Evaluation and Compilation of DOE Waste Package Test Data

NUREG/CR--4735-Vol. 2

TI88 900155

Biannual Report: August 1986-January 1987

Manuscript Completed: May 1987
Date Published: October 1987

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U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A4171

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ABSTRACT

This report summarizes results of the National Bureau of Standards (NBS) evaluations of Department of Energy (DOE) activities on waste packages designed for containment of radioactive high-level nuclear waste (HLW). The waste package is a proposed engineered barrier that is part of a permanent repository for HLW. Metal alloys are the principal barriers within the engineered system. Technical discussions are given for the corrosion of metals proposed for the canister, particularly carbon and stainless steels, and copper. In the section on tuff, the current level of understanding of several canister materials is questioned. Within the Basalt Waste Isolation Project (BWIP) section, discussions are given on problems concerning groundwater, materials for use in the metallic overpack, and diffusion through the packing. For the proposed salt site, questions are raised on the work on both the primary candidate material (ASTM A216 Steel) and the alternate alloy (Ti-Code 12). NBS work related to the vitrification of HLW borosilicate glass at the West Valley Demonstration Project (WVDP) and the Defense Waste Processing Facility (DWPF) is covered. NBS reviews of selected DOE technical reports and a summary of current waste-package activities of the Materials Characterization Center (MCC) is presented. Using a database management system, a computerized database for storage and retrieval of reviews and evaluations of HLW data has been developed and is described.

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EXECUTIVE SUMMARY

This is the second biannual progress report on the National Bureau of Standards (NBS) assessments of the Department of Energy (DOE) activities related to the waste package for disposal of radioactive high-level waste (HLW). It contains NBS reviews conducted over the period August 1, 1986 to January 31, 1987 on DOE reports, and status reports on NBS assessments of DOE activities over the same period. This executive summary highlights key issues and accomplishments.

The initial implementation of software required to operate a computerized database for storage and retrieval of reviews and evaluations of HLW data has been made. On March 2, 1987, the database system was activated to serve users at the Nuclear Regulatory Commission (NRC) and the NBS. The database system will be used to store all completed NBS reviews. In the future, other selected information will be stored in this database so as to permit rapid searches of the most pertinent technical documents on high-level waste package topics.

In the status report for the Nevada Nuclear Waste Storage Investigations (NNWSI) Project, questions are raised regarding the following topics:

1. Sampling methods and analytical data for groundwater samples.
2. The possibility for either localization of corrosive action or accelerated corrosion in irradiated ferrous borehole liner materials.
3. The viability of a 9Cr-1Mo steel and nodular cast iron in a repository environment.
4. The validity of the conclusion that 304L stainless steel will undergo uniform corrosion in a simulated repository environment.
5. The effectiveness of Zircaloy as a final barrier to radionuclide sorption.
6. The current level of understanding of the behavior of copper and its alloys for this repository environment.
7. The available data on solubility and sorption for spent fuel.

For the Basalt Waste Isolation Project (BWIP), further discussions are given on the three major deficiency areas that were highlighted in the previous biannual report [Interrante 1986b]. These areas are (1) the groundwater chemistry, (2) the available data pertaining to packing material breakdown, and (3) the corrosion behavior of the metallic overpack. Further work on organic substances

found in irradiated basalt groundwater is called for. On the topic of packing materials, the tendencies for actinides to be adsorbed by basalt/bentonite packing materials is highlighted, as the packing may serve to slow the rates at which radionuclides are released to the borehole of a waste package. For the candidate metallic overpack materials in basalt groundwater, pitting has been observed in two carbon steels, ASTM A27 and A36, as well as in two low-alloy cast steels tested, and the pitting was observed to be accelerated by gamma radiation. Other adverse effects of basalt groundwater have been indicated by a demonstrated sensitivity to basalt groundwater in slow-strain-rate tests of ASTM A27 steel and ASTM A387-9 steel.

For the Salt Repository Project (SRP), better characterization of the Deaf Smith site is needed so that long term corrosion tests can be initiated in appropriate environments. In addition, questions are raised on the work being done on both the primary candidate material for waste containment, ASTM A216 steel, and the alternative alloy Ti-Code 12. For the A216 steel, the weldments may be susceptible to hydrogen embrittlement. Non-linear corrosion kinetics indicate the formation of surface films which may lead to non-uniform attack. This suggests that further study of uniform and non-uniform corrosion is required to model the corrosion processes. For Ti-Code 12, the alternative alloy, hydrogen embrittlement has not been eliminated as a probable failure mode. The resistance to degradation in repository conditions for either material is, at this time, questionable. Further materials research is called for. Other alternative materials should be sought and new approaches should be explored. Some general suggestions are made.

NBS studies of the vitrification of HLW and the durability of borosilicate glass are being accelerated, prompted by two recent information meetings with DOE personnel involved in the West Valley Demonstration Project and the Defense Waste Processing Facility. In addition, our technical staff has been enhanced by three qualified contractors with expertise in glass technology.

Waste-package related activities of the Materials Characterization Center (MCC) are summarized for the period May 1, 1986 through December 31, 1986. The work is to ensure that qualified materials data are available on waste materials.

1.0 INTRODUCTION

A principal objective of the NBS studies on high-level waste (HLW) is to review and assess the DOE's waste package development activities. This is done by following the work of the three DOE project offices and the Materials Characterization Center (MCC). In this biannual progress report, status reports on the activities of the three project offices are given, respectively, as sections 2 through 4: Nevada Nuclear Waste Storage Investigations (NNWSI), Basalt Waste Isolation Project (BWIP), and Salt Repository Project (SRP). Recent activities of the MCC are summarized in section 6.

NBS reviews and evaluations conducted over the period August 1, 1987 through January 31, 1987 are included as Appendix A; contributing reviewers for these works are acknowledged as a group on the cover page of this report. As reviews are completed and approved for publication, they will be included in the database discussed below.

A significant accomplishment of the NBS for this period is the implementation of a database for reviews and evaluations of HLW data. This system uses a database management system tailored to meet the needs of users at the NRC and the NBS. While much more work will be needed to make the system fully operational, the training aids for users, the entry instructions for the data clerk, and the implementation of a revised file structure have reached the level needed to begin data entry. At the beginning of March 1987 the system was made available to users for training at both the NRC and the NBS. Details are given in section 7 of this report.

The NBS work on vitrification of high-level waste increased during this period. NBS participated in two meetings at which presentations were made on the activities of the West Valley Demonstration Project (WVDP) for commercial HLW and the Defense Waste Processing Facility (DWPF) located at the Savannah River Plant (SRP) for defense HLW. This work is presented as section 5.

Studies involving laboratory testing at the NBS have begun in two areas that require further verification of DOE data and test methods. The overall objective of the laboratory testing studies at the NBS is to confirm the accuracy of DOE data and the validity of the conclusions deduced from it.

One of the studies, on pitting of low-carbon steels, was prompted by evaluations of the available data for the Basalt Waste Isolation Project (BWIP). These NBS tests will relate to the methods used in tests of the pitting potentials of low-carbon steels. NBS results will be compared with BWIP pitting-potential data and conclusions. Laboratory equipment needed for these tests is in place. Materials for use in conducting the tests include basalt from the Cohasset Flow and bentonite from Wyoming. These have been obtained from P. Soo of BNL. Synthetic groundwater has been prepared using the formulations and directions furnished by T.E. Jones of Rockwell Hanford Operations, Richland, Washington. These formulations are designed to represent the groundwaters of Grande Ronde -3 and -4.

The other laboratory study, related to the durability of Zircaloy cladding on fuel rods, was prompted by recent reviews of corrosion of this material. The reviews indicate that insufficient data are available on oxidation and stress corrosion cracking (SCC) of Zircaloy, and this insufficiency will preclude sound judgments on expected repository performance. Reviews of corrosion tests on Zircaloy planned for use by the NNWSI project indicate an absence of electrochemical data needed for reasonable conclusions regarding corrosion processes on this material. Therefore, surface oxidation behavior will be characterized at the NBS and related to expected localized corrosion, such as SCC. A literature review on corrosion of Zircaloy is in progress. Zircaloy that is used in simulating cladding of fuel rods and J-13 groundwater will be obtained for these tests.

Proposals for two additional NBS laboratory studies for verification of DOE data and test methods have been submitted to the NRC for approval. These studies are in fracture mechanics and underground corrosion. Plans for experiments to meet needs in these areas are in progress so that work can be initiated immediately upon approval of the work plans.

2.0 NNWSI- NEVADA NUCLEAR WASTE STORAGE INVESTIGATIONS

The proposed Yucca Mountain repository site is located in southern Nevada about 100 miles northwest of Las Vegas. The repository will be located approximately 400 meters (m) below the surface in a compacted, hardened volcanic ash referred to as welded tuff. This level is about 100 m above the water table, an attractive feature of this site. Water influx in the welded tuff is estimated to be less than one mm per year and the total moisture content is six percent by weight. Chemical analysis of water from local wells indicates that the environment is oxic and relatively noncorrosive.

The following is a summary of tuff data based on NBS reviews of publications from DOE sponsored research. The NBS reviews considered include those conducted over the period December 1985 to January 1987. Included with the following summaries are discussions of research areas where further work is needed. This evaluation is divided into six categories: 1) groundwater 2) steel 3) stainless steel 4) Zircaloy, 5) copper and 6) spent fuel. Preceding the discussions for each category are lists of the most pertinent documents reviewed to date.

2.1 Groundwater

"Groundwater Chemistry Along Flow Paths Between a Proposed Repository Site and the Accessible Environment", A. E. Ogard, J. F. Kerrisk, Los Alamos National Laboratory, LA-10188-MS, November 1984 [See Appendix A, p A-21].

As expected, the water analysis from several samples taken from the wells in the vicinity of Yucca Mountain indicate that, in general, total dissolved solids (TDS) are low (200 - 400 mg/l) with one sample from the carbonate aquifer containing over 1000 milligrams per liter (mg/l) TDS. The environment is oxidizing in the unsaturated zone, as expected, but the data indicate that the environment is reducing in the saturated zone below the proposed repository. This is significant in that it would tend to reduce the oxides of uranium (U), plutonium (Pu), neptunium (Np), and technetium (Tc) to their most insoluble forms.

Questions remain about the water samples: How representative are they for the repository and its surrounding environment. The oxidation/reduction buffering capacity of the tuff repository environment is unknown. Groundwater flow rates, quantities, and directions are also unknown.

2.2 Steel

"Corrosion Behavior of Carbon Steels Under Tuff Repository Environmental Conditions", R. D. McCright, H. Weiss, Lawrence Livermore National Laboratory, UCRL-90875, October 1984 [See Interrante 1986b].

The possibility of using carbon steel (A36, AISI 1020, and gray cast iron) borehole liners for nuclear waste containers in tuff has prompted studies on the corrosion behavior of steel in this environment. Initial results indicate that uniform corrosion rates are low, of the order of 27 micrometers per year ($\mu\text{m}/\text{y}$) after 5000 H in J-13 water and 12 ($\mu\text{m}/\text{y}$) in steam. However, localization of this attack can increase penetration rates to 15 times that of uniform corrosion. Corrosion in irradiated J-13 water results in increased attack of the steel. This increase in corrosivity of J-13 water is attributed to the formation of oxygen, hydrogen peroxide, and other oxidizing species by radiolysis of the water.

Stress corrosion tests in J-13 water and steam indicate that in the welded-and-stressed condition, both a 9Cr-1Mo ferrous alloy and a nodular cast iron will fail in this environment. However, carbon steel specimens under similar conditions of stress and in the same environment did not fail.

These preliminary tests indicate that corrosion of carbon steel and other ferrous alloys is significant, and studies in simulated repository environments must be continued.

2.3 Stainless Steel

"Attachment 1 to MRB-0418, Letter Dated 8 March 1985", L. B. Ballou and R. D. McCright [Review in progress].

"Behavior of Stressed and Unstressed 304L Specimens in Tuff Repository Environmental Conditions", M. C. Juhas, R. D. McCright, and R. E. Garrison, Lawrence Livermore National Laboratory, UCRL-91804, November 1984 [See Interrante 1986b].

"An Overview of Low Temperature Sensitization", M. J. Fox, R. D. McCright, Lawrence Livermore National Laboratory, UCRL-15619, December 1983 [See Interrante 1986b].

"Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff", R. S. Glass, G. E. Overturf, R. E. Garrison, and R. D. McCright, Lawrence Livermore National Laboratory, UCID-20174, June 1984 [See Interrante 1986b].

"Laboratory Experiments Designed to Provide Limits on the Radionuclide Source Term for the NNWSI Project", V. M. Oversby, R. D. McCright, Lawrence Livermore National Laboratory, UCRL-91257, November 1984 [See Interrante 1986b].

"Metallurgical Analysis of a 304L Stainless Steel Canister from the Spent Fuel Test - Climax", H. Weiss, R. A. Van Konynenburg, and R. D. McCright, Lawrence Livermore National Laboratory, UCID-20436, April 1985 [See Interrante 1986b].

"Stress Corrosion Cracking Tests on High-Level-Waste Container Materials in Simulated Tuff Repository Environments", T. Abraham, H. Jain, P. Soo, Brookhaven National Laboratory, NUREG/CR-4619, June 1985 [See Appendix A, p A-10].

Austenitic stainless steel type 304L is the candidate canister material for the proposed tuff repository. One laboratory reports very low (0.001-0.008 mils per year) corrosion rates for this material in a simulated repository environment, J-13 water. However, these results are based on electrochemical-polarization-data calculations that assumed attack to be by uniform corrosion. This assumption is questionable based on other studies performed by this same laboratory that show 304L susceptible to crevice corrosion. This assumption is important because the assumption of uniform corrosion over the surface of the specimen leads to lower penetration rates as compared to rates obtained from an assumption that the same weight loss (total amount of corrosion) occurs over only a few localized areas. This issue of uniform versus localized corrosion must be investigated in depth, in order to develop a reliable data base for modeling purposes.

Perhaps of greater importance to the acceptance of type 304L as a waste container material, is its reported susceptibility to stress corrosion-cracking. Stress corrosion studies using C-ring specimens indicate that Types 304L, 316L, and 321 are all susceptible to cracking in the steam/air phase of J-13, and 10x J-13 water at 100°C. These studies also report that Incoloy 825 appears to be resistant to failure in this same environment.

2.4 Zircaloy

"Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Nuclear Fuel in a Tuff Repository", A. J. Rothman, Lawrence Livermore National Laboratory, UCID-20172, Sept 1984 [See Appendix A, p A-40].

"Spent Fuel Cladding Corrosion under Tuff Repository Conditions - Initial Observations", H. D. Smith and V. M. Oversby, Lawrence Livermore National Laboratory, UCID-20499, June 1985 [See Interrante 1986b].

"Zircaloy Spent Fuel Cladding Electrochemical Corrosion Experiment at 170°C and 120 PSIA H₂O", H. D. Smith, Hanford Engineering Development Laboratory, HEDL-7545, April 1986 [Review in progress].

The Zircaloy cladding that sheaths the spent fuel is an important barrier to the release of radionuclides into the environment. Calculations based on known corrosion rates of Zircaloy in a water/steam environment show that penetration of the 800 μ m Zircaloy wall will not occur in 10,000 years. However, other forms of failure, such as brittle fracture due to oxygen segregation and Zr₃O precipitation arising from local deformation, may develop. Furthermore, embrittlement of the Zircaloy may occur through hydriding for two reasons: 1) Hydrogen will be present as a result of oxidation reactions that form hydrogen and 2) hydrolysis of water will occur in the presence of gamma radiation. Failure of the embrittled Zircaloy can then take place if internal fuel rod pressure is sufficient to cause failure at the embrittled site. Questions remain on depth of cracks and possible hoop loads that may develop in 10,000 years.

Stress corrosion cracking of Zircaloy fuel rods has been observed in service. This form of failure is attributed to rapid temperature increases during operation, a condition unlikely to arise in a repository. However, more research is needed. The effectiveness of Zircaloy, as the final barrier before radionuclide sorption can begin, has not yet been established.

2.5 Copper

"NNWSI Test Plan For Copper and Copper Base Alloys", Attachment 2 to MRB-0418, L. B. Ballou, R. D. McCright, March 1985 [Review in progress].

"FY 1985 Status Report on Feasibility Assessment of Copper-Base Waste Package Materials in a Tuff Repository", R.D. McCright, Lawrence Livermore National Laboratory, UCID-20509, September 1985 [See Interrante 1986b].

"Corrosion of Copper-Based Materials in Gamma Radiation", W. H. Yunker, Hanford Engineering Development Laboratory, HEDL-7612, June 1986 [See Appendix A, p A-37]

Experiments conducted in J-13 water and water vapor indicate that pure copper (CDA 101) has less tendency to localized attack when compared with the other alloys tested, and that the pitting occurs in both a Bronze alloy, (CDA 613), containing seven percent aluminum and a 70Cu-30Ni (CDA 715) alloys. The presence of moisture droplets on the metal surface enhances corrosion. A significant enhancement of corrosion with gamma radiation exposure was found only for the 70Cu-30Ni alloy.

2.6 Spent Fuel

"Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater", C. N. Wilson, V. M. Oversby, Lawrence Livermore National Laboratory, UCRL-91464, March 1985 [See Interrante 1986b].

"Derivation of a Waste Package Source Term for NNWSI from the Results of Laboratory Experiments", V. M. Oversby, C. N. Wilson, Lawrence Livermore National Laboratory, UCRL-92096, September 1985 [See Interrante 1986b].

"Low Temperature Spent Fuel Oxidation Under Tuff Repository Conditions", R. E. Einziger, R. E. Woodley, Hanford Engineering Development Laboratory, HEDL-SA-3271FP, March 1985 [See Appendix A, p A-7].

"Evaluation of the Potential for Spent Fuel Oxidation Under Tuff Repository Conditions", R. E. Einziger, R. E. Woodley, Hanford Engineering Development Laboratory, HEDL-7452, March 1985 [See Appendix A, p A-28].

"Test Plan for Series 2 Thermogravimetric Analyses of Spent Fuel Oxidation", R. E. Einziger, R. E. Woodley, Hanford Engineering Development Laboratory, HEDL-7556, February 1986 [See Appendix A, p A-31].

"Important Radionuclides in High Level Nuclear Waste Disposal: Determination Using a Comparison of the EPA and NRC Regulations", V. M. Oversby, Lawrence Livermore National Laboratory, UCRL-94222, February 1986 [See Appendix A, p A-43].

"Test Plan For Series 3 NNWSI Spent Fuel Leaching/Dissolution Tests", C. N. Wilson, Hanford Engineering Development Laboratory, HEDL-7577, April 1986 [See Appendix A, p A-34].

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 and other radioactive components of the spent fuel are examined in these studies. Preliminary findings using fuel rods with holes or slits in the Zircaloy cladding and using bare fuel indicate that Tc and Cs are released preferentially over the actinides, and that Np is released congruently with U. Grain-boundary dissolution, leading to higher dissolution rates, was observed on specimens exposed to deionized water but not in those tested in J-13 water.

The consequence of changes in the oxidation state of uranium oxide are important and can be twofold. First, higher oxidation states may leach at a faster rate than UO_2 ; and second, as UO_2 oxidizes, it increases in volume. This swelling of the oxide places tensile loads on the cladding and may lead to a possible increase in cladding breaches. Measurements indicate that moisture content has only a minor effect on short-term oxidation rates, but there is some uncertainty whether radiolysis of water might cause fuel oxidation. There is evidence that, given enough time, even at room temperature, UO_2 will convert to UO_3 .

Data from the NRC regulations that govern the rate of release of radionuclides to the environment and EPA regulations that govern the allowable total release of the radioactive waste inventory of a generic repository show that solubility and sorption data are needed for americium and plutonium. Data are also needed for the actinides, and for carbon and nickel.

