metals for Nuclear Waste Package Applications in Grande Ronde Basalt Groundwater," November 1983

3. SD-BWI-TS-008, "Slow-Strain-Rate Testing of 9% Cr, 1% Mo Wrought Steel and ASTM A27 Cast Steel in Hanford Grande Ronde Groundwater," October 1984

4. RHO-BW-SA-509P, "Thermal Analysis of Waste Package Preliminary Reliability Assessment," March 1986

5. B047154, "BWIP General Corrosion Studies, Summary Report of Activities in FY-1984," October 1984

6. SD-BWI-TS-012, "Short-term Stress-Corrosion-Cracking Tests for A36 and A387-9 Steels in Simulated Hanford Groundwater," January 1985

7. SD-BWI-TI-165, "Technical Progress Report on BWIP Canister Materials Crack Growth Study for FY 1983," January 1984

8. RHO-BW-CR-148P, "REPREL Computer Code: User Guide," June 1985

9. RHO-BW-SA-560P, "Status of Environmentally Assisted Cracking Studies by the Basalt Waste Isolation Project," Symposium on Radioactive Waste Management '86, March 1986

10. "Electromechanical Testing of AISI 1020 Steel in Hanford Grande Ronde Groundwater," S. G. Pitman, July 1983

11. RHO-BW-LD-48, "Sorption Behavior of Selected Radionuclides on Columbia River Basalts," August 1986

BWIP -- Review has been initiated on the following reports.

1. "Methods of Simulating Low Redox Potential (Eh) for a Basalt Repository," Materials Research Society Proceedings, 1984

2. "Control of Oxidation Potential for Basalt Repository Simulation Tests," Scientific Basis for Nuclear Waste Management, 1985

3. NUREG/CR-4309, ORNL-6199, "Valence Effects on Solubility and Sorption: The Solubility of Tc(IV) Oxides," March 1986

4. B023959, "Environmental Testing of AISI 1020 Steel in Hanford Grande Ronde Groundwater," July 1983

NNWSI -- NEVADA NUCLEAR WASTE STORAGE INVESTIGATIONS

The effects of radiation, as an area of study, has been added to the six previously identified areas (groundwater, steel, stainless steel, Zircaloy, copper, and spent fuel) for the Nevada Nuclear Waste Storage Investigation (NNWSI). Reviews conducted by NBS on three papers that deal with the effects of gamma radiation on groundwater chemistry in relation to glass as a waste form for the proposed repository in tuff are appended to this report. In one of these three papers, "One-Year Results of the NNWSI Unsaturated Test Procedures: SRL 165 Glass Application" [ANL-85-41, 1985], a series of tests was conducted on the Savannah River Reference Glass, SRL 165. The purpose of these tests was to determine whether the NNWSI Unsaturated Test could be used to produce useful data in assessing waste form performance under the unsaturated conditions anticipated at NNWSI. The two other papers deal with the effect of gamma-irradiated groundwater and its relationship to the leaching of glass as a waste form. Five additional papers in the area of glass leaching in gamma-irradiated groundwater in a tuff environment are also in the NBS review process. The NBS reviews in this area parallel other NBS reviews listed in the section under Glass.

Papers continue to be reviewed in several of the previously identified areas, Zircaloy, copper, and spent fuel. One paper in the review process, "Feasibility Assessment of Copper-Base Waste Package Container Materials in a Tuff Repository," by C. F. Acton and R. D. McCright discusses corrosion testing in potentially irradiated environments [UCID-20847, 1986]. These tests received emphasis during the second year of a two-year study on the feasibility of using copper or a copper-base alloy as a container material for a waste package in a potential repository in tuff.

Stopping or retarding the release of radionuclides to the environment is the ultimate goal of the high-level waste package program. It is anticipated that 0.01 percent of the fuel rods placed in the tuff repository will have cladding defects in the form of splits or pinholes. When the primary containment vessel is breached, only the fuel rod cladding will remain as a barrier to moisture ingress and radionuclide egress. The solubilities and rates of release of the uranium oxide and other radioactive components of the spent fuel are examined in several reports under review. Spent-fuel-rod cladding, as a potential barrier for radionuclide release, is under active investigation at NNWSI. Three reports, "Test Plan for Long-Term Low-Temperature Oxidation of Spent Fuel Series I" [HEDL-7560, 1986], "Experimental Study of the Dissolution Spent Fuel at 85°C in Natural Groundwater" [UCRL-95633, 1986], and "Carbon-14 in Waste Packages for Spent Fuel in a Tuff Repository" [UCRL-94708, 1986] discuss technical issues important to radioactive release, and these reports continue in the NBS review cycle. Two other reports, "Spent Fuel as a Waste Form -- Data Needs to Allow Long Term Performance Assessment Under Repository Disposal Conditions" by V. M. Oversby [1968] and "LWR Spent Fuel Characteristics Relevant to Performance as Wasteform -- in a Potential Tuff Repository" [UCRL-92891, 1985] discuss the performance of spent fuel as a waste form.

Two other reports, "NNWSI Phase II Materials Interaction Test Procedures and Preliminary Results" [ANL-84-81, 1985] and "NNWSI Waste Form Test Method for Unsaturated Disposal Conditions" [UCRL-15723, 1985], are under review. Test methods have been developed to measure the release of radionuclides from the waste package under simulated repository conditions and results of these tests are reported. Reports such as these should provide information on materials interactions that may occur in the proposed repository in tuff.