3.0 BWIP -- BASALT WASTE ISOLATION PROJECT

In our first biannual report in September 1986, the major deficiencies in the available data pertaining to a waste repository in basalt were noted. Briefly, these deficiencies fell into three major areas: (1) packing material breakdown (Is it stable, what are its thermal and hydraulic conductivities, what are its absorption characteristics, and will the 25% bentonite/75% basalt mixture be sufficient to seal off all the interstices between the particles?) (2) groundwater chemistry (Is it reducing or oxidizing?) (3) corrosion of the metallic waste container (What are the corrosion modes, how does radiation affect corrosion, how are short-term corrosion data going to be extrapolated to long times?). In the last five months, some information on these topics has been received from DOE and reviewed by NBS. The following discussion summarizes the main conclusions drawn from this data.

3.1 Groundwater Chemistry

No improvements have been made in determining the groundwater chemistry in the basalt repository site. However, [Gray 1984] found that, in the presence of alpha and gamma radiation, radiolysis of basalt groundwater containing methane (as found in the Umtanum flow region of the Grande Ronde basalt basin) resulted in the formation of high-molecular-weight organic substances, the identities of which are unknown. In subsurface oil field wastes, dissolved organic species can become complexing agents which can increase the solubility values of metals [Kharaka 1985]. Whether the unidentified organic substances found above in the irradiated basalt groundwater can serve a similar complexing role with the radionuclides in a repository in Grande Ronde basalt is not known. Further work in this area is required.

3.2 Packing Material Breakdown

Evidence is accumulating [Wood 1986; Barney 1986] that reduced actinide species have a strong affinity for basalt or bentonite. Therefore, when in contact with the basalt/bentonite packing material, the tendency is for the reduced actinides to become immobilized by adsorption, thereby reducing their rate of migration out of the repository site. The adsorption rates for ^{239}Pu , ^{240}Pu , ^{237}Np , ^{238}U , ^{76}Se , and ^{226}Ra were found to be quite high when in air at 90°C. However, ^{99}Tc was found to be an exception. For this material the basalt/bentonite mixture was not found to be an effective adsorber. Another possible problem with the packing material's filtering ability was found for reduced selenium (Se). Following

hydrothermal alteration (300°C for 950 hours) of the basalt/bentonite packing material, the adsorption of ^{75}Se onto the packing decreased significantly. It was suggested that this was caused by the measured loss of calcite during the thermal treatment. For the other radionuclides, no change in sorption or desorption characteristics was measured following the hydrothermal alteration.

Thermal calculations on the basalt repository have been updated [Yung 1986] using the most recent waste package and repository design. Temperature-time profiles were determined for various locations both within the repository and on the waste container. The maximum temperature of 260°C (at the vertical midpoint of the waste form) is expected to occur 10 years following emplacement. At the bottom of the waste container, the maximum surface temperature is expected to be only 200°C. Following this time, the latter temperature is expected to decrease to near 110°C in 10,000 years.

3.3 Corrosion of the Metallic Container

On the corrosion-behavior of the metallic waste-container, the available results of DOE work done to date on A27, A36, A387-9, and two low-alloy steels is summarized below. For ASTM A27 (AISI 1025) cast steel, some non-uniform corrosion behavior (after 20 months of attack) was observed in synthetic Grande Ronde basalt groundwater at 150 and 250°C [Westerman 1984]. Surprisingly, this steel exhibited much higher rates of general corrosion and pitting at 150°C when compared with that at 250°C. Weight losses for this material were found to have also increased an order of magnitude when the material and groundwater were subjected to a continuous dose of 3×10^5 rads per hour (rd/h) of gamma radiation [Westerman 1984]. Further evidence of the possibility of environmental interactions between this material and basalt groundwater was also suggested by a loss in ductility found in slow-strain-rate tests [Pitman 1985].

For ASTM A36 (AISI 1020) wrought steel, the 20-month corrosion tests [Westerman 1984] at 150 and 250°C resulted in the same observations as noted for the cast A27 steel above: the observed pitting, the higher general corrosion and pitting behavior at 150°C than at 250°C, and the order of magnitude increase in weight losses when subjected to gamma radiation. Despite the presence of localized corrosion in this material, environmentally-assisted cracking was not observed [James 1985] in fracture mechanics tests of this steel immersed in simulated Hanford groundwater at 250°C. In this latter study, no crack extension was observed in a precracked specimen after 2000 hours (~4 months) immersion. Further work needs to be

performed to determine (1) whether or not this wrought A36 steel possesses a long incubation time for crack extension, and (2) the limiting conditions for crack extension in A36 steel.

ASTM A387-9 steel (9Cr-1Mo) has been found to have a tendency for environmentally-assisted cracking [Pitman 1985; James 1985; James 1984; Duncan 1986]. For this wrought steel, slow-strain-rate tests [Pitman 1985] indicated a loss in ductility when in basalt groundwater. In addition, for a precracked specimen of this material stressed at 60 MPa/m, crack extension was suspected [James 1985] to have occurred when immersed in simulated Hanford groundwater at 250°C for 2000 hours. These two results indicate that a crack will propagate faster in this material immersed in Hanford groundwater than when surrounded by air. Since a "time to failure" cannot be calculated from these preliminary results, and because an environmental interaction obviously exists, the conservative approach to the selection of materials for HLW containment vessels would be to withdraw this material from consideration. If the material is to receive further consideration, much more work is required.

Two low-alloy cast steels, 1.2Cr-0.5Mo and 2.5Cr-1.0Mo, have been found to pit [Westerman 1984] when immersed in Grande Ronde basalt groundwater at both 150°C and 250°C. Their general corrosion rates also increased when subjected to gamma radiation, not unlike the behavior described above for the A27 and A36 steels. Further work needs to be performed on these materials to determine whether they are susceptible to environmentally-assisted cracking.

Thus, in basalt groundwater, pitting has been observed in two carbon steels, ASTM A27 and A36, as well as in two low-alloy cast steels tested, and the pitting was observed to be accelerated by gamma radiation. Other adverse effects of basalt groundwater have been indicated by a demonstrated sensitivity to basalt groundwater in slow-strain-rate tests of ASTM A27 steel and ASTM A387-9 steel.

4.0 SRP - - SALT REPOSITORY PROJECT

The Deaf Smith site is located 35 miles southwest of Amarillo in the north-central part of Deaf Smith County, Texas. The Battelle Office of Nuclear Waste Isolation (ONWI) will be purchasing land and office space in Deaf Smith County and should begin moving the Salt Repository

Project Office to Hereford Texas in March 1987. ONWI has a new contract with General Electric Company, Inc. to study welding, handling and emplacement technology.

4.1 Potential Problem Areas

On-Site Characterization: On-site sampling and characterization of the repository environment has not yet been performed. Long-term corrosion testing is required to get accurate estimates of performance. It is important to both characterize the salt environment thoroughly and begin long-term corrosion testing on the entire range of expected environments as soon as possible. Further, transport processes for brine in this bedded salt must be more thoroughly understood, so as to permit a more complete assessment of the quantity of brine to be expected in a repository at this site.

Container Alloys: At present, the primary candidate material for the waste container is A216 Steel. The reasons for this selection and the NBS reservations with it were discussed in the previous biannual report [Interrante 1986b]. In addition to those points, it appears that its weldments may be susceptible to hydrogen embrittlement [Cialone 1986]. This steel was selected to allow adoption of the uniform corrosion assumption and the elimination of other forms of corrosive attack. The assumption that uniform corrosion is the only form of attack is equivalent to assuming that surface films do not form. However, non-linear corrosion kinetics (decreasing corrosion rate with increasing time) indicate that surface films do form. Surface films can lead to localization of corrosion, e.g. pitting, crevice corrosion and/or stress corrosion cracking. Thus, it would be contradictory to apply non-linear corrosion kinetics and still assume that uniform corrosion is the only form of attack. The area of uniform corrosion requires further study. Localized corrosion, stress corrosion cracking and hydrogen embrittlement are potential failure modes that must be eliminated by mechanistic modeling based on experimental observations and not by assumption and by peer review.

The alternate alloy for container fabrication is Ti-code 12. This alloy was selected because of its corrosion resistance, which is the result of the formation of a passive film. Therefore, the uniform corrosion assumption cannot be applied to this alloy. Also, this alloy is susceptible to hydrogen embrittlement and radiation-induced hydrogen evolution must be considered.

As a result, it appears that the SRP program will have to identify and test other alternate candidate materials.

Perhaps, alloys that have greater thermodynamic immunity to corrosion and that do not passivate in the salt environment should be considered and studied. These materials would allow the uniform corrosion approach but would have a slower rate of uniform corrosion because of the reduced thermodynamic driving force. Considerable research has been performed on various alloys for geothermal applications, desalination plants and oil exploration. Examination of this literature should help identify additional candidates. Also, composite designs should be considered, as for example, those where the active and sacrificial metal is on the outside of the waste canister and a more noble alloy is on the inside. The outer layer would corrode consuming the water, oxygen, etc. in the environment while galvanically protecting the inner layer.

4.2 Reviews

Documents produced by the SRP have been collected and are being reevaluated. A document previously reviewed and those currently under review are identified below:

Documents Reviewed:

BMI/ONWI-Revision 1 (7/85), G. Jansen
"Expected Waste Package Performance for Nuclear Waste Repositories in Three Salt Formations"

Reviews in Progress:

BMI/ONWI-517 (2/86), Westinghouse
"Waste Package Reference Conceptual Designs for a Repository in Salt"

BMI/ONWI-592 (3/86), J. E. Balon
"ERG Review of Salt Constitutive Law, Salt Stress Determinations, and Salt Corrosion and Modeling Studies"

BMI/ONWI-612 (7/86), K. Vafai and Javad Ettetfagh
"The Effect of Stabilizers on the Heat Transfer Characteristics of a Nuclear Waste Canister"

BMI/ONWI-597
"Buckling Design Criteria for Waste Package Disposal Containers in Mined Salt Repositories"

BMI/ONWI-626 (11/86)
"ERG Review of the SRP Salt Irradiation Effects Program"

DOE/CH-21 (8/86)
"Systems Engineering Management Plan for the Salt Repository Project"

UCRL-53726 (10/86)

"Reference Waste Package Environment Report"

BNL 32001 (9/84), P. W. Levy and J. A. Kierstead
"Very Rough Preliminary Estimate of the Colloidal Sodium
Induced in Rock Salt by Radioactive Waste Canister
Radiation"

5.0 VITRIFICATION OF HIGH-LEVEL WASTE

The purpose of this project is to assist NRC in consulting with DOE on the acceptability of vitrified waste for disposal in a geologic repository. Two plants within the U.S. are currently involved in programs on HLW vitrification: the West Valley Demonstration Project (WVDP) for commercial HLW and the Defense Waste Processing Facility (DWPF) located at the Savannah River Plant for defense HLW.

The principal technical problems with which NBS is concerned are as follows: (1) whether the glass compositions that are being proposed will exhibit a leaching behavior in repository environments that will meet the NRC criteria for radionuclide release, (2) whether the facilities proposed by the WVDP and DWPF will produce a glass of the composition specified, and (3) how to verify the composition of the production glass product.

The NBS work related to high-level waste increased during the past few months. NBS participated in two meetings at which presentations were made giving some details and a brief review of the activities of the WVDP and DWPF.

A presentation of the DOE waste glass program involving both the WVDP and DWPF took place at the Forrestal Building, Washington, D.C., on December 8 and 9, 1986. Individuals from NRC, NBS, WVDP, DWPF and other organizations attended. The talks were of an overview nature and did not present sufficient technical detail on the leaching behavior of the waste glasses under investigation to allow determination of the efficacy of the approaches pursued by DOE. Future NBS reviews of the appropriate technical reports are expected to accomplish that end. Nevertheless, these meetings were very informative and did furnish sufficient details to accelerate the NBS studies aimed at understanding the DOE efforts in HLW vitrification.

A group from NBS (Dr. Melvin Linzer, Dr. Ernest Plante and Dr. Charles Interrante) and two of the three NBS consultants on glass technology (Mr. Bruce Adams and Dr. John Wasylyk) attended a briefing given by management and contractors of the WVDP in West Valley, NY, on March 18 and 19, 1987. At this meeting, additional details were provided on the leaching behavior of the class of borosilicate glasses under investigation for vitrification of HLW. After discussion between the DOE and NRC and their contractors, the following were determined to be among the unresolved licensing issues: How well will the glass withstand exposure to repository environments and how will the glass logs that do not meet specifications be stored? Moreover, DOE has not yet announced how much credit (for rate of release of radionuclides), if any, it will take for the glass.

In the near future, NBS staff will continue to identify and acquire documents relevant to the DOE waste glass program, and the most relevant documents will be identified and reviewed.

An effort was made to engage outside consultants who were expert in the field of glass leaching and yet met the strict NRC conflict-of-interest standards. Three individuals who satisfied both criteria were identified and placed on contract to NBS. These are Mr. J. Bruce Adams, Dr. John Wasylyk and Dr. David Cronin. Their initial task is to review the Pacific Northwest Laboratory (PNL) document, "Final Report of the Defense High-Level Waste Leaching Mechanisms Program," published in August, 1984 [PNL 1984]. This document is an extremely important summary of work in glass corrosion, and will be reviewed on a chapter by chapter basis. A review of Chapter 1, "Mechanisms of Defense Waste Glass Corrosion: Dissolution of Glass Matrix," is in preparation. Draft reviews of chapters 3, "Environmental Interactions," and 4, "Dissolution of Specific Radionuclides," will be available shortly.

6.0 MCC -- MATERIALS CHARACTERIZATION CENTER

This report on the Materials Characterization Center (MCC) covers activities of the MCC from May 1, 1986 through December 31, 1986. A description of the MCC and its objectives was given in the previous biannual report [Interrante 1986b]. The MCC is operated by the Department of Energy (DOE) to ensure that qualified materials data are available on waste materials. Methods of accomplishing this goal include developing standardized and approved test methods, testing materials using the approved methods, and

publishing procedures and data in the Nuclear Waste Materials Handbook. MCC is funded by DOE and the projects discussed in this report. The MCC distributes monthly reports covering a wide range of activities to at least thirty recipients, including the National Bureau of Standards.

A more detailed summary of MCC activities for the past eight months is given in Appendix B. The eight sections of the MCC reports are as follow.

- A. MCC Program Administration
- B. MCC Quality Assurance
- C. MCC Support to the Office of Geologic Repositories (RW-20) (OGR)
- D. MCC Support to the Salt Repository Project (SRP)
- E. MCC Support to the Basalt Waste Isolation Project (BWIP)
- F. MCC Support to Defense HLW Technology Program (DP-12)
- G. MCC Support to the Transportation Technology Center
- H. MCC Support to the West Valley Demonstration Project (WVDP)

The staff of the MCC consists of ten scientific professionals and sometimes some additional members. The task groups and leaders are shown in Figure 1. During this period, John E. Mendel left the position of Manager of the MCC and became chief scientist and also leader for task group 3, Spent Fuel Characterization. Max R. Kreiter became Manager of the MCC in September.

One of the goals of the MCC is to develop the Nuclear Waste Materials Handbook containing standardized and approved test methods and data. There are fourteen MCC test methods now in progress plus others in conjunction with the SRP and BWIP repositories. There is one approved reference material and five approved testing materials. These were listed in the previous report [Interrante 1986b]. These test methods and data must be approved by the Materials Review Board (MRB). During this eight-month period, the following test methods were sent to the MRB for approval; MCC-7P, MCC-1, and BWIP/MCC 105.1S Test Method. Data reports, MCC-D5 on one-year leach data of Approved Reference Material (ARM)-1 and MCC-6 on agitated powder leach testing of ARM-1 using MCC-3S were submitted to the MRB.

MCC interactions with the American Society for Testing and Materials (ASTM) continued and dealt with the ASTM committee C26 on Nuclear Waste and also with the development of accelerated testing procedures (C26.07).

The format of ASTM E632-82 was selected for use in the development of the accelerated testing document. Two meetings of the accelerated testing group were held, comments on the topic were solicited from the repositories.

Quality assurance coordination involved a number of activities including preparation and transfer of records through Pacific Northwest Laboratories (PNL) to BWIP. Plans were made to transfer records biennially to SRP. Many of these records dealt with ATMs, in this case, ATMs -9, -11 and -101, and these records were duplicated and copies are also kept at the MCC.

Plans were in progress to obtain spent fuel for ATM-107 and for transporting the material to Hanford, Washington. Characterization plans for ATMs-103 through 106 were agreed on by NNSWI, BWIP, ONWI and the Office of Crystalline Repository Development (OCD).

Fourteen SRP test methods were handled. Approval was requested to send three tests to the MRB for approval, and nine other tests were sent for SRP approval prior to being sent to the MRB. The SRP test, SRP-WPP-18/MCC-102, was balloted and approved by ASTM subcommittee C26.07.

In November, 1986, all experimental work on the development of the BWIP corrosion tests was stopped. Specimens were stored and records were sent to Rockwell Hanford Operations, RHO-BWIP.

Plans are to develop a comprehensive data base for the nuclear materials handbook. This work is supported by WVDP, OGR and Defense HLW Technology. Approximately 2000 citations have been found in a reference search on the subject of dissolution of nuclear waste forms.

Figure 2 and Tables 1 and 2 were taken from a paper written by G. B. Mellinger entitled "Standardized Test Methods for Use in Waste Compliance Testing Within the Department of Energy's Waste Acceptance Process". Table 1 shows the route of the waste compliance acceptance process and indicates the important role of test methods. Table 1 lists the advantages of using standardized test methods and indicated that peer review of standardized test development will be beneficial. Duplicate effort of parallel development of tests by different organizations can be eliminated with the use of standardized tests. The rigorous critical reviews and needed precision and accuracy statements of the standardization process can result in acceptable and useful test methods.

Figure 1. Organization Chart of the Materials Characterization Center.

Figure 2

Waste Compliance Portion of DOE Waste Acceptance Process

Table 1

Advantages of the Use of Standardized Tests in Waste Compliance Testing

- Elimination of parallel test development by different organizations (development of more than one test for a particular testing need)
- Extensive peer review of tests methods as a result of standardization process
- Waste management community consensus about utility of test methods
- Likely acceptance by the NRC of their use
- Users should gain sufficient experience with the tests such that reliable estimates of attainable precision and accuracy can be made

Table 2

Waste Acceptance Preliminary Specifications

WAPS Specification	Testing Requirements	Applicable Test Methods		
		Standardized Tests	Undergoing Standardization	Under Development
<u>Waste Form Specifications</u>				
Chemical Specification	Composition, Microstructure	MCC-7		MCC-13 MCC-17
Radionuclide Inventory Specification	Radiochemical Composition			MCC-17
Specification for Radionuclide Release Properties	Waste Form Chemical Durability	MCC-1 MCC-3	MCC-14.4	
Specification for Chemical and Phase Stability	Glass Transition Temperature, TTT Diagram	ASTM E 228-71 MCC-7		MCC-13
<u>Canister Specifications</u>				
Fabrication and Closure Specification	Canister Closure Leak Tightness	----- No Standard Test -----		
<u>Canistered Waste Form Specifications</u>				
Gas Specification	Gases Released as a Result of Heating of Waste Form	----- No Standard Test -----		
Specification for Removable Radioactivity on External Surfaces	Determination of Amount of Removable Contamination	----- No Standard Test -----		
Specification for Maximum Dose Rates	Maximum Canister-Surface Dose Rate			MCC-17
Chemical Compatibility Specification	Corrosion of Canister Interior		MCC-105.5	
Subcriticality Specification	Assure that Waste Form Will Remain Subcritical			MCC-17
Drop Test Specification	Assure that Canister Will Not Breach or Deform Excessively if Dropped	MCC-15		

7.0 DATABASE ACTIVITIES

A status report on the Waste Package Database and the Database for Reviews and Evaluations on High-Level Waste (HLW) is given in the following section.

7.1 High-Level Waste Database

After an analysis of the requirements for a computer assisted database for reviews and evaluations on HLW data was completed, the commercial software "Revelation" was chosen as the most compatible data base management system [Interrante 1986a]. The Waste Package Database developed using Revelation consists of three parts: the Reference file, the Review File, and the Automated Tracking System. The Reference File contains bibliographic information for pertinent technical reports, books, papers, and journal articles. The Review File contains reviews which have been approved by the Washington Editorial Review Board (WERB).

At the NBS, the database will also contain an internal Automated Tracking System which includes information on each review, whether in process or completed. This integrated system is designed so that five files are shared: (1) Reviews (REV) (2) Abstracts (ABS) (3) Figures (FIGS) (4) Key Words (KEY) and (5) References (REF). Because the five files are shared by the entire system, it is necessary to key in information only once. For example, the title of a report is keyed in, stored in the reference file, and is accessed from other files using a pointer to the title. This type of system design eliminates unnecessary information processing and storage.

Key words have been converted to files and these files have been entered into the database. The programming necessary to create the five primary files has been completed, the systems have been implemented on microcomputers and data entry has begun. In addition, various output formats have been developed for displaying the results of a search of the database.

At either a terminal screen or a printer, records may be displayed in many different formats. The names of report formats developed for the database are PACKIT, PACK, Formatted Text Fields (FTF), Superscripts and Subscripts in Fields (SUPER) and Superscripts and Subscripts in List (SUPERLIST). A sixth report format, LASER, also is planned. Included in Appendix C is a sample of the most concise report format, PACKIT. PACKIT condenses the information for each computer record and displays on one terminal screen as many records as possible. PACK is similar to PACKIT but includes an additional feature. When

a record is displayed on the terminal screen or printer using PACK, the field name is highlighted making it easier for the viewer to scan the information. Since there is no ASCII standard character for superscripts and subscripts, a program to enter and display superscripts, subscripts, and floating accents (umlaut and Angstrom) has been developed. In FTF each field is on a separate line and each record is separated. SUPER also separates each field and each record. In addition, SUPER allows the user to see the text in a more legible fashion. The output from the program SUPERLIST is similar to that of SUPER; however, each new record (review) begins on a separate page. An example of SUPERLIST is included in Appendix C.

The features of the report formats PACKIT, FTF, SUPER and SUPERLIST are not hardware dependent and, therefore, are available on any terminal screen. The highlight feature of PACK is hardware dependent and may not be available on every terminal or printer model. LASER is planned for future development. It will give report quality output on a laser jet printer.

Additional fields within the Reference File will provide specific information related to each report entered into the system. For example, fields have been developed to track readers assigned to a report. A field has also been developed to note the decision points necessary when assigning reports for review. Fields such as these will allow NBS personnel to record information related to each report entered into the Waste-Package Database. A copy of each report cited in the database Reference File will be available in the High-Level Waste Data Center. A laser jet printer has been received and interfaced with both Revelation and Word Perfect. It and a dot matrix printer are now operational in the data center.

One element of a completed NBS review of a HLW document is a list of keywords. Keywords are assigned by NBS reviewers and are used to conduct a rapid search of the database. Appendix C, pages C-5 to C-10, includes this NBS list of keywords in its current form. The list of keywords is organized according to the hierarchy shown in Appendix C, page C-4, the "Keyword Checklist Tree"; it contains four main categories and numerous subcategories of keywords. Keywords and categories will be revised in this Keyword Checklist, as needed. By having each reviewer carefully assign keywords for a document under review, the person searching the database can find relevant citations. For example, by combining the keywords cladding and spent fuel, the database user can quickly find reports which deal specifically with those two topics. If the searcher then adds in the term key data, the only citations selected will

be those containing key data (related to licensing) and cladding and spent fuel.

7.2 Database for Reviews and Evaluations on High-Level Waste (HLW) Data

The first magnetic disk containing the "Database for Reviews and Evaluations on HLW Data" was delivered to the NRC in March 1987. That database is a subset of the total NBS database, as it includes only the contents of each completed (and WERB-approved) NBS review. The storage medium for the database is one Bernoulli 20-megabyte disk. As additional reviews are completed, updated disks and new tapes will be delivered to the NRC. Documentation within the database will include a tutorial program which contains help screens and menus to train and assist the user in querying the database. These features will make the database easy to use, i.e., "user friendly".

7.3 Status of the Automated Tracking System

An automated tracking system has been developed for tracking each review as it passes through the review cycle at the NBS. For each review, this system will provide current status information. Bibliographic information from the main database is used in this tracking system, therefore, this information needs to be keyed into the total system only once. In addition to providing information on review status, the tracking system may be used for generating reports. The Automated Tracking System will be a valuable aid to managing the total review process.

7.4 Future Enhancements

A user's guide for searching the Database for Reviews and Evaluations on HLW Data will be written and provided to database users. This guide will introduce the database to the user, describe the file structure, and provide clear directions for searching and retrieving highly relevant citations.

A great deal of information is created and stored in computer-readable formats. Sometimes it is necessary to transfer information from one data entry format to another (e.g. Revelation, Word Perfect, MacWrite, MicroSoft Word). Commercially available software has been purchased and will be utilized to transfer encoded text into the database. The need to keyboard information into the database will be greatly decreased when this software is operational.

It is anticipated that selected Brookhaven (BNL) Reviews may be included in the Database for Reviews and Evaluations on HLW Data. NBS staff would identify the most relevant reviews, e.g. Brewster 1985 and Sullivan 1985, and begin magnetic media encoding of them. Magnetic media encoding would facilitate merging selected Brookhaven reviews in the most cost-effective manner.

User menus and system routines will be improved in future upgrades of the system. Currently, the Text Control Language (TCL) of the database management system is used to search and retrieve information and perform other functions.

In conducting a query of the database using TCL in the present configuration, key fields and keyword lists are used to provide rapid searches of the database. The text fields of the database may also be searched for the presence of any alphanumeric string. While TCL must be used at present for making these types of queries, menus will replace the TCL in the future. Menus will be developed to make searching easier for the user.

System routines can be used to conduct daily/repetitive operations, within the NBS HLW program, in a manner that is more cost effective. Those planned for the immediate future include the following: (1) Routines for mapping (converting) data written by reviewers, using their own word processor formats, into the form required by the database management system used for the HLW database; (2) financial status reports needed by managers of the program; (3) routines that help to track the reviews in progress, for example, by furnishing lists of activities that are taking more time than anticipated by management.

ACKNOWLEDGEMENTS

The editor wishes to thank the following staff members for their support in preparing this biannual report.

Joyce F. Harris, typed, proofread, and coordinated reviews of this work.

Karen A. Heilers assisted with typing.

Debra Lohr assisted with typing.

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WASTE PACKAGE DATA REVIEW FORM GUIDELINES

DATA SOURCE

Full document reference. This section will be completed for the reviewer before he/she receives the document.

DATE REVIEWED

The date the document review was completed.

TYPE OF DATA

- (1) Scope of the Report:
e.g., Experimental, Theoretical, Literature Review, Data Analysis
- (2) Failure Mode or Phenomenon Studied:
e.g., Corrosion, Creep, Fatigue, Leaching, Pitting, Hydrogen Embrittlement, Debonding, Dealloying

MATERIALS/COMPONENTS

Description of the material studied (and/or component part, if specifically addressed as such; e.g., the screw-on type cap on a waste cylinder): e.g., 304L Stainless Steel, Brass, Zircaloy Cladding, Welds in 316 Stainless Steel, Packing Material, Basalt.

TEST CONDITIONS

- (1) the State of the material being tested:
e.g., Cold Worked or Annealed 304L Stainless Steel, Thermo-mechanical History of the material (or component) studied
- (2) Specimen Preparation:
e.g., prestressed, precracked, size, type of specimen
- (3) Environment, pressures, and other test parameters of the material being tested: e.g., Aqueous environment, Radioactive surrounding, Electrolytes or corrosive agents present, Temperature and pressure (externally applied or not) during the test

METHODS OF DATA COLLECTION/ANALYSIS

Includes Data Measurement Methods and Types of data measured, as well as Data Analysis Techniques, e.g.:

Electron microscopy, weight loss vs. time, slow strain rate tensile test, x-ray diffraction, differential thermal analysis, A.C. electrical resistivity using a Wheatstone bridge, mass spectroscopic chemical analysis of the corrosive environment, Latin Hypercube method, Monte Carlo techniques

AMOUNT OF DATA

Includes the number of tables and graphs of data together with their titles and axes (indicating the range in values), e.g.:

5 tables of temperature and time data for five molten glass pouring operations, each table including the data from ten sensor locations. The temperatures ranged from 1100°C to 0°C over a time period of 24 hours.

UNCERTAINTIES IN DATA

Included here are error bars and uncertainties in the data as stated by the author. This also includes qualitative statements by the author on the reliability of the data, e.g.:

Temperatures carry an accuracy of +5°C while the times are reported to within +15 sec. It was felt that under real glass pouring operations (without well controlled crucible cooling) the temperature-time curves will be shifted to somewhat higher temperatures than shown here.

DEFICIENCIES/LIMITATIONS IN DATABASE

Includes statements by the author on the applicability of the data, e.g.:

Extrapolation of the temperature-time (time < 24 hrs) data presented here to times in excess of 100 years should not be performed. The data presented here is useful only for indicating trends and qualitative parameter relationships, not for the purpose of presenting absolute values.

KEY WORDS

These are to be checked off on the keyword checklist. In general, these keywords should reflect the information given in the above categories. Additional keywords, which are truly different from terms on this list, should be added to the list as necessary.

GENERAL COMMENTS

The reviewer's general comments on the document. This category is wide open as far as content. It contains information the reviewer did not enter into any of the above categories, but which is considered important for the reader to know. It would also be in this section that the reviewer would put any of his comments on the deficiencies and uncertainties in the data and analysis. e.g.:

This is a very comprehensive review of the literature on the temperature sensitization of stainless steels. Even though it neglects the definitive work of Bertocci, Shull, Kaufman, and Escalante [Phys. Rev. J13, (1879), pp. 15-358] in this area (presumably because of the difficulty in locating this document), this review still considers a sufficiently large number of other investigations to provide a good understanding of the present status of the field. The one discordant note here, however, is that it would have been a much more useful review if stainless steel types 301, 303, 304, 316, and 440C had also been addressed.

Abstracts taken from paper will be available in the database. References to the abstract may be made.

RELATED HLW REPORTS

The report number(s) of any report(s) known to be directly related to the report being reviewed should be entered here so that these reports may be cross-references in the database.

The reviewer might also indicate any other reports taken from the reference list (in the report being reviewed) that might be useful to acquire and include in the database.

APPLICABILITY OF DATA TO LICENSING -- READ, BUT DO NOT
COMPLETE THIS SECTION, NOT TO BE FILLED IN BY THE REVIEWER

Indicated here is the licensing issue addressed by this paper. It is either (a) a specific Listed licensing Issue in an NRC Site Characterization Plan (ISTP) or (b) a new issue not yet identified in an ISTP.

The ranking of the paper is determined as follows: The "Key Data" box is marked if the paper contains data that is of sufficient quality that it must be considered by NRC in an evaluation of a license application. Such a paper must meet at least one of the following criteria: (1) it is an in-depth review of the pertinent literature, (2) it contains data that is found to be especially significant after being assessed for scientific merit and quality, or (3) it contains data with such a small uncertainty that it must be considered in a performance evaluation of a license application. If the paper does not meet any of the above three criteria, it is indicated as "Supporting Data". Reviewer's comments on the listing of the document may be included with the appropriate Issue Listing in subcategory (a) or (b).

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data:

Westinghouse Hanford Company, Richland, Washington
99352

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

Einziger, R. E., and Woodley, R. E., "Low Temperature Spent Fuel Oxidation Under Tuff Repository Conditions," HEDL-SA-3271FP, March 1985.

DATE REVIEWED: 11/29/86

TYPE OF DATA

Experimental rate of oxidation of spent fuel (UO_2).
Theoretical analysis to determine whether mechanism of oxidation is by grain boundary or bulk diffusion.

MATERIALS/COMPONENTS

PWR spent fuel (UO_2) from PWR Turkey Point Unit 3.

TEST CONDITIONS

PWR (UO_2) spent fuel in moist air (16000 ppm) and dry air (3 ppm) using fragments or pulverized fuel at temperatures of 200-225°C up to 737 hours.

METHODS OF DATA COLLECTION/ANALYSIS

Oxidation rate by weight gain vs time at fixed temperature and atmospheric pressure in air. Thermogravimetric and ceramographic analysis. (Ceramographic analysis involves study of micrographs of non-metallic materials).

AMOUNT OF DATA

7 figures and 4 tables. Figures; Binary phase diagram of (oxygen/uranium) from O/U molar ratio of 2 to 3 and temperatures from 0 to 1000°C., Schematic of oxidation rate vs $1/T$ when oxidation mechanism changes at temperature below experimental range., Schematic of Thermogravimetric Analysis (TGA) system., TGA test sample weight changes, percent weight change and change in $^{16}\text{O}/\text{Metal}$ or $^{18}\text{O}/\text{M}$ vs time in hours., 2 figures of polished cross sections of UO_2 , Plot of $[1-(1-C)^{1/3}]$ vs

$t^{1/2}$ showing that (bulk) diffusion limits the oxidation rate. Tables; Free energy change for oxidation of UO_2 to U_3O_8 and U_3O_8 to UO_3 , Characteristics of Turkey Point Unit 3 fuel. Experimental test parameters., Results of preliminary oxidation tests.

UNCERTAINTIES IN DATA

Not quantitatively discussed.

DEFICIENCIES/LIMITATIONS IN DATABASE

The authors state that the conclusions are preliminary. Additional testing for longer times and at lower temperatures are needed before spent fuel oxidation mechanisms are established and extrapolation of oxidation behavior in a tuff repository can be made.

KEY WORDS

Experimental data, TGA, Water vapor, Spent fuel, Oxidation rate, Degradation (spent fuel), Mechanism of oxidation.

GENERAL COMMENTS

This work demonstrates that oxidation of spent fuel in breached fuel canisters may take place but the authors state that the test temperatures are somewhat higher than the estimated repository temperatures. Literature estimates put the repository at 145-160°C after 300 years. Although the work is preliminary, there is no error analysis of the temperature dependence of the activation energy. The air flow rate of 400 cc/min used in the measurements is probably much higher than would exist in a repository. It may be possible that in the repository the rate of arrival of oxygen at the surface could be the rate determining step in the oxidation process. If the leach rate of higher uranium oxides is an important consideration, measurements of this leach rate should be obtained.

The authors conclude that "(1) The TGA system appears to have sufficient stability to provide excellent oxidation data. The small samples appear to adequately represent the fuel being studied. (2) The size of the fuel sample does not appear to influence its long-term oxidation rates. It appears that grain boundary diffusion allows rapid access to the oxidizing atmosphere to most of the grains making up the fuel specimen. This rapidly increases the surface area available for bulk diffusion and oxidation. (3) Over a range of 3 to 16,000 ppm of H_2O , the moisture content of the air has only a minor effect on the short-term oxidation rate, which may decrease with increasing moisture content."

APPLICABILITY OF DATA TO LICENSING

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

- (a) Relationship to Waste Package Performance Issues Already Identified.

Related to issue 2.3.6 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project. This issue involves potential damage and failure mechanisms for fuel rod cladding. Oxidation of spent fuel through pinholes or other fissures in the cladding is a potential failure mechanism.

- (b) New Licensing Issues
- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Report

Brookhaven National Laboratory, Department of Nuclear Energy, Upton, New York 11973

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

T. Abraham, H. Jain, and P. Soo, "Stress Corrosion Cracking Tests on High-Level-Waste Container Materials in Simulated Tuff Repository Environments," NUREG/CR-4619, BNL-NUREG-51996, June 1986.

DATE REVIEWED: 1/13/87

TYPE OF DATA

(1) Scope of the Report

This report contains the procedures for and results of stress corrosion cracking tests as well as a brief literature survey of stress-corrosion behavior of candidate container materials.

(2) Failure Mode or Phenomenon Studied

Stress corrosion cracking susceptibility.

MATERIALS/COMPONENTS

Types 304L, 316L, and 321 austenitic stainless steels and Incoloy 825. All of these materials are under consideration for use as container materials.

Compositions given by the vendors (in weight percent, bal. = balance):

Alloy	C	Mn	Si	P	S	Cr	Ni	Mo	Ti	Al	Fe
304L SS	0.016	1.95	0.48	0.038	0.025	18.54	10.55				bal.
316L SS	0.016	1.94	0.37	0.035	0.010	16.66	12.80	2.02			bal.
321 SS	0.028	1.03	0.74	0.026	0.002	17.41	10.75		0.24		bal.
Incoloy 825	0.020	0.44	0.27		0.001	22.34	44.14	2.78	0.84	0.07	27.26

Mechanical properties given by the vendors for as-received alloys:

Alloy	Ultimate		0.2% Offset		Elongation	Rockwell
	Tensile Strength		Yield Strength			
	(MPa)	(ksi)	(MPa)	(ksi)	(%)	(R _B)
304L SS	536	77.8	276	40.1	65.3	85.2
316L SS	562	81.5	254	36.9	59.2	69.4
321 SS	558	81.0	225	32.7	65.0	78.0
Incoloy 825	799	116.0	475	68.9	36.0	95.0

TEST CONDITIONS

(1) State of the Material being Tested

Specimens of the as-received (solution annealed) alloys and heat-treated (sensitized) material were tested. The heat treatment consisted of maintaining the tubing at 600°C for 100 hour and then furnace cooling. The treatment was expected to sensitize the alloys by forming chromium-depleted zones adjacent to grain boundary carbides. Micrographs show that the treatment probably sensitized the steels by precipitating carbides at grain boundaries. In Incoloy 825 the treatment resulted in the precipitation of second-phase particles (probably TiC) within grains. Sensitization in the steels is slightly stronger near the surfaces of the specimens, most likely due to unavoidable contamination from lubricants during mill operations.

(2) Specimen Preparation

C-ring specimens were fabricated from seamless 0.75-inch o.d. x 0.125-inch wall tubing of the stainless steels and 0.84-inch o.d. x 0.109-inch wall tubing of Incoloy 825. Testing was done according to ASTM procedures (1979). Tubing surfaces were polished with 600-grit SiC paper. Notched C-ring specimens were machined from the polished tubing. Before stressing, each specimen was degreased in trichloroethane and then cleaned with Alconox soap, methanol, and distilled water.

(3) Environment of the Material being Tested

Specimens were tested in boiling synthetic J-13 water and a ten-times concentrated J-13 water in the presence of crushed Topopah Spring tuff. The concentrated solution (arbitrarily chosen as 10 times the J-13 water analysis) represents the situation when salts, precipitated during initial evaporation of groundwater, are redissolved as new cooler water subsequently percolates towards the repository horizon. Specimens were also tested in the steam phase of the two solutions.

METHODS OF DATA COLLECTION/ANALYSIS

C-ring specimens were loaded so that stress levels at the apex were 90 percent of the elastic limit at ambient temperature calculated for the unnotched condition. Therefore, one smooth C-ring for each of the eight test alloys (four as-received and four sensitized specimens) was tested on an Instron machine to obtain stress-strain characteristics. The C-ring was stressed in the same orientation as that expected from the usual nut and bolt arrangement.

Notched C-ring test specimens were stressed using commercial stainless steel nuts and bolts so that the change in o.d. was 90 percent of the deflection at the elastic limit. In the procedure used, the applied tensile stress in most of the volume of a specimen will be within the elastic limit, which could correspond to those stresses expected in a normal waste container. However, the actual stress in the cross section of a test specimen through the notch will be higher than the "90 percent of elastic limit" value because of stress concentration effects.