NNWSI -- Review is continuing on the following reports.

1. UCRL-15723, "NNWSI Waste Form Test Method for Unsaturated Disposal Conditions," March 1985

2. UCID-20847, "Feasibility Assessment of Copper-Base Waste Package Container Materials in Tuff Repository, September 1986

3. HEDL-7560, "Test Plan for Long-Term Low-Temperature Oxidation of Spent Fuel Series 1", June 1986

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

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

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

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

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

9. DP-MS-85-141, "Leaching Fully Radioactive SRP Nuclear Waste Glass in Tuff Groundwater in Stainless Steel Vessels," May 1986

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

11. "Spent Fuel as a Waste Form - Data Needs to Allow Long Term Performance Assessment Under Repository Disposal Conditions," V. M. Oversby, 1968

12. UCRL-92891,"LWR Spent Fuel Characteristics Relevant to Performance as a Wasteform in a Potential Tuff Repository," June 1985

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

SRP -- SALT REPOSITORY PROJECT

There are thirteen reports in various stages of the review process for the Salt Repository Project (SRP). The most notable of them are two reports by Westerman and co-workers on the corrosion and fracture behavior of iron-base alloys in salt environments [PNL-3484, 1980 and PNL-SA-14029, 1986]. These reports contain a large quantity of information and data. Careful review will take some time but should be beneficial. Also under review are three reports by Levy and co-workers on radiation damage in salt and colloid formation [BNL-32001, 1984, BNL-29909, 1981, and Nuclear Technology, 1983]. The other reports under review concern the chemistry of the environment [UCRL-53726, 1986], materials testing [BMI/ONWI-583, 1986 and BMI/ONWI-490, 1983], heat transfer [BMI/ONWI-612, 1986], corrosion modeling [BMI/ONWI-592, 1986], and buckling criteria for the container [BMI/ONWI-597, 1986].

SRP -- Review is continuing on the following reports.

1. BMI/ONWI-592, "ERG Review of Salt Constitutive Law, Salt Stress Determinations and Salt Corrosion Modeling Studies," March 1986

2. BMI/ONWI-612, "The Effects of Stabilizers on the Heat Transfer Characteristics of a Nuclear Waste Canister," July 1986

3. BMI/ONWI-597, "Buckling Design Criteria for Waste Package Disposal Containers in Mined Salt Repositories," December 1986

4. DOE/CH-21, "Systems Engineering Management Plan for the Salt Repository Project," August 1986

5. UCRL-53726, "Reference Waste Package Environment Report," October 1986

6. "Radiation Damage Studies on Natural Rock Salt from Various Geological Localities of Interest to the Radioactive Waste Disposal Program," Nuclear Technology, 60, 231-243, February 1983

7. BMI/ONWI-626, "ERG Review of the SRP Salt Irradiation Effects Program," November 1986

8. BMI/ONWI-611, "ERG Review of Waste Package Container Materials Selection and Corrosion," July 1986

9. BMI/ONWI-583, "Waste Package Materials Testing for a Salt Repository: 1983 Status Summary Report," September 1986

SRP -- Review has been initiated on the following report.

1. BMI/ONWI-490, "Waste Package Materials Testing for a Salt Repository: 1982 Status Report," August 1983

2. PNL-3484, "Investigation of Metallic, Ceramic, and Polymeric Materials for Engineered Barrier Applications in Nuclear-Waste Packages," October 1980

3. BNL-29909, "Radiation Damage Studies on Synthetic NaCl Crystals and Natural Rock Salt for Radioactive Waste Disposal Applications," Technology of High-Level Nuclear Waste Disposal, Vol. 1, 1981

4. PNL-5426, "Corrosion and Environmental-Mechanical Characterization of Iron-Base Nuclear Waste Package Structural Barrier Materials Annual Report -- FY 1984," March 1986

5. PNL-SA-14029, "Corrosion of Iron-Base Waste Package Container Materials in Salt Environments," Nuclear Power Conference, Philadelphia, PA, July 20, 1986

WASTE FORM DEGRADATION

Zircaloy Cladding

The question regarding Zircaloy cladding is whether it is possible to allow any credit for the cladding as a barrier to radionuclide release. The Environmental Protection Agency (EPA), in 40 CFR Part 191, requires that no more than one part in 10^5 of a 1000 year inventory of radionuclides be released annually for 10,000 years. Zircaloy cladding with a thickness of approximately 1 mm is used as a tube to surround the UO_2 nuclear fuel pellets during the time when the fuel is in service in the reactor, in temporary storage after use, and in long-term nuclear waste storage.

Determinations of whether the cladding will act as a barrier to radionuclide release must consider the specific Zircaloy, its metallurgical condition, its corrosion resistance and its history in service and storage. Additional considerations include defects such as holes or slits in the Zircaloy tube. Some work has been done by the NNWSI to measure radionuclide release through holes or slits. In a report by C. N. Wilson and V. M. Oversby, "Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater" [UCRL-91464, 1985] show that intact cladding did control radionuclide release and that radionuclide release increased with increased exposed area of the spent fuel. A review was completed for an NNWSI report by A. J. Rothman entitled "Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Fuel in a Tuff Repository" [UCID-20172, 1984]. Rothman concluded that failure due to oxidation probably would not occur but that susceptibility to stress corrosion cracking was uncertain and needed further study. (A review of Zircaloy corrosion is in progress as part of one of the NBS experimental testing programs.)