To determine the strain present on the surface of a smooth C-ring, three longitudinal and one transverse strain gauges were attached on the outer surface and the specimen was then compressed in an electro-mechanical test machine.

Tests were conducted in six test vessels each corresponding to six different test conditions. In each unit, specimens along with crushed Topopah tuff rock and test solution were stacked in a 4-liter Pyrex vessel. About 6/7 of the vessel height was enclosed within an electric furnace. The lid of the vessel has space for four connections, two of which were kept closed, and two of which contained a thermometer and a cold water condenser. Flowing cold water condensed steam to water which dripped back into the vessel. The top of the condenser was open to the atmosphere so that some air was present during the tests.

Triplicate samples were used for each alloy under each test condition. Specimens were stacked within the crushed tuff in two layers at two levels in the test vessels. Samples at the lower level were submerged in solution and those at the upper level were surrounded by tuff, air, and steam. The temperature was kept at the boiling point during tests and fluid samples were taken periodically to test for the concentration of various ions, the pH, and oxygen content. The test matrix provided for testing the four alloys in both as-received and sensitized conditions, in J-13 water and steam above it, and in the ten-times concentrated J-13 water and the steam above it. All of these combinations were tested for 3-, 6-, and 12-month test periods.

At the end of the test period, solution was withdrawn for analysis. The test vessel was cooled to ambient temperature and the C-ring specimens removed and examined visually for cracks. Representative specimens were cleaned ultrasonically in distilled water and examined under an optical or scanning electron microscope. Cross sections cut perpendicularly to the notch through the center of the bolt holes were polished and examined for microscopic cracks.

AMOUNT OF DATA

There are 20 tables in the report. The first seven contain information from the literature survey.

Table 1--"Candidate metals for overpacks" lists commercial alloys with the chemical compositions.

Table 2--"SCC test in boiling deionized water (first test)" lists the maximum depth (0.0 to >2.0 mm) of stress-corrosion cracks in double-U-bend samples, both plain and v-notched, of eight alloys. The tests were made for varying exposure times (7 to 180 days) in both non-irradiated and in gamma-irradiated conditions.

Table 3--"Chemical composition of 316ELC specimens used (wt %)."

Table 4--"Chemical composition of steels used for SCC testing," lists compositions for 22 steels.

Table 5--"Slow-strain-rate test results for 304L stainless steel in 150°C tuff-conditioned J-13 water," lists elongation (48.0 to 54.0 %) and yield stress (25.8 to 29.6 ksi) results of tests for two metallurgical conditions, two environments, and two strain rates.

Table 6--"Stress corrosion cracking test results from U-bend specimens exposed to irradiated J-13 water, crushed tuff rock, and water vapor. Results after 3 months exposure." The ratios of the no. of specimens cracked/no. of specimens tested for two steels at two temperatures, and two radiation dose rates are listed.

Table 7--"Status of stress corrosion cracking test results for four-point load, bent-beam specimens exposed to J-13 water and steam and stressed to 90 percent yield stress." The ratio of the no. of specimens cracked/no. of specimens tested for four steels each tested under two process conditions and two exposure times are listed.

The rest of the tables pertain to the experimental tests performed and given in this report.

Table 8--"Reference groundwater composition for tuff repositories based on composition of water in Jackass Flats Well J-13 at the Nevada Test Site," includes the pH as well as the concentration (mg/l) of the ionic constituents.

Table 9--"Test matrix for stress corrosion of candidate stainless steel and Incoloy 825."

Table 10--"Vendor supplied chemical analysis of test alloys (weight percent)," gives the composition for the four alloys to be tested.

Table 11--"Mechanical properties of as-received test alloys (vendor supplied data)," lists ultimate tensile strength and 0.2 percent offset yield strength in both MPa and ksi, elongation (%), and Rockwell Hardness (R_B).

Table 12--"Stress-strain properties of eight alloys as obtained from Instron machine tests on smooth C-ring specimens." Data are given for four alloys tested in both as-received condition and sensitized. The data in the table are Load at 100 percent Elastic Limit (kg), Deflection at 100% Elastic Limit (cm), Deflection at 90 percent Elastic Limit (cm), and Longitudinal Strain at Elastic Limit (10^{-6}).

Table 13--"Geochemical analysis of Topopah Spring tuff (values in ppm unless percent is specified.)"

Table 14--"Amount of chemical compounds used in preparing 20 liters of synthetic J-13 water (mg)," lists the weight (mg) of the 14 chemicals used.

The following four tables list test results both as the ratio of specimens with microcracks/number of specimens examined and the maximum crack length (micrometers) for both as-received and sensitized alloys. The data are for specimens tested for three time periods, in steam tests and water tests using both J-13 composition water and water with a composition 10 times that of J-13 water.

Table 15--"Stress corrosion cracking tests for Type 304L stainless steel."

Table 16--"Stress corrosion cracking tests for Type 316L stainless steel."

Table 17--"Stress corrosion cracking tests for Type 321 stainless steel."

Table 18--"Stress corrosion cracking tests for Incoloy 825."

Table 19--"Composition of J-13 and 10xJ-13 solution during stress corrosion experiments," lists the pH and the amount [apparently mg/l] of nine constituents of solutions sampled during the stress corrosion tests.

Table 20--"Chemical composition of test solutions at the end of corrosion tests (micrograms/mL) (undiluted and filtered solution)," lists the pH and the amount for the same nine constituents as in Table 19.

There are 46 figures in the report. The first nine figures contain information from the literature survey.

Figure 1--"Proposed relationship between chloride and oxygen content of alkaline phosphate treated boiler water and susceptibility to stress corrosion cracking of austenitic stainless steel exposed to the steam phase with intermittent wetting." The data are plotted for oxygen

versus chloride in ppm on $\log_{10}$ scales from 0.1 to 1000. Figure 2--"The relationship between original strain in bent-beam specimen and time-to-crack for (a) Type 321 stainless steel at 300°C and (b) Type 304 stainless steel at 200°C." For both figures, original strain (0 to 2×10^{-3}) is plotted versus time in hours (1 to 1000). Data for pickled and electropolished specimens both cracked and uncracked are plotted.

Figure 3(a)--"Schematic diagram of double U-bend type specimen."

Figure 3(b)--"Schematic diagram of gamma-ray irradiation (first test)."

Figure 4--"Microstructure of cracks," shows micrographs of 304, 316, 316L, and 316ELC stainless steels.

Figure 5--"Time-temperature sensitization curve for Type 316ELC for tests conducted in pure water at 250°C," contains data for cracks less than or equal to 50 micrometers and less than or equal to 20 micrometers at temperatures in °C (550 to 800) versus time in minutes (5 to about 4000 on a $\log_{10}$ scale).

Figure 6--"Effect of carbon on SCC of Type 316SS in high temperature pure water at 250°C." Maximum Crack Depth (micrometers) is plotted versus Carbon content (%).

Figure 7--"Comparison of SCC susceptibility on several representative austenitic stainless steels sensitized at 620°C for 24 hours." Maximum crack depth in mm (0 to 2.0) is plotted against test temperature in °C (100 to 300).

Figure 8--"Effect of chloride added as NaCl on cracking of Type 304 stainless steel at 100°C. Solution transported by porous material to specimen." Data are plotted for three chloride concentrations as cumulative percent of cracked specimens (0 to 100) versus test time in hours (10 to 10,000 on a $\log_{10}$ scale).

Figure 9--"Percentage of specimens unbroken plotted against exposure time for various commercial alloys exposed to a boiling magnesium chloride solution at 154°C. Specimens stressed to 90 percent of the yield stress." Data for five alloys is plotted as percent specimens unbroken (10 to 100) versus exposure time in minutes (100 to 100,000 on a $\log_{10}$ scale).

The rest of the figures pertain to the tests performed and to the results of the tests. Three figures are either diagrams or pictures of apparatus or specimens.

Figure 11--"C-ring specimen design."

Figure 14--"C-ring stress corrosion cracking test apparatus."

Figure 15--"Schematic of the specimen arrangement in a test vessel."

Four figures contain test results.

Figure 12--"Load vs deflection of Type 304L stainless steel sensitized C-ring test specimen under stress." Load in lbs. (0 to 400) is plotted versus deflection in inches (0 to 0.03).

Figure 13--"Load vs in-situ strain at the apex of a Type 316L stainless steel C-ring specimen." Load in lbs. (0 to 350) is plotted versus strain in inch/inch (0 to 0.004) for four gages, one transverse and three longitudinal.

Figure 45--"Concentration changes in J-13 water during reaction with crushed tuff at 100°C." The concentration of seven ions in micrograms/milliliter (0 to 900) is plotted versus the test time in months (0 to 12).

Figure 46--"Concentration changes in 10xJ-13 water during reaction with crushed tuff at 100°C." The concentration of seven ions in micrograms/milliliter (0 to 1400) is plotted versus the test time in months (0 to 12).

There are 29 figures which are micrographs of alloy specimens.

Figure 10--"Microstructure of Type 304L stainless steel. (a) As-received, etched in glyceric acid. (b) Sensitized at 600°C for 100 hours, etched in oxalic acid."

Figure 16--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 304L stainless steel specimen tested in steam/air phase above boiling J-13 groundwater for 3 months (x2000)."

Figure 17--"An SEM micrograph of part of the outer diameter of a sensitized Type 304L stainless steel specimen tested in steam/air phase above boiling 10xJ-13 solution for 6 months (x2000)."

Figure 18--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 304L stainless steel specimen tested in boiling 10xJ-13 solution for six months (x2000)."

Figure 19--"An SEM micrograph of part of the notch root of a sensitized Type 304L stainless steel specimen tested in boiling 10xJ-13 solution for six months (x750)."

Figure 20--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 304L stainless steel specimen tested in boiling J-13 solution for six months (x2000)."

Figure 21--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 304L stainless steel specimen tested in boiling J-13 solution for twelve months (x3000)."

Figure 22--"An SEM micrograph of part of the outer diameter of a sensitized Type 304L stainless steel specimen tested in steam/air phase above boiling 10xJ-13 solution for twelve months (x2000)."

Figure 23--"An SEM micrograph of part of the outer diameter of an as-received, untested Type 316L stainless steel specimen (x1000)."

Figure 24--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 316L stainless steel specimen tested in boiling J-13 water for three months (x2000)."

Figure 25--"An SEM micrograph of part of the outer diameter of a sensitized Type 316L stainless steel specimen tested in steam/air phase above boiling 10xJ-13 water for three months (x2000)."

Figure 26--"An SEM micrograph of part of the notch root of an as-received (solution annealed) Type 316L stainless steel specimen tested in boiling J-13 water for six months (x2000)."

Figure 27--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 316L stainless steel specimen tested in boiling J-13 water for six months (x2000)."

Figure 28--"An SEM micrograph of part of the notch root of a sensitized Type 316L stainless steel specimen tested in boiling J-13 water for six months (x2000)."

Figure 29--"An SEM micrograph of part of the notch root of a sensitized Type 316L stainless steel specimen tested in the steam/air phase above boiling J-13 water for six months (x1000)."

Figure 30--"An SEM micrograph of part of the notch root of an as-received (solution annealed) Type 316L stainless steel specimen tested in boiling 10xJ-13 solution for six months (x2000)."

Figure 31--"An SEM micrograph of part of the notch root of an as-received (solution annealed) Type 316L stainless steel specimen tested in boiling 10xJ-13 solution for six months (x2000)."

Figure 32--"An SEM micrograph of part of the notch root of a sensitized Type 316L stainless steel tested in boiling 10xJ-13 solution for six months (x3000)."

Figure 33--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 316L stainless steel tested in boiling J-13 water for twelve months (x2000)."

Figure 34--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 316L stainless steel tested in boiling J-13 water for twelve months (x2000)."

Figure 35--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 316L stainless steel tested in steam/air phase above boiling 10xJ-13 solution for twelve months (x2000)."

Figure 36--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 321 stainless steel tested in steam/air phase above boiling 10xJ-13 water for six months (x2000)."

Figure 37--"An SEM micrograph of part of the outer diameter of an as-received (solution annealed) Type 321 stainless steel specimen tested in boiling 10xJ-13 solution for six months (x100). X-ray line profiles for Fe, Cr, and O are shown."

Figure 38--"An SEM micrograph of part of the outer diameter of a sensitized Type 321 stainless steel specimen tested in boiling 10xJ-13 solution for one year (x1500)."

Figure 39--"An SEM micrograph of part of the notch root of an as-received (solution annealed) Incoloy 825 specimen tested in steam/air phase above boiling 10xJ-13 water for six months (x700)."

Figure 40--"An SEM micrograph of part of the notch root of an as-received (solution annealed) Incoloy 825 specimen tested in steam/air phase above boiling 10xJ-13 water for six months (x2000). X-ray line profiles for O, Cr, Ni and Fe are shown."

Figure 42--"An SEM micrograph of the surface of a sensitized Type 304L stainless steel specimen showing black spots. The specimen was exposed to J-13 groundwater for three months. (x1000)"

Figure 43--"Optical micrograph of the surface of a sensitized Type 316L stainless steel specimen tested in 10xJ-13 water for three months. The dark circle shows the area from where a salt layer has been removed mechanically. The boundary inside the circle corresponds to the area where a deposit predominantly containing Si, Ca, Al was present (x500)."

Figure 44--"An SEM micrograph of the white compound found on the inside of the lid of the test vessel (x1000)."

One figure shows the results of energy-dispersive x-ray analysis.

Figure 41--"(a) An EDAX analysis of the crack tip shown in figure 40." "(b) An EDAX analysis of the specimen matrix material shown in Figure 40." "(c) An EDAX analysis of the mounting material shown in Figure 40."

DEFICIENCIES/LIMITATIONS IN DATABASE

The results show that surface contamination by lubricants of a stainless steel container during production and handling could enhance susceptibility to sensitization and, therefore, stress corrosion cracking.

Note that although the carbon contents of the alloys are within specifications, they are all at the low end of the permissible range. Because of this low content, these alloys are less likely to become sensitized compared with steels with higher carbon content.

The objectives of the tests were to determine if microcracking was present after exposure and to determine the factors governing initiation of cracking, and the type

of crack (intergranular, transgranular, or mixed). Determining the type proved very difficult in most cases because the sensitizing heat treatment may not have led to significant precipitation of chromium carbide at the boundary (due to the exceptionally low carbon content). The low carbon of the 304L and 316L steels made it difficult to etch the boundaries and determine whether the microcracks followed the paths. The low carbon of the 321 and the Incoloy 825, coupled with their tendency to form intergranular titanium carbides rather than intergranular chromium carbide, would tend to minimize both the sensitization and grain boundary etching.

KEY WORDS

Experimental data, Microscopy, Visual examination, Laboratory, Air, J-13 water, Tuff composition, Cl, Tuff, High temperature, Ambient pressure, Static (no flow), Nickel base, Stainless steel, 304L stainless steel, 316L stainless steel, 321 stainless steel, High-nickel alloy 825, Annealed, Sensitized, Bolt or wedge loading, Prestressed (before exposure), C-ring, Corrosion (stress cracking) SCC.

GENERAL COMMENTS

The literature review of SCC is good but not exhaustive. In the literature review the authors criticize an earlier literature survey of candidate container materials as giving insufficient attention to stress corrosion cracking and, as the brief survey in this report indicates, there is a strong potential for stress corrosion cracking under Tuff repository conditions.

The carbon contents of the alloys tested were at the low end of the permissible range for each alloy. Samples of the same alloy which contain more carbon, but still in the permissible range, would be more susceptible to sensitization.

Since the tests were performed with standard ASTM v-notched C-ring specimens with no pre-cracking, the time of crack incubation and initiation as well as for crack propagation is part of the total exposure period.

Because the carbon content of the alloys tested was at the low end of the permissible range for each alloy, the "sensitized" samples tested were not sufficiently sensitized to exhibit extensive intergranular attack during metallographic etching.

Cracks were found in samples of all alloys tested that were examined. These cracks were not always present at the notch tip and the role of stress is uncertain. That is, distinction between intergranular corrosion and/or SCC is uncertain.

Not all cracks were intergranular and branching was observed. These conditions caused confusion about the relative role of stress and the distinction between SCC and IG-corrosion.

The environment changed during the course of the test (and was monitored) and as a result the specific conditions which caused cracking are uncertain.

The authors have clearly achieved their objective of demonstrating that SCC in the Tuff repository environment should be a subject of container selection criteria.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues
Already Identified

Related to issues 2.2.4, potential corrosion failure modes for the waste package container, and 2.2.4.2, the effects of radiation on the corrosion failure modes and associated corrosion rates for the waste package container, in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Report

Los Alamos National Laboratory, Los Alamos, New Mexico

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

Ogard, A.E. and Kerrisk, J.F. "Groundwater Chemistry Along Flow Paths Between a Proposed Repository Site and the Accessible Environment," LA-10188-MS, November 1984.

DATE REVIEWED: 10/24/86

TYPE OF DATA

1. Scope of the Report

Chemical analysis of groundwater sampled around the Yucca Mountain, Nevada, proposed repository site. The site is the Topopah Spring Member tuff in the unsaturated zone of Yucca Mountain.

2. Failure Mode or Phenomenon Studied

The speciation and solubility of nuclear waste elements in the groundwaters was calculated using the analysis of the groundwater chemistry along potential flow paths from the proposed repository.

MATERIALS/COMPONENTS

The groundwater in selected Yucca Mountain, Nevada, sites was sampled and the chemistry studied.

TEST CONDITIONS

1. State of the Material being Tested

The groundwater was obtained from wells drilled by the U.S. Geological Survey into the saturated zone around the proposed site.

2. Specimen Preparation

Samples of groundwater were obtained in three different ways. (1) Samples were taken aerobically and sometimes anaerobically during the USGS's pumping tests. Integral samples were obtained because all producing zones, at all depths, contributed to the groundwater sampled.

(2) Individual permeable zones were isolated by inflatable packers and the water from between these packers was pumped and sampled. These tests provide the best information on the change in groundwater composition with depth because the isolation of individual permeable zones yields water from a particular depth rather than integral samples from all depths. Values of Eh measured from such wells also provide the best estimates of water Eh at depth, because measurements were made on water from an isolated zone and without exposing samples to air.

(3) Individual samples were taken from selected depths in static holes by 1) lowering an evacuated stainless steel bottle to a selected depth, 2) opening the valve electrically to allow the bottle to fill, 3) closing the valve, and 4) raising the bottle to the surface. Such samples are designated as "thief" samples.

3. Environment of the Material being Tested

Deep wells were drilled and tapped at several Yucca Mountain sites, and also at Pahute Mesa and in the Amargosa Desert.

METHODS OF DATA COLLECTION/ANALYSIS

Composition of the groundwater was determined by analysis for dissolved cations and anions, by electrode measurements for Eh, pH, sulfide, and dissolved oxygen, by alkalinity titrations, and by analysis for detergents.

Cation concentrations (for Ca, Mg, Na, K, Li, Fe, Mn, Al, and Si) were determined using a Beckmann SpectraScan IIIB Multielement Emission Spectrometer with a DC Plasma Excitation Source. Groundwaters were normally filtered through a 0.05-micrometer Nuclepore membrane under anaerobic conditions at the well head, then acidified with ultrapure HNO and sent to the laboratory for analysis.

Anion concentrations were determined using a Dionex Model 16 Ion Chromatograph in the mobile laboratory at the well site. Samples of anaerobically filtered water, water taken directly from the well, and water exiting the mobile laboratory were all used as sample for anion analysis. Varying sampling procedures did not produce discernable differences in the anion content.

Alkalinity was determined by using a Metrohm E636 Titroprocessor to titrate unfiltered samples with HCl.

The detergent content of the water was determined spectrophotometrically with a Hach Model DR-EL/4 Portable Laboratory. Detergent was a good indicator or tracer of drilling fluids in the well.

Eh was measured with a Sensorex S500C-ORP electrode, pH with an Orion "Ross" Model 81-02 combination electrode, sulfide with a Beckman #39610 Sulfide/Silver Electrode, and oxygen with a Fellow Springs Instrument Model 54 ARC dissolved-oxygen meter and electrode.

AMOUNT OF DATA

Four tables give the analyses of groundwaters from the pumped wells. Tables I and III are both titled "Elemental Concentrations in Groundwaters from the Vicinity of Yucca Mountain," and provide cation data; in I the concentrations are in mg/l and in II they are in mmols/l. Table II is titled "Anion Concentrations and Other Measurements for Groundwaters from the Vicinity of Yucca Mountain;" anion and detergent concentrations are in mg/l and Eh is given in mV vs H₂ electrode. Table IV titled "Alkalinity and Anion Concentrations in Groundwater from the Vicinity of Yucca Mountain" gives anion content in mmols/l and alkalinity in meq/l. In the tables the wells are arranged in the order of location, from west to east--the direction of downward slope of the stratigraphy.

In-situ organic content of the water from two wells were measured at Battelle Columbus. Total organic carbon contents were 0.15 mg/l in an integral sample from a producing well and 0.55 mg/l in a sample from a packed-off zone in a well.

Two tables give the analyses of the "thief" samples for two wells; some data for integral samples from the same wells are also included. Tables V and VI are both titled "Composition of Groundwaters from Yucca Mountain Wells 'Thief' Samples". Table V contains data for cation content (mg/l), Table VI data for anion content (mg/l) as well as Eh and alkalinity (meq/l).

Two tables provide data for groundwater from two wells in Pahute Mesa and one well in the Amargosa Desert. Tables VII and VIII are both titled "Elemental Concentration in Groundwater from Pahute Mesa Wells and a Well in the Amargosa Desert." In Table VII the pH and cation and anion concentrations in mg/l are given. In Table VIII the alkalinity in meq/l and the cation and anion concentrations in mmols/l are given.

Table IX, titled "Waste-Element Solubilities in Water from Three Yucca Mountain Wells," lists the solubility (m/l), the identity of the solid controlling solubility, and the primary aqueous species for the six waste elements in the three waters. The solubilities have been calculated with the EQ3/6 chemical equilibrium computer program and the

current thermodynamic data base. For the calculations, the compositions of the waters was taken from Tables I and II. One of the waters is typical of water below the proposed repository site, one is a "thief" sample representing the carbonate aquifer underlying much of the area and the most concentrated groundwater possible along the flow path, and the third is typical of wells surrounding Yucca Mountain. The six waste elements are U, Pu, Am, Sr, Ra, Tc.

Table X, titled "Repository Loading after 1000 Years," lists the moles or equivalents of multivalent waste elements (Np, Pu, Tc, and U) in a 70,000-MTHM (metric tons of heavy metals) repository loading of spent fuel after 1000 years.

Two figures are ternary diagrams which compare element concentrations in the various well waters. Figure 3 is titled "Relative Na-K-Ca concentration in Yucca Mountain water" and Figure 4 is titled "Relative Na-K-Mg concentration in Yucca Mountain water."

Five figures give the pH of water from an integral sample taken from a producing well as a function of various added chemicals. In Figure 5, "The pH of Well J-13 water and pure water as a function of added HCl," the data are plotted as pH (1.0 to 8.0) vs. HCl Added (0.0 to 10.0 mmoles).

In Figure 6, "The pH of Well J-13 water and pure water as a function of added NaOH," the data are plotted as pH (6.0 to 13.0) vs. NaOH Added (0.0 to 10.0 mmoles).

In Figure 7, "The pH of Well J-13 water as a function of pyrite oxidation," the data are plotted as pH (1.0 to 8.0) vs. Quantity of Pyrite Oxidized (0.0 to 2.0 mmoles).

Figure 8 is titled "The pH of Well J-13 water as a function of pyrite and local mineral addition. Local minerals were Na-clinoptilolite (2 mols/mol pyrite added), K-clinoptilolite (2 mols/mol pyrite added), Ca-clinoptilolite (2 mols/mol pyrite added), and cristobalite (5 mols/mol pyrite added)." The pH scale is from 5.0 to 9.0 and the Quantity of Pyrite Added scale is from 0.0 to 2.0 mmoles.

Figure 9 is titled "The pH of Well J-13 water as a function of NaOH and local mineral addition. Local minerals and addition rates are the same as those listed for Fig. 8." The pH scale is from 6.0 to 13.0 and the NaOH Added scale from 0.0 to 10.0 mmoles.

Five figures showing additional relationships among the compositional variables of the wells tested are given: Figure 10 "Relative fluoride content as a function of relative sodium content for waters from Yucca Mountain" ($F/[F+Cl]$ from 0.0 to 0.6 and $Na/[Na+Ca+K]$ from 0.5 to 1.0); Figure 11 "Ratio of ion activity product to equilibrium constant for calcite as a function of the ratio for magnesite for waters from Yucca Mountain. The dolomite saturation line is also shown" ($\log_{10}[IAP/K]$ for Calcite from -3.0 to 1.0 and $\log_{10}[IAP/K]$ for Magnesite from -4.0 to 2.0); Figure 12 "Ratio of ion activity product to equilibrium constant for calcite as a function of the ratio for fluorite for water from Yucca Mountain" ($\log_{10}[IAP/K]$ for Calcite from -3.0 to 1.0). Figure 13 "Total sulfate content as a function of total chloride content for water from Yucca Mountain" (Total Sulfate Content in mmoles/l from 0.0 to 1.5 and Total Chloride Content in mmole/l is from 0.0 to 2.0); Figure 14 "Total sulfate content as a function of total chloride content for water from Yucca Mountain--expanded scale" (Total Sulfate Content in mmoles/l from 0.0 to 0.5 and Total Chloride Content in mmoles/l from 0.0 to 0.5).

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

The integral samples obtained in the pumping tests have the disadvantage of being composite samples, having been obtained from all producing zones. Also, the wells may not have been pumped long enough to clear the well of drilling fluids. (Detergent-free water is termed formation water.)

It has not been established whether or not the results obtained on the "thief" samples are representative of water that is in equilibrium with the particular zone sampled. The data from the "thief" samples cannot be interpreted at this time.

KEY WORDS

Experimental data, Electrochemical, Spectroscopy, Ion chromatography, Field site, Yucca Mountain, J-13 water, Tuff composition, Tuff, Ambient temperature, Ambient pressure.

GENERAL COMMENTS

The report contains a large amount of data but much more data are needed to characterize the groundwater as a function of geographical area, depth, and time. The

authors question whether the reducing groundwater is representative of the entire repository block, how the water mineral reactions affect oxidation/reduction buffering capacity, whether calculations and laboratory experiments accurately represent the actual repository processes of solubility, particulate filtration, etc. These and other related topics are proposed by the authors for future work. This report has a rather long list of references, a number of which seem directly related to the data in this report.

RELATED HLW REPORTS

Los Alamos reports -

Reference	Report No.
9	LA-9328-MS
30	to be published
7	LA-9577-PR
8	LA-10032-PR
10	LA-9666-PR
12	LA-9484-PR
13	LA-9793-PR
14	LA-9846-PR
15	LA-10006-PR
11	LA-9331-PR

U.S. Geological Survey reports -

19	USGS-OFR-83-856
1	USGS-OFR-83-854
2	USGS-OFR-84-063
3	USGS-OFR-83-4171
4	USGS-OFR-83-856
5	USGS-OFR-83-855
6	USGS-OFR-83-853
20	USGS-OFR-83-542

Office of Nuclear Waste Isolation -

16	ONWI-448
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APPLICABILITY OF DATA TO LICENSING

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

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

This report deals with issue 2.1.3 regarding the chemical characteristics (Eh, pH, chemical composition, etc.) and issues 2.3.2 and 2.3.2.3 concerning the solubility of the waste form and radionuclides.

- (b) New Licensing Issues

- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Hanford Engineering Development Laboratory, P. O.
Box 1970, Richland, Washington 99352

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

Einziger, R. E., and Woodley, R. E., "Evaluation of
the Potential for Spent Fuel Oxidation under Tuff
Repository Conditions", HEDL-7452, March 1985.

DATE REVIEWED: 8/6/86

TYPE OF DATA

Review of literature data concerning oxidation of spent
fuel in breached zircaloy cladding.

MATERIALS/COMPONENTS

Spent fuel.

TEST CONDITIONS

Conditions possible in a tuff repository which would
impinge on the oxidation of spent fuel in rods with
breached cladding are discussed. These include water,
moist air or inert gas, temperature, and radiation field,
as a function of time.

METHODS OF DATA COLLECTION/ANALYSIS

Scenarios based on literature data and assumed conditions
in the repository are reviewed.

AMOUNT OF DATA

Twelve figures and 5 tables. Figures; Phase diagram, O/U
ratio from 2 to 3, temperature 0-1000°C; Log P (O₂)
(atm), (-10 to -50) vs T°C (0 -300); Time to spallation
(1-10000 h) vs 1/T(K) (360-190°C), measure of oxidation
rate of UO₂; Summary of conditions for fuel oxidation
experiments; Percent weight increase (0-4) vs time
(0-700) for oxidation of CANDU irradiated or unirradiated
fuel at 230 and 250°C; Log rate of weight increase in
percent per hour (10⁻⁵, -1) vs 1/T(K) (395-180°C) for spent
fuel and UO₂ in air; Initial potential tuff repository
environmental and waste package condition; Summary of

time dependence of tuff repository environmental and waste package conditions, 10^{-10^4} y vs atm, canister condition, rod integrity, fuel temperature and irradiation field; Rate of oxidation of spent fuel vs $1/T$ for two possible oxidation mechanisms with lower temperature mechanism having lower activation energy (schematic); Spent fuel scenarios 0-1000 y, showing possible pathways dependent on initial conditions; Cumulative damage fraction vs year breach occurs (used to estimate if breached cladding will split in 300 y); Spent fuel disposal scenarios beyond 300 years dependent on radiation field, temperature, atmosphere, moisture, and canister condition; Tables; Free energy changes for oxidation of UO_2 and U_3O_8 from 25-250°C; List of bare fuel oxidation experiments and conditions from literature; List of literature oxidation experiments on defected rods; Gamma field as function of time; Summary of fuel oxidation evaluation for time and atmosphere conditions.

UNCERTAINTIES IN DATA

Error bars on oxidation rate data of up to 20 percent. There may be real differences in fuels depending on previous history. With respect to correlating different methods of measuring oxidation rate the authors state, "It is difficult to relate the measures to one another since, in many cases, insufficient data are measured."

DEFICIENCIES/LIMITATIONS IN DATABASE

The effect of radiolysis of moisture on spent fuel oxidation rates and oxidation products formed in an inert atmosphere need to be determined. Low-temperature oxidation studies of UO_2 to determine low-activation-energy processes are needed.

KEY WORDS

Literature review, Tuff, Water vapor, Zircaloy, Spent fuel, Spalling, Oxidation of spent fuel.

GENERAL COMMENTS

This work focuses on the oxidation of UO_2 in breached cladding when stored in an air filled repository. The essential points are that oxidation of the spent fuel, which forms oxides of lower density, may lead to splitting of the cladding. The contention is that higher oxides of uranium may leach at higher rates than UO_2 , although data corroborating this contention is apparently

minimal. The authors state that only about 0.01 percent of the fuel rods contain defects in the form of small pinholes or splits. As oxidation of the spent fuel can occur only in rods with defective cladding this problem is clearly dependent on the cladding lifetime.

APPLICABILITY OF DATA TO LICENSING:

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

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

Related to issue 2.3.6, the potential damage and failure mechanisms for the fuel rod cladding in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

- (b) New Licensing Issues

- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Westinghouse Hanford Company, Richland, Washington
99352

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

Einziger, R. E., and Woodley, R. E., "Test Plan for Series 2 Thermogravimetric Analysis of Spent Fuel Oxidation", HEDL-7556, February 1986.

DATE REVIEWED: 8/5/86

TYPE OF DATA

Planned work involving oxidation of spent fuel.

MATERIALS/COMPONENTS

Turkey Point spent fuel.

TEST CONDITIONS

Oxidation in air or N_2 and $^{18}O_2$ at 175 and 155°C with 3, 1000, or 16000 ppm water vapor.

METHODS OF DATA COLLECTION/ANALYSIS

Rate of oxidation measurements will be made by thermogravimetric analysis (TGA). Samples to be characterized by ceramography, XRD, SEM, and ion microprobe.

AMOUNT OF DATA

Three tables and 6 figures. Figures; Flow sheet showing technical approach to measurements; Selection of test data; Temperature dependence of the bulk diffusion rate by assuming Arrhenius behavior of higher temperature oxidation rate data, rate constant, 10^{-3} to 3×10^{-4} vs $1/473-1/498$; Relationship of test time to oxidation behavior, schematic diagram showing regions of grain boundary and bulk diffusion at 225 and 140°C; Schematic of TGA system; Plot of $[1-(1-C)^{1/3}]$ vs $t^{1/2}$; showing grain boundary and bulk diffusion ranges at 200°C (from previous experimental work); Tables; TGA series 2 test matrix, delineates temperatures, durations, and moisture contents for proposed measurements; Moisture levels for

testing, dew point vs ppm water vapor concentration;
Characteristics of H. B. Robinson Unit 2 and Turkey Point
Unit 3 spent fuel.

UNCERTAINTIES IN DATA

Not discussed.

DEFICIENCIES/LIMITATIONS IN DATABASE

The authors point out that low temperature tests may indicate that oxidation takes place by a specific mechanism, such as grain boundary diffusion or bulk diffusion but that this mechanism may break down after a long period of time. They hope to overcome this limitation by running tests for a 3 month period.

KEY WORDS

Oxidation of spent fuel, Oxidation rate, Degradation of spent fuel, Mechanism of oxidation.

GENERAL COMMENTS

The impetus behind these proposed measurements is that oxidation of UO_2 in defective fuel rods may lead to the rupture of the zircaloy cladding due to the volume expansion of UO_2 that occurs on oxidation. In addition the leach rate of the higher oxides of uranium is apparently unknown but is thought to be higher than UO_2 . I believe the authors may be overly confident concerning the accuracy of their data because scatter in rate data by factors of 10 for measurements in different laboratories is typically found in the literature. I would also suggest that less sophisticated but longer term experiments (years) which could involve multiple samples might be more useful than those proposed. It is difficult to imagine how oxidation of UO_2 will take place in fuel rods until defects occur in the cladding. More definitive data on lifetime of the cladding and the leach rate of higher uranium oxides is important to this problem.

APPLICABILITY OF DATA TO LICENSING

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

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

Related to issue 2.3.6, the potential damage and failure mechanisms for the fuel rod cladding in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

- (b) New Licensing Issues

- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Hanford Engineering Development Laboratory, P. O. Box
1070, Richland, WA 99352

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

Wilson, C. N., "Test Plan for Series 3 NNWSI Spent
Fuel Leaching/Dissolution Tests", HEDL-7577, April
1986.

DATE REVIEWED: 8/23/86

TYPE OF DATA

Plan for leaching/dissolution study.

MATERIALS/COMPONENTS

Spent fuel from H. B. Robinson and Turkey Point PWR
reactors. Tests include undefected, laser drilled
defects, and slit defect cladding, as well as bare fuel,
in J-13 water.

TEST CONDITIONS

Four specimens at 85°C, one at 25°C from H. B. Robinson;
one specimen from Turkey Point at 85°C, in 250 ml of J-13
water. Samples characterized with respect to cladding and
position in fuel rod. Duration of test is to be 180 ds
with scheduled intermediate sampling. Two cycles are
planned.

METHODS OF DATA COLLECTION/ANALYSIS

Planned schedule for volume and time of sample withdrawal.
Filtered and unfiltered portions of solution will be
obtained. Analysis includes pH, uranium, alpha and gamma
spectrometry, ^{99}Tc , ^{90}Sr , ^{14}C , ^{126}Sn , ^{237}Np , and ^{129}I .
Ion chromatography analysis for anions and inductively
coupled emission spectrometry for cations will be
performed on selected solution samples. Filters will be
examined by SEM. Syringes will be stripped in 8 M HNO_3
and the solution analyzed for U and ^{241}Am by alpha
spectrometry.

AMOUNT OF DATA

Four figures, 4 tables, and 2 appendixes. Figures: Two figures showing fuel rod segments superimposed on ^{137}Cs gamma scan; Sectioning diagram for Turkey Point fuel rod; Series 3 test configuration for bare fuel tests and defected or undefected cladding tests. Tables: Test specimens giving identification, specimen configuration and fuel type; Characteristics of H. B. Robinson and Turkey Point fuels; Sampling schedule for leaching study listing time, volume, sample type and analyses; Analysis Methods for radionuclides listing nuclide, method, detection limits in pCi/ml, ppb, and 10^{-5} inventory in pCi/ml; Appendix: Discussion of apparent misidentification of at least two pin segments of a fuel rod identified as ATM-101 and deficiency report; Detailed drawings of NWWSI series 3 spent fuel leach test vessels.

UNCERTAINTIES IN DATA

The authors estimate that the leach bath temperature will be uncertain by $\pm 2^\circ\text{C}$. Error limits for radionuclide concentrations are listed.

DEFICIENCIES/LIMITATIONS IN DATABASE

Not discussed.

KEY WORDS

Leaching, J-13 water, Spent fuel, H. B. Robinson fuel, Turkey Point fuel, NWWSI.

GENERAL COMMENTS

This is a detailed report on plans for a third series of leach tests on spent PWR nuclear fuel. These tests differ from the series 2 tests in that the series 3 tests are to be run at 85°C in sealed type 304[L?] stainless steel vessels. The tests are semi-static in that periodic solution samples will be taken for analysis and replaced with an equal volume of fresh J-13 water. Primarily, there is too much detail of how procedures are carried out and very little discussion of the purposes of the tests. Neither the diagrams of the reaction vessels nor the discussion indicate what if anything is above the solution level. One might assume it is air. Later on it is stated that "Before removing the sample approximately 50 ml of air will be bubbled from deep in the vessel using a syringe. The bubbling has two purposes, to provide some limited convection mixing of the solution before sampling and to introduce air into the test, which are otherwise

sealed from the atmosphere." This is a poor procedure to introduce air into the test. If the presence of air is important to the test the amount should be better quantified than is allowed by this method.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues
Already Identified

Related to issue 3.3, the future geochemical processes/conditions that will affect release and transport of radionuclides to the accessible environment in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

WASTE PACKAGE DATA REVIEW

DATE SOURCE

(a) Organization Producing Data:

Hanford Engineering Development Laboratory, Operated
by Westinghouse Hanford Company, PO Box 1970,
Richland, WA 99352

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

W. H. Yunker, "Corrosion of Copper-Based Materials in
Gamma Radiation," HEDL-7612, June 1986.

DATE REVIEWED: 10/7/86

TYPE OF DATA

Experimental, corrosion, SCC, crevice corrosion

MATERIALS/COMPONENTS

Pure Cu (CDA 101), seven percent aluminum bronze
(CDA 613), 70Cu-30Ni (CDA 715)

TEST CONDITIONS

Three types of specimens: flat samples, bent strips for
SCC, crevice specimens. Environment: water vapor-air at
95°C and 150°C, and immersed in J-13 water at 95°C. Gamma
radiation field of 10^5 rd/h. Exposure: 1, 3 and 6 months

METHODS OF DATA COLLECTION/ANALYSIS

Changes in chemical composition of gas and water.
Weight changes, oxide film weights, visual examination.
Chemical composition of oxide films by Auger electron
spectroscopy and x-ray diffraction.
Dye-penetrant tests for SCC.

AMOUNT OF DATA

18 figures: 9 are drawings or photographs of the assembly
and specimens.