The history of a specific cladding could be important as it relates to the exposure conditions in the nuclear reactor, the composition of crud which was present, and how well the crud was cleaned. Storage and handling of the spent-fuel cladding also are important. Zircaloy is a highly corrosion resistant material, but under certain environments and conditions it is subject to various forms of local corrosion including stress corrosion cracking, nodular corrosion and possibly pitting corrosion. It is not certain, at this time, that any credit can be given for radionuclide containment by the Zircaloy cladding.

Spent Fuel

Oxidation is a factor relevant to performance of spent fuel as a wasteform. Approximately 0.01 percent of spent fuel rods have cracks or fissures through which air may enter and lead to oxidation of the UO_2 spent fuel. This oxidation process could lead to stress on the Zircaloy cans because of the expansion of fuel which could cause additional cracks to develop. Einziger and Woodley, [HEDL-SA-3271P, 1985], [HEDL-7556, 1986], and [HEDL-7452, 1985], have been investigating the oxidation of spent fuel at temperatures of 110 to 175°C. They have concluded that it is possible that a low-energy activation process of grain-boundary diffusion could lead to oxidation rates greater than that predicted by extrapolation of data obtained at higher temperature. Additional measurements on the diffusion process have been planned [HEDL-7560, 1986]. An important aspect of the oxidation process is the possibility of higher solubilities or leach rates of the oxidized product. Some leach measurements of oxidized fuel have been planned [UCRL-92891] but to our knowledge are not yet available.

Glass

The Pacific Northwest Laboratory report, "Final Report of the Defense High-Level Waste Leaching Mechanisms Program," [PNL-5157, 1984] has been assigned a high priority in the NBS reviews of DOE studies on glass because it is a comprehensive review of the state of the art through 1984. The seven chapters of the report, each of which covers a specific topic, are being treated as individual documents for review purposes. Draft reviews of chapters 1, 3, and 7 will be completed by the end of July. A draft review of chapter 4 is expected to be finished by mid-September. Decisions on expert reviewers for chapters 5 and 6 should be made during August.

There has been considerable recent theoretical and experimental work on the role of Eh in glass leaching. This topic, for example, was discussed during the Defense Waste Processing Facility Technical Exchange Meeting at Savannah River in April. Because the concept of Eh is sometimes misunderstood and misapplied, a number of papers on the role of Eh have recently been assigned for critical review.

WASTE FORM DEGRADATION -- Review is continuing on the following reports.

1. "Chemical Durability Studies on Glass Compositions Pertaining to Waste Immobilization at West Valley," A. Barkatt, et al., Waste Management '86: Waste Isolation in the U.S. Technical Programs and Public Education, March 1986

2. "Long Term Leach Behavior of West Valley HLW Glasses," P. B. Macedo, et al., ANS Spectrum, 1986

3. "Leach Mechanisms of Borosilicate Glass Defense Waste Forms -- Effects of Composition," A. Barkatt, et al., Waste Management '86: Waste Isolation in the U.S. Technical Programs and Public Education, March 1986

4. "Chemical Determination of West Valley Waste Form Products," D. M. Oldman, J. R. Stimmel, and J. H. Marlow, March 1987

5. "Startup and Initial Experimental Results for the West Valley Vitrification Demonstration Project," Waste Management '86: Waste Isolation in the U.S. Technical Programs and Public Education, Volume 2 High-Level Waste, March 1986

6. "Method for Showing Compliance with High-Level Waste Acceptance Specifications," Waste Management '86: Waste Isolation in the U.S. Technical Programs and Public Education, Volume 2 High-Level Waste, March 1986

7. "Solubility Tests on Borosilicate Glasses for West Valley Waste Immobilization, High-Level and Transuranic Waste Management," X. Feng and A. Barkatt, ANS Transactions, 1986

8. "Effects of Composition on the Leach Behavior of West Valley HLW Glasses," X. Feng, et al., September 1986

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

10. WVDP-056, "Description of the West Valley Demonstration Project Reference High-Level Waste Form and Canister," July 1986

11. "Physical Chemistry of Glass Surfaces," J. Non-Cryst. Solids, 1978

12. "Methods of Simulating Low Redox Potential (Eh) for a Basalt Repository," Materials Research Society Proceedings, 1978

13. "Control of Oxidation Potential for Basalt Repository Simulation Tests," Scientific Basis for Nuclear Waste Management, 1975

14. "Glass Performance in a Geologic Setting," Summer National Meeting of the American Institute of Chemical Engineers, 1986

15. "Repository Simulation Tests," American Ceramic Society Annual Meeting, 1984

WASTE FORM DEGRADATION -- Review has been initiated on the following reports.