- 1) Gamma flux (0 to 5.5×10^{-5} rd/h) vs position
from top of irradiation tube specimens (0 to
 0.6 m)
- 2) Vessel gas pressure (0 - 300 KPa) vs time up to
115 days

Corrosion data on axes of 1 to 100 mg, 10^2 to 10^3 h:

- 1) log weight loss (mg) vs log time (h) for CDA 101
- 2) log weight loss (mg) vs log time (h) for CDA 613
- 3) log weight loss (mg) vs log time (h) for CDA 715

Four Graphs are composition profiles of the surface film vs. sputter time from Auger data, two for CDA 613 and two for CDA 715.

- 14 Tables:
- 1) Number of specimens exposed (254)
 - 2) J-13 water composition
 - 3) Estimated γ -Dose rates for various kinds of waste
 - 4) Composition of specimens
 - 5) Data logger input channels
 - 6) Example of data report
 - 7-8) Gas composition
 - 9) Water composition in vessel T-1 at 95°C
 - 10) Water Composition
 - 11) Auger analyses for CDA 101, 613 and 715
 - 12) X-ray diffraction results
 - 13) Uniform corrosion data
 - 14) Stoichiometry of Cu oxide on CDA 101

Appendix: Results of visual examination of weight loss and bend specimens after oxide film removal.

UNCERTAINTIES IN DATA

No statistical analyses of the results

DEFICIENCIES/LIMITATIONS IN DATABASE

Data are considered preliminary. Longer exposure times (more than 6 months) are currently in progress.

KEYWORDS

Copper, Copper alloys, Corrosion, SCC, Crevice corrosion, γ -field.

COMMENTS

Results indicate that pure copper (CDA 101) has less tendency to localized attack when compared with the alloys tested and that the bronze alloy, (CDA 613), containing seven percent aluminum shows lower uniform attack. The presence of liquid on the surface enhances corrosion. Only for 70Cu-30Ni alloy (CDA 715) a significant enhancement of uniform corrosion with γ -ray exposure was found. No cracking was observed.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

This document addresses issue 2.2.4.2 (What is the effect of radiation on the corrosion behavior of the waste package container?). Particular emphasis in the document is devoted to radiation effects on localized modes of corrosion.

(b) New Licensing Issues

(c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data:

Lawrence Livermore National Laboratory, Livermore CA 94550.

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

Rothman, A.J., "Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Nuclear Fuel in a Tuff Repository, Attachment 10 to MRB-0418," UCID-20172, September 1984. An informal report intended for internal and limited external distribution.

DATE REVIEWED: 3/5/87

TYPE OF DATA

This review and analysis contains experimental and theoretical data from other publications as well as calculations by the author on failure mechanisms involved in oxidation of Zircaloy and in stress related effects.

MATERIALS/COMPONENTS

Spent fuel rods from both pressurized and boiling water reactors consisting of 600 to 700 μm thick Zircaloy-4 or 800 to 900 μm thick Zircaloy-2 cladding, respectively.

TEST CONDITIONS

Test conditions for this review and analysis include: (a) repository conditions with low flow J-13 water after 300 years and temperatures of 150°C, and (b) repository conditions in virtually stagnant water after 1000 years with temperatures just below 100°C and one atmosphere pressure to provide data for 10,000 year prediction; storage involves horizontal emplacement at 44 kW/acre, 3.3 kW/package in a 0.50-m diameter canister with a package length of 4.5 m and a borehole spacing of 52 m. Test conditions in papers reviewed are numerous and include some Zircaloy exposure in water, fused salt, air, acid solutions, autoclaves with various environments, stresses up to 140 MPa (20,000 psi), temperatures of 450°C and below.

METHODS OF DATA COLLECTION/ANALYSIS

Some data are taken from other publications, other data are calculated or obtained through analysis of earlier experiments or postulates.

AMOUNT OF DATA

There are 7 tables and 3 figures. The tables include: Table 1, Composition of Zircalloys; Table 2, Experimental Data Indicating Water Chemistry for a Repository in the Topopah Spring Tuff; Table 3, Calculated Depth of Oxidation Assuming Constant Temperature (180°C) for 10,000 Years; Table 4, Composition of Waters at 90°C; Table 5, Effects of Some Hydroxides on Aqueous Zircaloy Corrosion; Table 6, Pressure vs Time in Fuel Rods (Based on Calvert Cliffs Combustion Engineering Fuel Rod #5); Table 7, Calculations of Stress Intensity Factor (K) for Crack Lengths Ten Times the Crack Depth. The figures are: Fig. 1, Peak Fuel and Canister Temperatures vs Time (44 kW/acre); Fig. 2, Oxidation of Zircaloy - Weight Gain vs Time; Fig. 3, Peak Fuel and Canister Temperatures vs Time (82 kW/acre) (Horizontally Emplaced PWR Spent Fuel with 12 Internal Fins).

UNCERTAINTIES IN DATA

Experiments have not been conducted to explain kinetics of Zircaloy oxidation below 280°C. The oxidation process described is an idealization. Details of oxidation and corrosion need to be confirmed by experiments. It is not known whether failure mechanisms established for times of up to twenty years will hold for longer times. Uncertainties for modeling stress corrosion cracking (SSC) failures include insufficient knowledge of chemical environment inside the fuel rod (Cs, I, Cd, etc.), the effect of chemical defects/inhomogenieties, internal pressures due to fission gas, mechanical flaw size, and others.

DEFICIENCIES/LIMITATIONS IN DATABASE

Evidence indicates that SCC will not occur in a repository but stronger proof and further work are needed on this least understood failure mechanism. Additional work is needed to show effect of fluoride ions. More data are needed on delayed hydride cracking and on slow cooling which reorients hydrides radially to form an easier crack path for a given hoop stress.

KEYWORDS

Literature review, Corrosion, Surface film, Air, J-13 water, Zircaloy 2, Zircaloy 4, Spent fuel cladding, J-13 steam, Oxidation, Stress corrosion cracking, Hydrogen embrittlement.

APPLICABILITY OF DATA TO LICENSING

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

- (a) Relationship to Waste Package Performance Issues
Already Identified: 2.3.6 Potential damage and failure mechanisms for spent fuel cladding
- (b) New Licensing Issues
- (c) General Comments: This is an informative review and analysis. Careful attention should be paid to the uncertainties, deficiencies and limitations listed by the author. Data and assumptions of other authors were used in this paper to make further assumptions and calculations. Much more experimental data are needed for making a long term extrapolation of Zircaloy cladding durability.

P. Soo and E. Gause in NUREG/CR-2482, Vol. 7, April 1984, reviewed some of the same papers and also concluded that stress corrosion could occur. Soo and Gause also indicated that crud on the outside of the cladding was a source of radionuclide release. More data are needed on SCC susceptibility, protectiveness of the oxide film, effects of fluoride and other ions and on the amount and type of nuclides present or released from the cladding at different times extending over a long period of time.

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data:

Earth Sciences Department Lawrence Livermore National Laboratory, Livermore, CA 94550

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

Oversby, Virginia M., "Important Radionuclides in High Level Nuclear Waste Disposal: Determination using a Comparison of the EPA and NRC Regulations," UCRL-94222, Preprint, February 1986.

DATE REVIEWED: 8/11/86

TYPE OF DATA

Analysis of amount and decay of radionuclides in spent fuel.

MATERIALS/COMPONENTS

PWR spent fuel.

TEST CONDITIONS

Calculations based on data for PWR spent fuel.

METHODS OF DATA COLLECTION/ANALYSIS

The analysis identifies those radionuclides most likely to be important in meeting NRC (10 CFR Part 60) and EPA (40 CFR Part 191) regulations pertaining to licensing and performance objectives for nuclear waste repositories.

AMOUNT OF DATA

Eight tables; Table 1 gives the inventory of significant radionuclides in PWR spent fuel at 10, 100, 1,000, and 10,000 years in Ci/MTIHM (Curies/Metric Ton of Initial Heavy Metal). Also included is the inventory at 300 years for those nuclides changing greatly in abundance between 100 and 1000 years; Table 2, Performance objective for control of release rates, lists nuclide, inventory at 1000 y, 1/100,000 of 1000 year inventory, and level to which release rate must be controlled in Ci/MTIHM per year, ratio of controlled release rate to 1/100,000 inventory; Table 3, Comparison of release rates allowed in 10CFR60 to the cumulative release limits in 40CFR191, lists nuclide,

table 1 EPA limit in Ci/MTIHM (from Table 1 EPA 40CFR191), release under 10CFR60 in Ci/9000 year, ratio of NRC/EPA, factor to reach $R=0.035$, (.035 is used to insure that no more than .035 of the total EPA limit is due to each radionuclide); Table 4, Comparison of the 1,000 year inventory of radionuclides to the EPA release for low probability events, lists nuclide, 10 times EPA limit Table 1 and inventory at 1000 years in Ci/MTIHM, ratio of inventory to 10 times EPA Table 1, factor to reduce ratio to $R=.035$; Table 5, Reduction factors for maximum NRC allowed releases, radionuclides listed by order according to the amount by which their release exceeds the amount that could be released to the accessible environment, assuming that each radionuclide accounts for no more than 0.035 of the total EPA limit; Table 6, Comparison of releases controlled by matrix dissolution at 1/100,000 per year for 9000 years with EPA cumulative release limits; lists nuclide, release for 9000 year and EPA Table 1 limit in Ci/MTIHM, ratio of release to EPA limit, factor to reduce ratio to .035; Table 7, Reduction factors for release at the rate of 1/100,000 per year for all radionuclides plus a one percent spiked release for Cs, C, I, and Tc in the 1000 year inventory, lists nuclides and factors by which release at edge of engineered barrier system exceeds EPA limit assuming no radionuclide accounts for more than .035 of EPA limit; Table 8, Same as Table 6 except matrix dissolution is at 1/1,000,000. All values except EPA Table 1 limit are reduced by a factor of ten.

UNCERTAINTIES IN DATA

Not discussed.

DEFICIENCIES/LIMITATIONS IN DATABASE

None mentioned.

KEY WORDS

EPA regulations, NRC regulations, Repositories, Commercial high level waste, Spent fuel, Power reactors

GENERAL COMMENTS

This report analyzes the radioactive waste inventory of a generic repository with respect to EPA regulations governing the total release to the accessible environment and the NRC regulation governing rate of release from the repository. Seventeen elements are identified as materials for which data on solubility and sorption would be needed for use in site performance assessment but by far the most important elements are americium and plutonium because of their initial concentrations and long half lives. Note that the EPA regulation puts a limitation

on release to the accessible environment in terms of Ci/MTHM starting 1000 years after closure of the repository, while the NRC regulation requires that less than 1/100,000 per year of the inventory at 1000 years is released from the engineered barrier system. Under these regulations, release into the environment is only allowed after 1000 years so that ^{137}Cs and ^{90}Sr , either of which presents problems during initial handling of spent fuel, have essentially decayed away by the time that this requirement comes into effect. The term Ci/MTHM is used in the generic sense in the EPA regulation while Ci/MTIHM refers to the spent fuel and is used in the NRC regulation. These terms are equivalent when allowance is made for radioactive decay in MTIHM.

APPLICABILITY OF DATA TO LICENSING

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

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

Related to issues 3.3.1, the solubility/concentration of radionuclides under various flow regimes affect radionuclide release and transport and 3.3.1.2, the speciation affect radionuclide solubility/concentration in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

- (b) New Licensing Issues
- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data:

Pacific Northwest Laboratory, Richland, Washington
99352

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

Gray, W. J., "Analysis of the Effects of Radiation on the Chemical Environment of a Waste Package in a Nuclear Waste Repository in Basalt", B036177, March 1984.

DATE REVIEWED: 10/28/86

TYPE OF DATA

Experimental data on the effects of gamma and alpha radiation on simulated Grande Ronde basalt groundwater. Some computer calculations on the radiolysis of methane are made but the major emphasis is on an experimental approach.

MATERIALS/COMPONENTS

Simulated basalt groundwater. Some experiments included the presence of basalt and/or bentonite. To simulate the composition of basalt ground water, which contains up to 700 mg/l of methane at 25°C, samples were pressurized under gaseous N₂ and CH₄.

TEST CONDITIONS

The simulated basalt groundwater was placed in stainless steel vessels with quartz liners and pressurized to about 0.14 MPa with gaseous N₂ and 3.30 MPa with gaseous CH₄. Gamma irradiation was done in a ⁶⁰Co facility at dose rates of 2x10⁴ to 5x10⁶ rd/h. Alpha irradiation was done by spiking the solutions with ²³⁸Pu at a concentration of 4x10⁻⁴ M resulting in a dose rate of 2x10⁴ rd/h. The Pu concentration exceeds the solubility in the simulated groundwater at a pH above 3 so the solutions were stirred to provide uniform alpha exposure. The volume ratio of liquid to gas was about 4/1. For the gamma irradiation about 3/4 of the gas volume was exposed to the high radiation field. For the alpha irradiation, only the liquid phase was irradiated.

METHODS OF DATA COLLECTION/ANALYSIS

Gases were analyzed by mass spectrometry. Analysis methods for liquids included Inductively Coupled Plasma (ICP) spectroscopy for cations and carbon analysis for organic and inorganic carbon. Solids were analyzed by Fourier Transform Infrared spectrometry. Gel Permeation Chromatography was used to establish molecular weight distribution of the polymeric radiolysis products. Elemental fractions of C, H, and O were determined with an elemental analyzer.

AMOUNT OF DATA

Tables: Composition of Synthetic Grand Ronde Basalt Groundwater; Solution Concentrations used in Radiolysis Experiments; Composition of Water Used in Irradiation Tests; Irradiation Conditions and Post-Irradiation Analysis Performed; Irradiation Conditions and Gas Compositions of Irradiated Samples; Concentrations of Cations Found in Irradiated Solutions; Analytical Results from some of the Gamma Irradiation Tests; Gas Concentrations and Total Pressure Predicted at Total Dose of 46 Mrd (Dose rate was 1.5 Mrd/h); Gas Concentration and Total Pressure at Total Dose of 77 Mrd (Dose rate was 1.1 Mrd/h).

Figures: IR Absorption Spectra for Tests 45 and 50; Gas Pressures Resulting from the Radiolytic Decomposition of Methane, Gas Pressure (10^{-4} to 100 atm) vs Time (1 to 10^5 s) based on computer model.

UNCERTAINTIES IN DATA

Not dealt with except to note that deviant measurements are thought to result from mistakes in procedure.

DEFICIENCIES/LIMITATIONS IN DATABASE

There is some evidence that Si was leached from the quartz inserts during the radiolysis measurements.

KEY WORDS

Alpha radiation, Basalt, Basic solution, Bentonite, Computer calculation, Data analysis, Experimental data, Gamma radiation, High pressure, High temperature, Laboratory, Simulated basalt groundwater, Radiolysis of basalt groundwater.

GENERAL COMMENTS

The radiolysis of basalt groundwater, which has a high concentration (up to 700 mg/l at 25°C) of CH₄ (methane), by alpha and gamma radiation, results in the formation of higher molecular weight organic substances. Average molecular weights of the organic fractions vary from 100 to 50,000 and average from 2,000 to 3,000. The original intent of this work was to develop and use computer modeling to calculate concentrations of radiolytic products and performing only a few laboratory measurements to make adjustments in the model. However, because of the complexity of the process, many reaction rate constants are not known, so more emphasis has been placed on laboratory measurements. The primary concern has been to identify and characterize the higher molecular weight products which it is hoped may serve as absorption sites for leachate radionuclides. But, the report emphasizes that experimental problems caused by Si contamination may have affected some of the results. No effort is made in this report to relate the results to possible effects in a repository. The report suggests that this will be a future objective.

APPLICABILITY OF DATA TO LICENSING:

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

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

This report provides supporting data for issue 2.1.3.1 regarding how radiolysis effects the chemical nature of the groundwater reaching the waste package container and for issue 2.3.5 regarding how the release rate of radionuclides is likely to be affected by radiation.

- (b) New Licensing Issues

- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data:

Pacific Northwest Laboratory for Rockwell
Hanford Operations/BWIP, Richland WA

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

R. E. Westerman, "Corrosion Evaluation of Candidate
Iron-Base Nuclear Waste Package Alloys in Grande
Ronde Basalt Groundwater," SD-BWI-TI-235, February
1984.

DATE REVIEWED: 11/31/86

TYPE OF DATA

1. Scope: Status report containing experimental data
2. Failure Mode: This investigation was intended to study general or uniform corrosion behavior, however, non-uniform attack was observed.

MATERIALS/COMPONENTS

Coupons of six Fe based alloys:

1. Cast ductile iron
2. 1.25Cr-0.5Mo Cast Steel
3. 2.5Cr-1.0Mo Cast Steel
4. 9Cr-1Mo Steel (Rolled, Austenitized-932°C,
Tempered-704°C)
5. 1025 Cast Steel
6. 1020 Wrought Steel (Hot Rolled)

TEST CONDITIONS

1. Because earlier studies indicated that inexpensive iron-based alloys may meet the requirements for corrosion resistance in the environmental conditions expected in a repository constructed in a basalt formation, six iron-based alloys were studied in this investigation. These are:
 - Autoclave environment
 - Two test temperatures: 150 and 250°C
 - Synthetic Grande Ronde Basalt Groundwater (Differs from composition now considered to represent this environment)
 - Aerated with 6-8 mg/L of O₂

- Up to 20-months total exposure
- Flowing solution at 35 ml/h. (not recycled)
- Solution passed through crushed basalt

2. Influence of Irradiation on Weight Loss

- Same as above with ^{60}Co γ irradiation source (3×10^5 rd/h)
- Test Temperature = 250°C
- Up to 16-months total exposure
- Two of the alloys were not tested in this environment:
 - (a) 1.25Cr-0.5Mo Cast Steel
 - (b) 9Cr-1Mo Steel

METHODS OF DATA COLLECTION/ANALYSIS

Weight Loss after descaling (with formaldehyde-inhibited HCl)

Corrosion product analysis by X-ray diffraction

More detailed description of procedure given in Westerman, Pitman and Nelson, PNL-4364, 1982

AMOUNT OF DATA

5 Tables

- 1) Ferrous Alloy Compositions: All 6 alloys investigated
- 2) Composition of Umtanum Flow Basalt
- 3) Composition of Hanford Grande Ronde Basalt Groundwater
- 4) Summary of Corrosion data at 150°C and 250°C (6 Alloys)
- 5) Summary of Corrosion-Irradiation data at 250°C (4 Alloys)

3 Figures [Corrosion rate vs. Exposure Time]

- 1) Corrosion Rate at 250°C (0-30 $\mu\text{m}/\text{y}$)-(0-20 months)
- 2) Corrosion Rate at 150°C (0-30 $\mu\text{m}/\text{y}$)-(0-20 months)
- 3) Corrosion Rate During Irradiation at 250°C (0-30 $\mu\text{m}/\text{y}$)-(0-20 months)

UNCERTAINTIES IN DATA

Scatter in weight loss measurements

Non-uniform corrosion observed in all alloys tested (varied with temperature).

The synthetic groundwater used differs in composition from that now considered to represent this repository environment.

The γ radiation levels used were conservatively high.

DEFICIENCIES/LIMITATIONS IN DATABASE

Environment did not contain basalt/bentonite packing.

KEYWORDS

Experimental data, Visual Examinations, Weight change, Laboratory, Hanford, Simulated Groundwater, Dissolved Oxygen, Gamma, High Temperature, High Pressure, Dynamic, Steels, 1020 Carbon Steel, 1025 Carbon Steel, Cast, Wrought, ^{60}Co , Groundwater, Basalt, Corrosion (general), Corrosion (pitting), Radiation Effects.

GENERAL COMMENTS

Preliminary status report on the progress to date (February 1984) on the evaluation of candidate iron-base alloys for use in basalt.

The authors conclude that the corrosion rates are somewhat higher at 150°C and under irradiation, but that quantitative comparison is difficult. In fact, the scatter in the results is large compared to the differences and any conclusion based on these measurements is premature.

The authors do not state how the oxygen content of the solution was determined and maintained at the same level (6-8 mg/L) for the two different temperatures. Presumably, this is clarified in the referenced report and will be clarified in the final report.

The authors point out that the environment used is not the same as that currently expected for this repository environment, however, they do not discuss the potential influence of this difference on the observed corrosion rates.

While the authors acknowledge the lack of basalt-bentonite packing material in the environment, they imply that the packing material would mitigate corrosion and, as a result, the measurements are conservatively high. However, the presence of basalt and bentonite packing in physical contact with the samples may limit transport and exaggerate the localization of corrosive attack.

RELATED HLW REPORTS

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

APPLICABILITY OF DATA TO LICENSING

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

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

This paper addresses BWIP ISTP issues 2.2.4.1, the corrosion rates for various corrosion modes of the waste package container, and 2.2.4.2, the effect of radiation on the corrosion behavior of the waste container.

- (b) New Licensing Issues
- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Report

Westinghouse Hanford Co. for Rockwell Hanford
Operations and Westinghouse Hanford Operations.

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

L.A. James and L.D. Blackburn, "BWIP Crack Growth
Studies," SD-BWI-TI-120, Rev. 0, June 1983.

DATE REVIEWED: 10/6/86; Revised 11/13/86; 5/25/87

TYPE OF DATA

(1) Scope of the Report

Proposed work plan and background material including short
literature review for Basalt Waste Isolation Project.

(2) Failure Mode or Phenomenon Studied

Environmentally assisted cracking (EAC) in waste package
containers, includes stress corrosion cracking studies
conducted using fracture-mechanics and other test methods.

MATERIALS/COMPONENTS

Carbon steels, a 9Cr-1Mo steel and carbon-steel weldments,
as waste container materials, specifically:

1. AISI 1020
2. AISI 1025
3. ASTM A387 Grade 9, Class 2 (a 9Cr-1Mo steel)
4. ASTM A36

TEST CONDITIONS

(1) State of the Material being Tested

AISI 1020, in both the as hot-rolled and the as cast
conditions.

AISI 1025, both as hot-rolled and as cast.

ASTM A387 Grade 9, Class 2.

ASTM A36, as hot-rolled.

(2) Specimen Preparation

Specimens are to be precracked. Initial levels at the stress-intensity factor (K) are to be 30, 40, and 60 MPa/m. Specimens will be standard ASTM E647 "Modified Compact Type" in the "1T" size (1-inch thick by 2.5-inch wide by 2.4-inch high). Specimens will be fatigue tested according to ASTM E647-83.

(3) Environment of the Material being Tested

Specimens will be fatigue tested in autoclaves using simulated Hanford* groundwater at 150 and 250°C and pressures above the 4 MPa (580 psia) required at 250°C to maintain pure water as a liquid. The composition of the typical simulated groundwater is given and contains the carbonate (27 ppm) and hydrogencarbonate (55 ppm) ions which have the potential to cause environmentally assisted cracking (EAC) if present in sufficient concentration. Other ions present (in ppm) include sodium (358), potassium (3.4), calcium (2.8), chloride (312), sulfate (172), trihydrogensilicate (120), and fluoride (33).

*Referred to as synthetic Grande Ronde groundwater in the abstract of the report, but elsewhere as Hanford groundwater.

METHODS OF DATA COLLECTION/ANALYSIS

A test program involving static-load and cyclic-load tests is outlined. Linear-elastic fracture mechanics (LEFM) techniques will be used to permit use of quantitative relationships between the stress-intensity factor (K), the applied stress level and the flaw size. These relationships combined with experimentally-measured fracture mechanics parameters (e.g., crack-growth rates, crack-growth thresholds), allow quantitative estimates to be made for structural lifetimes under environmentally-assisted cracking (EAC) conditions. For repository lifetimes approaching 1000 years or longer, probably the most important property is the static crack-growth threshold, K_{Isc} , because the significant loadings under BWIP repository conditions are anticipated to be static. A threshold approach must be taken because it is impossible, from a practical experimental point of view, to measure crack-growth rates low enough to ensure (without extrapolation) that wall penetration or other unsatisfactory cracking behavior would not occur in 1000 years. To arrive at a conservative value of K_{Isc} , cyclic test results should be employed even though significant cyclic loadings are not anticipated under the BWIP conditions. ASTM E647-82 test methods for fatigue crack growth tests will be followed generally, i.e. for

preparation of the specimen, loading, crack length measurements, and LEFM data analysis. Selected parts of MCC-104S, which contains several sections pertaining to refreshed autoclave testing, may be used to supplement E 647-82 for these tests.

The following parameters are to be determined: static K_{ISCC} ; effects of temperature, cyclic frequency, loading waveform, and stress ratio upon corrosion fatigue; and effect of alloy composition. Exposure times are to be 2000, 10000, and 20000 hours for each temperature.

The proposed test program for the cyclic loading tests includes the following parameters: reference environment of either vacuum or dry air, test environment of refreshed simulated Hanford groundwater, temperatures 150 and 250°C, stress ratios 0.05 and 0.67, sawtooth waveform, frequencies 10, 1, 0.1, 0.01, and 0.001.

AMOUNT OF DATA

Data given are from other references which simply provide support for the background material given and the discussion.

UNCERTAINTIES IN DATA

Not addressed by the authors.

DEFICIENCIES/LIMITATIONS IN DATABASE

Not addressed by the authors.

KEY WORDS

Planned work, Linear-elastic fracture mechanics (LEFM), Laboratory, Hanford, Simulated groundwater, High temperature, High pressure, Static (no flow), Stainless steel, 1020 carbon steel, 1025 carbon steel, ASTM A36, ASTM A387, Cast, Wrought, Hot-rolled, Standard compact specimen, Crack growth rate, Crack growth threshold, Fracture toughness, Stress corrosion cracking, Fracture, Fatigue, Autoclave test, Environmentally assisted cracking.

GENERAL COMMENTS

For materials that exhibit any interaction, other than uniform corrosion, it will not be easy to demonstrate in a conservative manner that a material will be an effective barrier for 1000 years. First, a variety of screening tests are needed determine whether or not potentially unfavorable interactions are detectable. In some cases, tests must be conducted over extended times, and in other

cases they must be conducted under loading or environmental conditions that are conservatively chosen to be more severe than expected repository conditions.

Very sensitive measures of crack extension, such as acoustic monitors, should be used so as to permit detection of microscopic fracture events. This might improve the practical lower limit for crack defection to the extent needed for useful extrapolations of crack-growth-rate data. The authors correctly note that a mechanistic understanding (of the effects of the BWIP environment on the candidate materials) is also needed to furnish a bases for extrapolations used to predict performance over times much greater than those practical for actual materials testing.

While a number of tests are proposed by these authors for an initial series of tests, the proposal assumes that the pre-existing active-path mechanism is not expected to be important for the BWIP candidate materials. This reviewer notes that it must be demonstrated that the microstructural stability of the candidate materials (after fabrication) is sufficiently great (or sufficiently understood) so that this argument holds true over the 300 to 1000 year containment period. Microstructural stability is therefore an issue that must be resolved. In addition, various types of tests are needed to determine whether or not any observed interactions, between the candidate material and the environment, are significant in relation to repository performance. The results of these tests can be used to illucidate important test parameters. Then, further tests can be planned to predict performance over the long times needed for repository service.

In addition to determining threshold stress-intensity values for static loading conditions, crack extension rates in fatigue tests will be determined. The fatigue approach is a practical method of introducing a degree of conservatism into the experimental program. Variations in the environment that would introduce a degree of conservatism should also be considered, especially for the static case. In addition, plans should be made by DOE to initiate very long-term EAC (environmentally assisted cracking) studies, to supplement data obtained from these planned tests. Test times of the order of 10 to 50 years should be considered, and exposures should be initiated as soon as sufficient information is available to determine the most appropriate test matrix.

RELATED HLW REPORTS

SD-BWI-TI-165
SD-BWI-TS-012
RHO-BW-SA-560 P

APPLICABILITY OF DATA TO LICENSING

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

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

This is classified as supporting data. It addressed issue 2.2.4.1 (evaluation of rates of corrosion as a function of time) for various corrosion modes of the waste package container.

- (b) New Licensing Issues
- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Hanford Engineering Development Laboratory, P. O. Box
1970, Richland, WA 99352.

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

Funk, C. W. and Jacobson, L. D., "Inventory and
Characterization of Spent LWR Fuel", HEDL-TME 77-82,
April 1979.

DATE REVIEWED: 9/12/86

TYPE OF DATA

An inventory of light-water reactor (LWR) fuel discharged from reactors and stored in water basins, with discussions of its condition. Data obtained from reactor discharge records. Inventory covers the period up to and including 1976.

MATERIALS/COMPONENTS

UO₂ pellets, zircaloy cladding, stainless steel cladding.

TEST CONDITIONS

Fuel rods stored in water.

METHODS OF DATA COLLECTION/ANALYSIS

Certain analyses of the data gleaned from the discharge reports are mentioned under amount of data. Ratios of failed rods to total number of rods are given.

AMOUNT OF DATA

Tables:

- 1 Inventory of Discharged LWR Fuel Assemblies.
- 2 Summary of Reprocessed LWR Fuel at NFS (Nuclear Fuel Services).
- 3 Estimate of Stored Commercial LWR fuel Assemblies.
- 4 Comparison of Stored LWR Fuel Assemblies at End of 1976 (based on several sources).
- 5 Inventory of LWR Fuel Assemblies Releasing Radioactivity.
- 6 Trend of LWR Defective Fuel Assemblies (with time).
- 7 Crud Thickness.
- 8 Post Reactor Observations of Cladding Oxidation.

- 9 Examples of Rod Gas Plenum Analysis.
- 10 Mechanical Properties of Dresden Zircaloy-2 Cladding.
- 11 Microhardness of Zircaloy Cladding as a Function of Flux.
- 12 PWR Fuel Performance Status Report (Third Quarter 1976).
- 13 Long-Term Release of Activity in KWL (Kern Kraftwerk Lingen GmbH) Fuel Storage Pool.
- 14 Examples of Iodine Spiking.
- 15 Guidelines for Contamination Control.
- 16 Distribution of Generated Radioactivity for Typical LWR Spent Fuel Assembly.
- 17 Fission Product Generated Radioactivity Ranking of Largest Contributors.
- 18 Generated Radioactivity Ranking of Transuranic Activity Contributors.
- 19 Generated Radioactivity Ranking of Light Element Activity Contributors.

Figures:

- 1 Reported Number of Discharged LWR Fuel Assemblies from U. S. Reactors Each Year.
- 2 Reported Number of Defective LWR Fuel Assemblies (Breached and Leakers) from U. S. Reactors Each Year.
- 3 Examples of Pellet-Cladding Interaction (PCI).
- 4 Examples of Hydride Perforations Resulting in Release of Plenum Gases Including Fission Products.
- 5 Examples of Corrosion Deposits (Crud) on Exterior of Fuel Rods.
- 6 Examples of Spalled Crud and External Oxide on Zircaloy 2 Cladding
- 7 Examples of the Effect of Fission Product Iodine on Clad Cracking.
- 8 Examples of Fretting Damage.

Appendixes:

- A Definitions applicable to failed fuel rods
- B Sipping Procedures (A sampling method to detect leaking fuel elements)
- C Decay Heat and Fission Gas Product Inventory for LWR Spent Fuel - Figure C-1: Decay Heat of LWR Fuel
- D Burnup of Stored Fuel - Table D-1: Maximum Burnup of Spent Fuel
- E Storage Duration of Spent Fuel - Table E-1: Maximum Residence time of Spent Fuel Assemblies in Pool Storage

UNCERTAINTIES IN DATA

Not Applicable

DEFICIENCIES/LIMITATIONS IN DATABASE

The decreasing trend of defective assemblies (1976) needs validation by determining the ratio of defectives to inventory in 1977. The significance of incipient leakers on long-term storage needs more detailed evaluation. The amount of potential release through service-weakened cladding may require an additional load on the storage facility clean-up system.

KEY WORDS

LWR spent fuel, Fretting, Pellet-cladding interaction, Hydrogen attack on cladding, Crud.

GENERAL COMMENTS

In this report, spent light water reactor fuel discharged from reactors and stored in water basins has been inventoried through 1976. Defective fuel has been defined and categorized, and the trends of failed fuel have been evaluated for their impact on the operation of receiving facilities. Failure by mechanisms that may lead to release of radioactivity have been identified. The sources and levels of activity have been estimated for typical LWR spent fuel assemblies. By estimating the conditions of commercial spent LWR fuel assemblies, the survey was originally assembled to provide a basis for the conceptual design of the receiving operations of LWR reprocessing plant. However, the same information is valuable for final nuclear waste disposal. Some copies of the tables are illegible and a few pages are missing in the reviewers copy of this document.

APPLICABILITY OF DATA TO LICENSING

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

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

This paper relates to licensing issues 2.3.6 on damage and failure mechanisms of fuel rod cladding and 2.3.6.1 on predicted rate of failure for each given mechanism of failure.

- (b) New Licensing Issues
- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Argonne National Laboratory, 9700 South Cass Avenue,
Argonne, Illinois 60439

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

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, February 1986.

DATE REVIEWED: 9/17/86

TYPE OF DATA

Leach 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), PNL 76-68 glass ATM-1c which contained uranium,
PNL 76-68 glass ATM-8 containing added ^{237}Np , ^{239}Pu , and
 ^{99}Tc , J-13 water pre-equilibrated at 90°C for two weeks
with Topopah Spring tuff, (EJ-13), tuff, 304L SS, air.

TEST CONDITIONS

Leach tests were run in type 304L stainless steel reaction
vessels which contained the sample in about 16 ml of EJ-13
water and at least 4 ml of air. These vessels were sealed
from the environment with silicone gaskets and exposed to
a gamma radiation field produced by a ^{60}Co source of 2×10^5
rd/h as measured inside the reaction vessel. The
temperature was maintained at 90°C. Samples consisted of
two glass disks, two glass disks with tuff, or crushed
glass. Tests for each of the four glasses were run in
duplicate for 7, 14, 28, and 56 ds for a total of 96
tests. Similar tests were run on two blanks containing
EJ-13 water and EJ-13 water plus tuff for a total of 16
tests. When possible, the MCC-1 protocol 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 fluorescence (AF), Cs by atomic absorption spectroscopy (AA), and radionuclides by gamma and alpha counting. The solid test components were analyzed by scanning electron microscopy and associated energy dispersive X-ray analysis, and secondary ion mass spectroscopy.

AMOUNT OF DATA

Tables;

1. Composition of EJ-13 Water (ppm)
2. Composition of SRL Glasses Used in Testing
3. pH and Cation Analyses - Blank Samples
4. Normalized Elemental Mass Loss (Cations) from SRL Glasses U and A
5. Actinide Release from SRL A Glass
6. pH and Anion Analysis - Blank Samples
7. Anion Analysis - SRL Glass
8. Normalized Elemental Release of Actinides after 56 days
9. Results of Gamma Irradiation Experiments Done with R > 3, where
R = gas volume/liquid volume
- C1. Actinide Content of SRL A Glass

Figures;

1. pH Trends Observed for SRL A and SRL U Glass; pH, 6.4 to 7.2, vs time in days, 0 to 60.
2. Average Cation Trends Observed for SRL Glass; Normalized Elemental Mass Loss, 0 to 3 g/m², vs time in days, 0 to 60.
3. Total Am, Pu, and U Release from SRL A Glass, Disk-Only Test; Normalized Elemental Mass Loss, 0 to 3 g/m², vs time in days, 0 to 60.
4. Micrographs of the Surfaces of SRL U Glass
5. Micrograph of the Polished Cross Section of Sample 15
6. EDS Spectra of SRL U Glass Surfaces
7. SIMS Profiles for Selected Elements in Reacted SRL U Glass
8. (NL)_{Li} as a Function of Time^{1/2}; Normalized Elemental Mass Loss of Li, 0 to 3 g/m², vs Time^{1/2}, Days^{1/2}, 0 to 8.

Appendixes;

- A. Preparation of Tuff and Prereacted J-13 Water
- B. Glass Preparation and Test Procedures
- C. Glass Characterization
- D. Test Conditions and Weight Change Data for SRL Glass Testing
- E. Complete Analytical Results for Gamma Irradiation Tests

UNCERTAINTIES IN DATA

The pH values were unstable because of outgassing of the solution. The lowest pH values were accepted. Anion measurements were affected when trapped gas was introduced during the measurement. Errors in Si analysis were possible because of solution reaction with the silicone gasket material. Errors in Si analysis may have been encountered also in using inductively coupled plasma spectroscopy (ICP). Weight changes for disks with tuff present could be in error because it was difficult to remove. Some error and difficulty in interpreting actinide data was due to inhomogeneity in the composition of SRL A. The degree of reaction is likely to vary between tests because the tests were based on a specific surface area-to-volume ratio (R) leading to a varying ratio R. This variance would lead to a different amount of nitric acid being formed in each test and this might affect the degree of reaction. Estimated standard deviations from scatter in the data were: Na, $\pm 7\%$; B, $\pm 4\%$; anions, $\pm 5\%$; and actinides, $\pm 15\%$. Standard deviations for other cations were $< \pm 10\%$. Estimates of accuracy provided by the analysts were ± 3 to $\pm 10\%$ for ICP, $\pm 5\%$ for the anions, $\pm 10\%$ for Cs and $\pm 5\%$ for U. In each case, the inaccuracy depends on the amount of an element present in the sample.

DEFICIENCIES/LIMITATIONS IN DATABASE

The current tests were not run long enough to allow long-term reaction trends to be established or to result in deductions of a mechanism for the glass/water interaction; however, no large increase in the extent of glass reaction was observed due to radiation.

KEY WORDS

Air, Ambient pressure, Basic solution, Borosilicate waste glass, Experimental data, Gamma radiation field, High temperature, J-13 water, Laboratory, ^{237}Np , Leach data, ^{239}Pu , Radiochemical analysis of solution, ^{99}Tc , Weight change.

GENERAL COMMENTS

This report presents data on the influence of gamma irradiation on the reaction of actinide doped borosilicate glass (SRL 165) in a saturated tuff environment. The extent of the glass reaction results from a complicated interplay between the breakdown of the glass matrix, changes in the pH of the solution and interactions that occur with the metal and rock components of the test. 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. This report provides a lot of data on leaching of actinides from a borosilicate glass for a period of up to 56 days. There are a number of factors that the reviewer found problematical. The terms $(NL)_i$ and $(NL)_{wt}$ are used throughout the report but are not defined explicitly. The $(NL)_i$ is presumably the normalized elemental mass loss of species i calculated using the equation, $(NL)_i = (\text{volume of leachant} \times \text{concentration of species } i) / (\text{fraction of } i \text{ in sample} \times \text{surface area})$ while the $(NL)_{wt}$ relationship uses the total weight loss. There is no mention of the method or assumptions used to determine the surface area in the crushed glass experiments. The discussion of $(SA/V) \times t$, (surface area/per volume of solution) $\times$ time, correlation is obscure. The explanation for lack of a correlation is merely a restatement of the observation that none was observed.

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, regarding possible dissolution mechanisms of the waste form, in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

- (b) New Licensing Issues
- (c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Report

Rockwell Hanford Operations, Energy Systems Group,
Richland, Washington.

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

W.J. Anderson, "Conceptual Design of a Waste Package
for Emplacement in Basalt," RHO-BW-SA-280 P, February
1983.

DATE REVIEWED: 10/14/86

TYPE OF DATA

(1) Scope of the Report

The report is a reevaluation of Basalt Waste Isolation Project (BWIP) waste-package conceptual designs based on revisions of the environmental conditions to be expected in the repository.