1. DP-MS-83-135, "Process Technology for Vitrification of Defense High-Level Waste at the Savannah River Plant," Paper for presentation in the proceedings of the American Nuclear Society Meeting on Fuel Reprocessing and Waste Management, August 1984

2. DP-MS-86-96, "Process and Mechanical Development for the Savannah River TRU Waste Facility," Paper proposed for presentation at the American Nuclear Society International Meeting, Spectrum '86, September 1986

OTHER REPORTS/TECHNICAL PAPERS -- Review is continuing on the following report judged to have related scientific value sufficient to warrant its review (as reference material).

1. "Aging Degradation of Cast Stainless Steel," O. K. Chopra and H. M. Chung, October 1986

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.

TASK 3 -- Laboratory Testing

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

During this month, equipment was set up for making measurements on fracture mechanics test specimens of crack length and of acoustic energy emitted during slow crack growth. The crack length is calculated from measured values of electric-resistance of the test specimens. Both measurements will be made using a minicomputer.

Plans were made for the detection and computing of the energy of the acoustic signals associated with slow intergranular cracking in steels. The acoustic energy is to be obtained from the voltage output of a transducer attached to a slowly cracking fracture-mechanics test specimen. A small DC power supply and two converters (DC to rms) were built to assist in these measurements. Materials to be used to establish a relationship between the energy detected and that emitted from the test specimen during intergranular cracking were discussed but conclusions have not been reached. The minicomputer and its peripherals were reconfigured to handle the complex tasks associated with these measurements. Selected commercially available software items must be obtained before this process can be completed.

In the planned experiment, two types of calculations will be done simultaneously: Fracture mechanics parameters will be used in computations of crack length from electric-resistance measurements. This will be done while doing the main task, which is taking data needed to establish the relationship between energy detected by acoustic transducers and energy released by crack-extension events. Software needed for these tasks is being written. When the relationship between energy detected and energy emitted is understood, it can be used to establish the magnitude of the crack extension associated with each acoustic emission.

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

Though preliminary experiments have been carried out to test parts of the measurement system, most of the effort in this program has been directed at setting up the experiment. The experiment is configured to allow the following measurements to be made: (1) corrosion rate of the steel specimen, (2) diffusion rate through the environment, (3) electrical resistivity of the evironment, and (4) moisture content of the environment. Computer programs for the corrosion and diffusion measurement have been written and tested. To reduce effects of convection in the electrolyte, initial experiments have been carried out in agar, and indicate that the system is functioning as planned.

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

The work on this project is on schedule with the work statements listed in the proposal. The work statements and status of the work are given below.

1. Obtain steel, set up equipment and environment (three months from start)

The low-carbon steel, designated A27, was obtained from R. P. Anatatmula, Rockwell Hanford Operations (RHO), Basalt Waste Isolation Project (BWIP).

T. E. Jones of RHO provided the report, SD-BWI-TD-013, entitled "An Evaluation of the Stability of Synthetic Groundwater Formulations GR-3 and GR-4" for use in preparing the water. The water simulating GR-4 was prepared for use in the testing.

Peter Soo of the Brookhaven National Laboratory sent Cohasset Flow basalt and Wyoming bentonite for use in the testing.

The equipment was available already at NBS, so the steel, equipment and environment are ready for the study.

2. Determine the pitting potential, using stimulation techniques in simulated Grande Ronde No. 4 water at 95°C, (three months from start)

Six stimulation tests have been conducted by the co-worker on this project using the NBS standard reference material 1890 (316L stainless steel) in 0.9 percent NaCl water. This has been done to check the equipment and laboratory procedures. Stimulation tests of the A27 steel in Gr-4 water at 95°C, with and without the basalt:bentonite mixture, will be conducted in the near future.

3. Determine polarization behavior and pitting potential using cyclic polarization methods in simulated Grande Ronde No. 4 water at 95°C, (12 months from start)

Initial polarization measurements (anodic and cathodic curves) have been made in GR-4 water at room temperature (approx. 22°C).

These measurements are made in preparation for future tests, and the data will be reported later. Other work has been metallography of the A27 steel to determine the microstructure.

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

The work on this project is on schedule with the work statements listed in the proposal. The work statements and status of the work are given below.

1. Obtain materials and testing environment (three months from start)

Zircaloy-2 and Zircaloy-4 specimen materials were obtained from J. W. Loe of Teledyne Wah Chang Albany, Albany, Oregon. Zircaloy-4 cladding was received from Dave Baty of Babcock & Wilcox, Lynchburg, Virginia. Zircaloy-2 tubing has been requested from General Electric, Vallecitos Nuclear Center, Pleasanton, California. The J-13 water will be made according to the chemical contents listed in NUREG/CR-3091, Vol. 6 prepared by P. Soo. The chemicals were ordered and have been received for this J-13 water.

2. A brief literature survey on Zircaloy corrosion (nine months from start)

A literature search was made using the computerized Engineering Index. Ninety-one references and abstracts were printed on the subject of Zircaloy corrosion. Other indexes were checked and some references will be retrieved later. This literature review is in progress.

3. Anodic polarization curves for Zircaloy in J-13 water at 95°C, (12 months from start)

Anodic polarization curves have been run on Ti-6Al-4V in a saline solution to check the equipment and procedures in preparation for the Zircaloy studies. The Zircaloy-4 and Zircaloy-2 base materials have been prepared metallographically for observing the microstructures and photomicrographs were made.

TASK 4 -- General Technical Assistance

Our reply to your request of June 3, 1987 for general technical assistance, rendered under Task 4 of FIN-A-4171-6, on the topic "Pitting Corrosion Modeling," by Robert B. Moler was transmitted to E. Wick on June 17, 1987.

Dr. Richard Ricker attended a meeting of the NRC waste disposal staff on June 22, 1987 in Silver Spring, Maryland. The purpose of this meeting was for Michael McNeil to brief NRC staff on results of his recent meetings with the various DOE project offices and visits to research facilities.

TRAINING and TECHNICAL MEETINGS

Dr. Anna Fraker attended the first two days of the conference, "International Conference on Localized Corrosion," on June 1-5, 1987 in Orlando, Florida. This meeting provided an opportunity to hear presentations by leading corrosion scientists on sessions entitled "Passivity and Breakdown, Kinetics of Pitting and Crevice Corrosion, Pitting Stochastics and Dissolution Kinetics and Morphologies." Much of the information given dealt with basic principles of formation and growth of pits. Papers on pitting stochastics indicated increased interest in this approach. T. L. Yau and M. Maguire of Teledyne Wah Chang Albany, Albany, Oregon, presented a paper on "Control of Localized Corrosion of Zirconium in Oxiding Chloride Solution," which showed that zirconium is highly corrosion resistant in hydrochloric and most chloride solutions if the solution is not contaminated with an oxidizer and the zirconium is not anodically polarized.

Dr. Charles Interrante and Dr. Dale Hall went to Corning, New York, to meet with contractor, Dr. Bruce Adams. While at Corning they toured the melting operations at the Falls Brook Plant. From June 3-5, they attended a course given at Alfred University on the topic "Glass: Its Production and Properties." This training was needed to assist these workers in their development in this aspect of materials science. Previous formal training for each of them had not included glass production and properties, and this course provided an excellent survey of these topics.

Dr. Charles Interrante attended a course, "Electrochemical Test Methods" on June 22-26, 1987. The course served to improve this worker's understanding of the field, so as to permit him to better understand, represent and guide work being done by others on electrochemical testing related to waste package problems.

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

DATA SOURCE

(a) Organization Producing Data: Argonne National Laboratory

Chemical Technology Division, 9700 South Cass Avenue, Argonne, Illinois 60439

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

Abrajano, T., Bates, J., Ebert, W., and Gerding, T., "The Effect of Gamma Radiation on Groundwater Chemistry and Glass Leaching as Related to the NWWSI Repository Site". UCRL-15825, Preprint SANL-510-001, May 1986.

DATE REVIEWED: 4/13/87; Revised 6/23/87

TYPE OF DATA

Leach and pH data.

MATERIALS/COMPONENTS

SRL 165 glass containing U, Cs, and Sr, (SRL U glass); SRL U glass containing added ^{237}Np, ^{239}Pu, and ^{241}Am, (SRL A glass), J-13 water pre-equilibrated at 90°C for two weeks with Topopah Spring tuff, (EJ-13), tuff, 304L SS, air.

TEST CONDITIONS

Detailed test conditions have been presented by Bates (see related reports below) but are repeated here except for changes. Tests were run in type 304L stainless steel reaction vessels which contained the sample submerged in EJ-13 water with an air space at the top. Vessels were sealed from the environment with silicone gaskets and exposed to a gamma radiation field produced by a ^{60}Co source at $1x10^4$ rd/h. The temperature was maintained at 90°C. The test matrices were as follows: (1) two SRL U disks in pre-equilibrated J-13 water (EJ-13), Surface area/Volume (SA/V) = 0.3 cm^{-1}, R= gas volume/liquid volume =0.3; (2) two SRL A disks in EJ-13 water with a polished tuff core wafer, SA/V= 0.3 cm^{-1}, R= 0.3; (3) EJ-13, R= 0.3; (4) EJ-13 with a polished tuff core wafer, R= 0.3. Each matrix was run at 90°C for periods of 14, 28, 56, 91, and 182 days. Duplicate samples were run for each time period. When possible, the protocol of the Materials Characterization Center (MCC-1) was followed.

METHODS OF DATA COLLECTION/ANALYSIS

Solutions were cooled to room temperature and analyzed for pH, cations by inductively coupled plasma spectroscopy (ICP), anions by ion chromatography (IC), uranium by atomic fluoresence (AF), Cs by atomic absorption spectroscopy (AA), and radionuclides by gamma and alpha counting, and dissolved gases by Van Slyke gas chromatography. The solid test components were measured for weight change and were analyzed by scanning electron microscopy (SEM) and associated energy dispersive X-ray analysis (SEM/EDS), secondary ion mass spectroscopy (SIMS), nuclear resonance profiling and ion microprobe.

AMOUNT OF DATA

Tables;

1. Concentration of Fixed Nitrogen Species for blank, SRL U, and SRL A glass. Concentrations of NO_2^-, NO_3^- and total in nanomoles/ml for EJ-13 and EJ-13 + tuff after 14, 28, 56, 91 and 182 days.
2. Concentration, Normalized Transuranic Release, and pH Values for Tests Containing SRL A Glass for EJ-13 and EJ-13 + tuff. Lists period in days, pH, concentrations for unfiltered, and acid strip reported in disintegrations/second per 100 lamda of solution and normalized release rate for Am, Pu and Np respectively.

Figures;

1. Variation of pH with time for the non-tuff containing tests; pH (6.5 to 8.0) vs time in days (0 to 200) for SRL A, SRL U, and EJ-13 tests.
2. Normalized loss for (a) Li, (b) Na, (c) B, and (d) mass for SRL A, SRL U, SRl A + tuff and SRL U + tuff. Normalized release rate in g/m^2 (0 to 6) vs time in $days^{1/2}$ (0 to 16).
3. SIMS profile of SRL U glass + tuff leached for 91 days showing lithium (7), boron (11), sodium (23) and iron (56). Intensity ratio to m/e (mass/ionic charge) of 28 (0 to 4) vs sputter time in minutes (0 to 84). (Indicates concentration of element vs depth in glass).

UNCERTAINTIES IN DATA

Not dealt with.

DEFICIENCIES/LIMITATIONS IN DATABASE

All tests showed gradually increasing acidity for 91 days with an actual reversal of the trend at 182 days. This observation was believed to result from loss of acid volatiles from the system because of slight leakage from the test vessels. Loss of CO_2 and N_2 from the gas phase would reduce the amount of bicarbonate in the solution phase and decrease the production of nitric acid by radiolysis.

RELATED HLW REPORTS

Bates, John K., Fischer, Donald F., and Gerding, Thomas J., "The Reaction of Glass During Gamma Irradiation in a Saturated Tuff Environment, Part 1, SRL 165 Glass", ANL-85-62, (1985).

KEY WORDS

experimental data, supporting data, weight change, ICP, IC, AF, AA, SEM, SEM/EDS, SIMS, laboratory, air, J-13 water, tuff, cobalt 60, high temperature, acidic solution, basic solution, doped glass, ^{237}Am, ^{239}Pu, ^{241}Am

GENERAL COMMENTS

This report presents data on the influence of gamma irradiation on the reaction of actinide doped borosilicate glass (SRL 165 A and SRL 165 U) in a test environment containing tuff. This is a continuation of experiments at various radiation doses (in this case, 1 x 10^4 Rad). The objective is to determine the effects of gamma radiation on air and groundwater and how these interact to change the pH of the leachate and the resulting interaction with and nuclear glass in tuff. In the gamma radiation field, nitric acid is generated by radiolysis of air; thus the solution becomes more acidic. This effect is counteracted by dissolution of the glass and buffered by the bicarbonate in the tuff. Major changes in groundwater in the blank tests show a rapid lowering of the pH to the buffered region (6.4). This pH would be maintained unless the buffering capacity of the groundwater was exceeded by acid production, or acid production was decreased because of depletion of N_2 from the gas phase. The radiation levels at the end of the containment period are expected to be less than 100 rd/hr or about a factor of 100 less than those in the experiments reported in this paper.

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

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

Related to issue 2.3.2.1.2 in the ISTP for the NWWSI Project which involves the rates of dissolution associated with the potential mechanisms of waste form dissolution.

(b) New Licensing Issues

(c) General Comments

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

DATA SOURCE:

(a) Organization Producing Data

Argonne National Laboratory, Argonne, Illinois 60439

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

Bates, John K., and Gerding, Thomas J., "One-Year Results of the NWWSI Unsaturated Test Procedure: SRL 165 Glass Application", ANL-85-41, August 1986.

DATE REVIEWED: 1/5/87; Revised 6/12/87

TYPE OF DATA

Leach data (unsaturated conditions).

MATERIALS/COMPONENTS

J-13 well water equilibrated with tuff at 90°C (referred to as EJ-13 water)· type 304L stainless steel, tuff, glass made by mixing SRL-165 black frit with appropriate amount of UO_2, SrO, and $CsNO_3$ to provide a glass with approximately 1% U, and 0.1% Sr and Cs.

TEST CONDITIONS

The unsaturated test procedure refers to the moisture content of the tuff and the fact that the glass test material is not totally immersed in groundwater. The reaction vessel allows J-13 water droplets of 0.1 ml per drop to drip at a predetermined leak rate onto the waste form, which in this case is a cylindrical doped SRL-165 glass. Water drips into a reservoir at the base of the test vessel to allow for simulation of unsaturated conditions at a temperature of 90°C and pressure of 1 atm. The glass specimen, representing the waste form, is held in position by a stainless steel holder consisting of perforated plates above and below the specimen, the plates were held together by stainless steel rods. The rods were welded to the bottom plate using Tungsten Inert Gas (TIG) welding. Groundwater may be circulated, by evaporation and condensation due to temperature gradients in the test apparatus, but this is not made clear in the report. See general comments below.

METHODS OF DATA COLLECTION/ANALYSIS

Specimens representing two of the anticipated waste-package components (waste form and canister) are contacted intermittently by dripping water. The glass was analyzed for phase separation using x-ray diffraction (XRD) and scanning electron microscopy (SEM). The glass composition was determined by colorimetry (Si, P), atomic absorption (Cs), isotopic dilution mass spectrometry (B), and inductively coupled plasma spectroscopy (ICP), (all other elements). Many of the deposits, glass surfaces and steel parts of the test apparatus were characterized visually, by Scanning Electron Microscopy/Energy Dispersive Spectroscopy (SEM/EDS) and Secondary Ion Mass Spectrometry (SIMS).

AMOUNT OF DATA

Figures:
1. NWWSI Waste Form Test Apparatus. (a) Cutaway view, (b) Schematic, (c) Selected parts.
2. Photograph of a Reacted Sample of Glass from Test F-4.
   Figures 3a-18 contain twenty one photomicrographs of selected reaction sites or reaction products at specified locations; analysis of top and bottom of canister by SIMS; and SIMS Spectrum of the Nonsensitized Surface of the Reference Type 304L Stainless Steel.

Appendix 3. Modified Test Vessel Design.

Tables:
1. Test Matrix for Unsaturated Tests.
2. Composition of SRL Glass Used in Testing.
3. General Observations of Canister Sections
4. General Observations of the Glass after Disassembly.
5. Water Loss During Testing.
6. Analyses of Equilibrated J-13 Water Used in Unsaturated Testing.
7. Results of the Blanks for the Unsaturated Tests.
8. Solution Results for the Continuous Tests.
9. Solution Analyses for the NWWSI Unsaturated Tests.
10. Results from the Dissolution of Tuff Cups.
11. Analyses Performed on Components from the Unsaturated Tests.
12. Component Weight Changes in NNWSI Unsaturated Tests.
13. Composition of 304L Stainless Steel Used in the Unsaturated Tests.
14. SIMS Analysis of Tuff Samples.
15. Total B, Li, and U Release from the Waste Form.
16. Solution Results of Glass Parametric Test No. 1.
17. Test Conditions and Selected Results from the Analog Tests.

Appendix 1. Conditions and Selected Results of the Unsaturated Test Matrix.

Appendix 2. Detailed Solution Analyses for the Unsaturated Tests.

UNCERTAINTIES IN DATA

Uncertainties in some of the chemical analyses are discussed briefly.

DEFICIENCIES/LIMITATIONS IN DATABASE

The objective of tests was to obtain leach data as a function of time with a "reasonable" trend in elemental release required for modeling repository behavior was not achieved because apparently different release trends were observed owing to sensitization of stainless steel.

RELATED HLW REPORTS

J. K. Bates and T. J. Gerding, "NNWSI Phase II Materials Interaction Test Procedure and Preliminary Results," ANL-84-81 1984.
Van Konynenburg, R. A., "Radiation Chemical Effects in Experiments to Study the Reaction of Glass in an Environment of Gamma-Irradiated Air, Groundwater, and Tuff", UCRL-53719, May 1986.

KEY WORDS

data analysis, experimental data, supporting data, x-ray diffraction, visual examination, weight change, SIMS, SEM/EDS, SEM, laboratory, air, J-13 water, tuff, high temperature, stainless steel, 304L stainless steel,doped SRL-165 glass, ^{238}U

GENERAL COMMENTS

This report describes leach data obtained using an unsaturated test procedure. In this leach procedure, the waste form is not totally immersed in water but the method devised does allow liquid water to be in contact with the waste form if water is present and the temperature gradients in the canister are appropriate. The section of the report describing the means of water injection into the test apparatus is totally inadequate. In ANL 84-81 (1984), which is a more detailed report of the test method, the authors describe an injection system in which water and air are injected into the apparatus in prescribed amounts and intervals. In this report, the method described in ANL 84-81 is not mentioned. The statement that the "temperature differential between the waste package and the test vessel is based on the thermal equilibration of test component." could be interpreted as meaning that the water flow is driven by a temperature gradient. There is also some question concerning the pressure at which the measurements are made. Although it is stated that the pressure is 1 atm, van Konynenburg's analysis of similar measurements indicated higher pressures due to expansion of the initial air and the partial pressure of water at 90°C.

Much of this report deals with interactions, between the waste form, water, and stainless steel, resulting from sensitization of stainless steel due to tungsten-inert-gas weldments on the waste support holder. Since there may be weldments on the stainless steel canisters, the effect of sensitized stainless steel on the leaching kinetics of the waste form may be important. Most nuclear waste glasses may undergo leaching simultaneously by both hydration and matrix breakdown. Hydration initially occurs more rapidly than matrix breakdown but since it follows parabolic ($t^{1/2}$) kinetics, the penetration of the hydration front slows with time. When sensitized stainless steel is present, the formation of Fe, Ni, and Cr silicates increases the rate of matrix breakdown so that it may become the dominant leach process. Matrix breakdown is a process in which the constituents that form the matrix of the glass are simultaneously released so that normalized release rates of all the elements are the same. The kinetics are linear with time. The extent of interaction of leachate with stainless steel in the data obtained in this report varied broadly from extensive to slight.

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

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

This report is related to issue 2.3 concerning when, how, and at what rate radionuclides will be released from the waste form.

(b) New Licensing Issues

(c) General Comments

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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, University of California, Livermore, California 94550

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

Van Konynenburg, R. A., "Radiation Chemical Effects in Experiments to Study the Reaction of Glass in an Environment of Gamma-Irradiated Air, Groundwater, and Tuff", UCRL-53719, May 1986.

DATE REVIEWED: 5/6/87; Revised 6/9/87

TYPE OF DATA

Discussion of kinetic modeling of chemical reactions induced by gamma radiation of air, groundwater and other waste package components.

MATERIALS/COMPONENTS

Considers experimental data reported by Bates et al. and Abrajano et al. (See related HLW reports below). This report considers experiments involving SRL U glass, SRL A glass; J-13 water pre-equilibrated at 90°C for two weeks with Topopah Spring tuff (EJ-13 water), tuff, 304L SS, and air.

TEST CONDITIONS

As reported by Bates and Abrajano, leach tests were run in stainless steel reaction vessels. The samples were submerged in about 16 ml of EJ-13 water with at least 4 ml of air at the top. These vessels were sealed from the environment and exposed to a gamma radiation field produced by a ^{60}Co source of $2x10^5$ (Bates) or $1x10^5$ rd/h (Abrajano) at a temperature of 90°C. Samples consisted of two glass disks or two glass disks with tuff. Tests for each of the glasses were run in duplicate for 7, 14, 28, and 56 days (Bates) or 14, 28, 56, 91, and 182 days (Abrajano). Similar tests were run on two blanks containing EJ-13 water and EJ-13 water plus tuff.

METHODS OF DATA COLLECTION/ANALYSIS

Details of data collection and analysis are reported in detail in the review of Bates et al. and Abrajano et. al. cited below.

AMOUNT OF DATA

Tables:
1. Composition of AISI standard type 304L stainless steel.
2. Composition of SRL glasses.

3. Composition of Topopah Spring tuff expressed as oxides.
4. Composition of "Equilibrated J-13" water at 20°C (first set of experiments).
5. Parameters influencing composition of gas and liquid phases. Parameter,(e.g., vapor pressure of water), value at 20°C, value at 90°C, reference.
6. Changes in compositions of gas and liquid phases due to heating, in vessels originally containing 16 ml of water. Gas phase volume, pressure, liquid phase volume, concentration of species at 20 and 90°C.
7. Composition of standard dry air.
8. Primary products of gamma irradiation of liquid water for pH of 5 to 9.
9. Reactions and rate constants for pure water at 25°C (from 4 sources).

Figures:
1. Nitrogen fixation in 2 x 10^5 rd/h experiment, Increase in concentration of fixed nitrogen in 10^{-4} M (-.40 to 1.60) vs Time in days (0 to 60).
2. Nitrite-nitrate ratios in 10^4 rd/h experiments, Ratio $[NO_2^-]$(M)/$[NO_3^-]$(M), (0 to 7) vs Time in days (0 to 200).
3. Nitrite-Nitrate ratios in 2x10^5 rd/h experiments, Ratio $[NO_2^-]$(M)/$[NO_3^-]$(M) (0 to 2) vs Time in days (0 to 60).

UNCERTAINTIES IN DATA

Uncertainties in anion analyses of ± 5%, dose ± 10%, yield (No. of particles of a particular species created or destroyed per 100 ev) ± 15%.

DEFICIENCIES/LIMITATIONS IN DATABASE

The author states, "Since a thoroughgoing computer model of the system of interest was not available, I elected to perform an approximate analysis of the radiation chemistry considering only the dominant reactions."

KEY WORDS

air, cobalt 60, data analysis, effect of radiation on leaching, gamma radiation field, J-13, kinetic model, SRL U glass, SRL A glass, theory, tuff

COMMENTS

This report discusses the formation of a number of species which could be formed by interaction of gamma radiation with components present in a repository setting. Initially, in this report an attempt to model the experimental results obtained in leach experiments by Bates et al and Abrajano et al is reported. Van Konynenburg used an equation developed by Burns et al., which predicts that the formation of fixed nitrogen (i.e., nitrites or nitrates), will be proportional to the ratio of the volume of air to liquid and the total dose. This equation gives a reasonable fit to the experimental data. Depending on the conditions, many possible reaction products can result from radiolysis. For the experiments considered here, the principal reactions include the following: (1) a net radiolysis of water into H_2 and O_2, (2) nitrogen fixation to NO_2^- and NO_3^-

with the formation of an equivalent amount of H^+ which tends to lower the pH against the buffering action of dissolved carbonate, tuff, and glass, and (3) the formation of a colloidal Fe^{+3} substances suspended in the solution as a result of interactions with the vessel walls. It should be noted that when groundwater first breeches the waste canister some 300 to 1000 years after closure, the gamma radiation dose rate will be at least 3 orders of magnitude below those used here.

RELATED HLW REPORTS

Bates, John K., Fischer, Donald F., and Gerding, Thomas J., "The Reaction of Glass During Gamma Irradiation in a Saturated Tuff Environment, Part 1, SRL 165 Glass", ANL-85-62, (1985).

Abrajano, T., Bates, J., Ebert, W., and Gerding, T., "The Effect of Gamma Radiation on Groundwater Chemistry and Glass Leaching as Related to the NWWSI Repository Site". UCRL-15825, Preprint SANL-510-001.

Burns, W. G., Hughes, A. E., Marples, J. A. C., Nelson, R. S., and Stoneham, A. M., "Effects of Radiation on the Leach Rates of Vitrified Radioactive Waste," J. Nucl. Mater. 107, 245 (1982).

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

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

Related to issue 2.4.3 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project concerning how the radionuclide species (i.e., paticles, colloids and solubles) change with time in the waste package.

(b) New Licensing Issues

(c) General Comments