(2) Failure Mode or Phenomenon Studied

An earlier six-component waste-package design resulted from unrealistic assumptions about the waste-package environment. First, the bentonite/basalt backfill was originally expected to become saturated with water and swell, thereby filling the void space between an overpack and the host rock. The swelling pressure was expected to grasp the waste-package assembly and prevent easy retrieval. Therefore, a mullite sleeve was included in the original design to prevent this swelling backfill from clamping the overpack. However, subsequent examination of the backfill temperature and atmospheric pressure expected to exist between the time of waste emplacement and retrieval operations has shown that the assumed backfill saturation and swelling cannot occur prior to permanent closure of the repository.

Second, the original assumption was that high-oxygen groundwater would be in contact with the overpack throughout its life. Therefore, a material resistant to corrosion in such groundwater would be needed and the original design called for a titanium alloy overpack. Further evaluation and materials testing indicated that contact between the waste container and high-oxygen groundwater will be limited to, at most, a matter of only days or weeks after permanent closure of the repository. During this period, corrosion of a

carbon-steel overpack would be insignificant in relation to the thickness of the material required for structural adequacy. The corrosion allowance for canister/overpack materials should be based on low-oxygen groundwater conditions. The presence of basalt gives rise to low-oxygen in the groundwater.

MATERIALS/COMPONENTS

The early waste-package designs provided for a six-component package consisting of (1) a perforated aluminum borehole liner which surrounded (2) compressed blocks of sodium bentonite/basalt backfill, which, in turn, surrounded (3) a mullite retrieval sleeve. In addition, there was (4) a titanium alloy overpack containing (5) a carbon-steel canister of waste, with (6) graphite buffer material between the overpack and the canister. The temperatures, pressures, and chemistry of the groundwater of the proposed repositories are reexamined in this report.

The materials proposed for the new simplified waste-package conceptual design include stainless steel canisters for commercial high-level waste, carbon-steel canisters for spent fuel rods from pressurized water reactors or boiling water reactors, and a backfill consisting of 75% crushed basalt/25% sodium bentonite.

Because the site is not expected to fill with groundwater and establish hydrostatic pressure until after permanent closure, the need for the mullite retrieval sleeve is eliminated. The Eh of the groundwater in contact with the canister will be controlled by crushed basalt in the backfill thereby limiting corrosion and the solubility of radionuclides and eliminating the need for a buffer material and titanium alloy overpack. Alternate designs are referred to in the report. Changes in the positioning of the waste packages from a vertical to a horizontal orientation are also indicated in drawings of the new design.

TEST CONDITIONS

(1) State of the Material being Tested

Not applicable to this report.

(2) Specimen Preparation

Not applicable to this report.

(3) Environment of the Material being Tested

The report reviews the design values for pressure, temperature, and groundwater composition for the basalt repository. The design value for external pressure on the canister prior to closing is 0.15 MPa. The waste package storage boreholes are not sealed from the access tunnels in order to permit ventilation for operating personnel. After permanent closure, groundwater seeping into the backfill will pressurize the void between the borehole wall and the canister. The pressure on the waste canisters will rise until it reaches the ambient hydrostatic pressure, which is the design pressure used after permanent closure. The relationship between repository depth and hydrostatic pressure at the Hanford Site is given by:

$$\text{Pressure (MPa)} = 0.063 + 0.00979 \text{ Depth (meters)}$$

<u>Repository</u>	<u>Depth</u>	<u>Design Pressure</u>
Umtanum basalt flow	1,128 m	11.1 MPa
Middle Sentinel Bluffs basalt flow	945 m	9.3 MPa

The ambient temperature of the host rock used for design is based upon repository depth:

$$\text{Temperature (°C)} = 14.83 + 0.038 \text{ Depth (meters)}$$

<u>Repository</u>	<u>Depth</u>	<u>Ambient Temp.</u>
Umtanum basalt flow	1,128 m	58°C
Middle Sentinel Bluffs Basalt flow	945 m	51°C

Soon after emplacement of the waste, temperatures at the borehole surface exceed 100°C due to the decay heat of the radioactive waste. Groundwater seeping in before closure will vaporize quickly and displace the air in the boreholes with water vapor in a day or two. The temperature will continue to rise, reaching a peak of 235°C between 5 and 10 years after emplacement for commercial HLW from processed spent fuel. At the time of permanent closure, the rock temperature will decrease to the range of from 170 to 200°C. Vaporization caused at these temperatures should prevent contact between the canister material and liquid groundwater until after permanent closure. After closure, the temperature will continue to decrease reaching 100°C in about 400 years. At that time, the heat generation rate of the waste will be sufficiently low that the waste temperature will be within 5°C of the borehole surface temperature.

Groundwater composition in the Grande Ronde basalt flows is of low ionic strength (about 20 milliequivalent total cations plus anions). Sodium is the dominant cation, with varying amounts of minor cations, potassium, calcium, and magnesium. The chemically dominant anions are chloride and sulfate, and also a relatively high

fluoride content compared to potable water. Dissolved silica in groundwater in contact with basalt follows a solubility limit for amorphous silica as a function of temperature. The chemical behavior of the groundwater is characteristic of that for either sodium chloride or sodium chloride - sodium sulfate solutions.

Tests of the groundwater chemistry indicate that the ambient Eh and pH conditions favor limiting the solubility and, therefore, the transport of many radionuclides. At basalt temperatures of 58 and 300°C, the groundwater pH is nominally 9.4 and 7.8, respectively. Results of hydrothermal tests show that reaction between basalt and groundwater to remove dissolved oxygen occurs rapidly. In tests at 150°C and 300 bars, the dissolved oxygen content of simulated groundwater in contact with crushed basalt decreased from 8.5 mg/l to 0.4 mg/l in 190 hours. Reasonable extrapolation of the available data indicates that the dissolved oxygen would be $<10^{-3}$ mg/l in about 600 hours. Other data indicate that the Eh of groundwater in the repository near field will return to the Eh of ambient Grande Ronde basalt groundwater in a few months after permanent closure.

METHODS OF DATA COLLECTION/ANALYSIS

This report is a review of the BWIP waste-package design as of March 1983. It provides some of the repository characteristics (e.g. temperature and pressure) as well as calculated results for time-temperature relationships in the repository which are needed for evaluating the choice of container materials in the future. The review of the site-specific conditions for the basalt repository permits the design of a simplified waste package.

AMOUNT OF DATA

There are three figures. Figure 1, titled "Early NWRB Waste Package Emplacement Concept," is a diagram of the original waste package design. Figure 2 gives temperature estimates for the emplacement borehole surface over time for HLW from processed spent fuel. It is titled "Commercial High-Level Waste Package Temperature Estimates," and has a temperature scale from 50 to 400°C and a time scale from 10^0 to 10^4 years. Figure 3, titled "The NWRB Reference Commercial High-Level Waste Package Conceptual Design," is a diagram of the revised waste package design.

There are two tables. Table 1, titled "Waste Package Dimensions - Conceptual Design," gives dimensions for the carbon-steel canister, backfill thickness and borehole diameter for three types of waste. Table 2, titled "Waste Package and Excavation Costs," lists the estimated costs

for the parts of the waste package and the excavation for four designs and compares the totals for each design.

UNCERTAINTIES IN DATA

Not given by the author.

DEFICIENCIES/LIMITATIONS IN DATABASE

Not given by the author.

KEY WORDS

Conceptual design, Bentonite, Basalt, Stainless steel canisters, Carbon steel canisters, Grande Ronde basalt.

GENERAL COMMENTS

This NBS review contains a summary of only the contents of this technical report. It contains neither critical commentary nor analyses.

RELATED HLW REPORTS

These reports are cited in the bibliography.

RHO-BWI-LD-23
RHO-BW-ST-25 P
DOE/RL 82-3
RHO-BW-CR-136 P/AESD-TME-3142

APPLICABILITY OF DATA TO LICENSING

No specific issues are addressed here.

Appendix B. Materials Characterization Center Activities

MCC SECTIONS ACCORDING TO PROGRAMS SERVED

The MCC reports are presented in eight separate sections, each on a separate topic. These sections are as follow:

- A. MCC Program Administration
- B. MCC Quality Assurance
- C. MCC Support to the Office of Geologic Repositories (RW-20) (OGR)
- D. MCC Support to the Salt Repository Project (SRP)
- E. MCC Support to the Basalt Waste Isolation Project (BWIP)
- F. MCC Support to Defense HLW Technology Program (DP-12)
- G. MCC Support to the Transportation Technology Center
- H. MCC Support to the West Valley Demonstration Project (WVDP)

MCC Program Administration

The section on MCC Program Administration covers: (1) program planning, (2) interactions with the Materials Review Board (MRB), (3) ASTM interactions, (4) variance explanation and (5) capital status. Names of individuals are used in the monthly MCC reports, and are given in this report to provide the most current contacts in areas.

Program Planning

Program planning includes the appointment of task leaders for the MCC tasks. The MCC task groups and leaders are shown in Figure 1. The May 1986 monthly report stated that C. A. Knox would be the task leader of Spent Fuel Operations and P. J. Turner would be task leader for Repository Interactions Testing. At the end of August 1986, John E. Mendel left the position of Manager of the MCC and stepped into the dual role of Chief Scientist and leader for Task 3, Spent Fuel Characterization. Former task 3 leader, J. O. Barner, left to be section manager of the Reactor Technology Center and will continue to act as a consultant to the MCC. Mr. M. R. Kreiter became manager of the MCC on September 2, 1986.

In June 1986, D. E. Clark of the Salt Repository Project (SRP) visited MCC to discuss MCC technical support of the SRP. J. E. Mendel met with C. C. Chapman and J. M. Pope of West Valley Nuclear Services (WVNS) and R. G. Spaunburgh of the N. Y. State Development Agency at the Pacific Northwest Laboratory (PNL) to discuss MCC work on the West

Valley Program. A walk-through safety inspection was conducted in buildings 324 and 325.

In July, the fiscal year (FY) 1987 Technical Program Plan for SRP was submitted to R. O. Izatt. The Basalt Waste Isolation Project (BWIP) approved an additional \$30K to complete the FY 1986 work.

In August, a draft FY 1987 support work statement for the West Valley Demonstration Project was submitted to L. R. Isenstatt. The 1986 statement-of-work review for work in support of the Transportation Technology Center was sent to and approved by Marcella Madsen. A request to participate in the chairman round robin of a Repository Evaluation System Evaluation Test was submitted to J. D. White.

In September, M. R. Kreiter met with J. C. Haugen, DOE, Chmn., Materials Information Organization (MIO), M. J. Steindler, Chmn., MRB and D. E. Clark (SRP) at Argonne National Laboratory (ANL), Columbus, Ohio to discuss FY 1987 MCC programs and interfaces. M. R. Maslar, West Valley Demonstration Project (WVDP) and W. L. Delvin of Westinghouse Hanford Co. conducted an audit of the MCC work for the WVDP. M. R. Kreiter gave a program overview to French visitors.

In October, a revised FY 1987 statement was submitted to C. C. Chapman of West Valley Nuclear Services. The statement of work for the Defense High Level Waste Technology Center was submitted to D. A. Turner for approval. A request for continued support of the brine stability study was submitted to K. K. Wu. M. R. Kreiter conducted program reviews and laboratory visits throughout MCC.

In November, the FY 1987 Technology Program Plan DOE/MCC support to the Office of Geologic Repositories was submitted to Joel Haugen (MIO). Mr. Kreiter requested carry-over funds for BWIP projects, L2OER and L2GIR, from G. T. Harper.

In December, M. R. Kreiter completed and submitted a draft of the updated MCC Management Procedures Document to the DOE Richland Operations Office. A request for comment and review of the SRP-BNL-1 Standard Test Method for Salt Irradiation Testing was submitted to D. P. Moak, ONWI.

Interactions with the Materials Review Board (MRB)

MCC interaction with the Materials Review Board included submission of the final "mats" for MCC-7P for final approval prior to inclusion in the Nuclear Waste Materials Handbook. Data reports submitted in mat to the OTC-MRB

for MRB approval were MCC-D5, One Year Leach Test Data for ARM-1 Glass and MCC-D6, Agitated Powder Leach Testing of ARM-1 Reference Glass using the MCC-3S Test Method. In June, a letter (MCC0445 "Schedule for Resubmission of MCC-1 and Outline of Anticipated Changes") describing MCC's plans for a mini-round robin test on the effect of specimen surface finish variations on test results was sent to the MRB Chairman. In July, a request for approval of BWIP/MCC 105.1S Test Method (MCC:0465) was sent to the MRB Chairman (OTC/MRB). J. E. Mendel met with C. J. Haugen, S. Vogler, W. B. Seefeldt and M. J. Steindler to discuss a draft white paper, "Functions of the Nuclear Waste Materials Characterization Center".

American Society for Testing and Materials (ASTM) Interactions

MCC interactions with ASTM have been primarily with the ASTM Committee C26 on Nuclear Waste. D. N. Merz has been active on the ASTM committee for developing accelerated testing procedures (C26.07). W. M. Bowen represents MCC on statistical issues and contacted Roy Morrow regarding suggested revisions identified in the ASTM draft standard, "Determining Elements in Waste Streams by Inductively Coupled Plasma(ICP)-Atomic Emission(AE) Spectroscopy".

In June, M. D. Merz attended an ASTM C26.07 task group meeting on accelerated testing in New York. T. Johnson of the Nuclear Regulatory Commission (NRC) provided the task group with ASTM E632-82, "Standard Practice for Developing Accelerated Tests to Aid Prediction of the Service Life of Building Components and Materials," and it was decided that the format of the ASTM E632-82 would be appropriate for use with the ASTM C26.07 document. It was decided that the ASTM would take an active role in developing the document, that the document should be aimed at developing a consensus to help in repository licensing, that the document should be pertinent to repository project technology as perceived by the investigators involved, and that the activity should promote understanding on a national and international scale. An agenda, including identification of degradation modes and associated data bases, was set for the Seattle, Washington, meeting. A letter was sent to R. Stein (DOE Headquarters) regarding ASTM C26 activities on accelerated testing and a letter soliciting input for the Seattle meeting was sent to the proposed repository projects.

In July, it was reported that J. L. Daniel, D. N. Merz and W. M. Bowen participated in the ASTM C26 meeting in Seattle, Washington. J. M. Daniel was to make revisions to the standard method, "Analysis of Aqueous Leachates from Nuclear Materials Using Inductively Coupled Plasma" and will serve as co-editor of the ASTM Special Technical

Publication covering the Plasma Analytical Methods Symposium (STP-981). D. N. Merz prepared and distributed a preliminary draft revision of ASTM E632-82. The group discussed definitions of predictive testing and accelerated testing.

In August, it was reported that W. M. Bowen reviewed the procedure, "Standard Guide for Qualifications of Laboratory technicians," and found that the document. He agreed to collaborate with the author, D. E. Sandbag, to rewrite the procedure. W. M. Bowen also recommended changes regarding statistical issues in the procedure, "Determining Elements in Waste Streams by ICP-AE Spectroscopy".

In September, T. Thornton (SRP) sent input for the draft standard on accelerated testing. Additional input was expected from BWIP and the Nevada Nuclear Waste Storage Isolations (NNWSI). Plans for the ASTM task group C26.07 meeting continued at the Materials Research Society Meeting in Boston, Massachusetts. A revised draft was distributed to the task group. Written comments were received from BWIP. K. Chang (NRC) provided comments on the draft.

Quality Assurance

S. L. Sutter, Quality Assurance Coordinator, reported on quality assurance coordination. He stated that Finding No. 2 of the PNL audit regarding issuance, revision and distribution of the Nuclear Waste Handbook has not been resolved.

Records generated under BWIP support, including those for Approved Testing Materials, ATM-9, ATM-11 and ATM-101, development and historical records and records supporting TDM-1 (crushed depleted uranium oxide) preparation and management were prepared for transfer. Efforts are underway to develop a lifetime storage system for MCC records. S. L. Sutter met with D. L. Alamia and W. E. Brooks of PNL records management to plan the records transfer. W. E. Brooks will be responsible for this work, and will deliver the records for lifetime storage in the National Archives. Records are to be sent through PNL to the sponsors of the work. Records (1343 pages) were sent to BWIP. Plans were made to transfer SRP records biennially. Some problems have been identified with PAP-1701 Rev.1, Research Records System.

Software control procedures (SCPs) were presented by PNL in an all-day meeting. Eighteen staff members were trained in revised software procedures in September and training of numerous staff members occurred at other

times. New staff members were trained in the scope of the project.

An internal audit of the MCC conducted in August by the PNL Quality Assurance Department resulted in three findings and one observation. The findings were: (1) the project staff is not implementing fully the approved software control procedures (2) procedures for obtaining services from others are not being implemented fully and (3) reference material is not being controlled in accordance with PAP-801. The observation was that incomplete records were transmitted to the PNL record center. Suggested corrective actions included briefing eight staff members on software control procedures, briefing eleven MCC material custodians and material users on PAP-801 Rev. 2 which relates to reference material control, completing records and sending them to PNL, and on-the-job records training for other personnel. Corrective actions for this internal audit were completed in December.

Sandia National Laboratories and DOE-AL personnel audited MCC activities for the Transportation Technology Center. There was one finding and one observation, both related to documentation of Westinghouse Hanford Company (WHC) standards. Proposed corrective action will require Hanford Engineering Development Laboratory (HEDL) to document standards used by their laboratories.

S. R. Maslar, WVDP, and W. L. Delvin, WHC, reviewed projects for quality assurance and found problems with laboratory record book entries. An audit of work performed for WVDP indicated a delay in issuing a deficiency report. Observations were: (1) WVNS technical advisories are not controlled (2) there was no evidence of a technical review and (3) the MCC is using procedures not specified in their QA Plan, NWTP-1 Rev.0. Corrective actions included completing deficiency report DR-86-92 and revising PAP-1502 to allow QA representatives to initiate deficiency reports, establishing a printout of the technical advisory log, writing memos identifying technical reviewers and establishing a system to provide evidence of a technical review, and the preparation of a revised QA Plan WT-002.

W. M. Bowen attended the 40th Annual Quality Congress Meeting of the American Society for Quality Control. W. M. Bowen and J. L. Daniel reported that initial analysis was conducted on data for ATM-1 and NBS Standard Reference Material SRM-1411 glasses and they found general satisfactory agreement except for the silicon analyses.

J. L. Daniel and B. L. Neth reported action on two service procedures and four technical procedures, including one service procedure sent to the Office of Document Control for issue with SRP/BWIP exception and three technical procedures undergoing BWIP review. The SRP disapproved the revised MCC QA plan in September. Other actions on service and technical procedures were reported during this eight-month period.

PNL is committed to DOE to upgrade Quality Assurance Manuals, PNL-MA-65 (to be designated PNL-MA-70) and PNL-MA-60. MCC reviewed thirty procedures, made comments and submitted them to P. H. Bruce, PNL Procedure Coordinator, completed in September.

MCC Support to the Office of Geologic Repositories (RW-20) DB-05-11-03

Acquisition of Spent Fuel Approved Testing Materials (ATMs).

Acquisition of Spent Fuel Approved Testing Materials is headed by J. O. Barner and C. A. Knox. J. E. Mendel moved into this area in September when J. O. Barner left. A request for proposal (RFP) was sent to Northeast Utilities Service Company (NUSC) for the acquisition of ATM-107. This contract would be for approximately 205 fuel rods of stainless-steel clad fuel from the Connecticut Yankee Reactor. NUSC is unwilling to provide fuel rods from two different assemblies and would like to provide two total assemblies. Two total assemblies results in a large quantity of the relatively moderate priority stainless steel clad fuel and would require too much storage space. NUSC has been asked to provide another assembly, and this request is being evaluated. A RFP was issued for a shipper to transport ATM-107 from Battelle-Columbus to Hanford, Washington.

NNSWI, BWIP, ONWI and the Office of Crystalline Repository Development (OCD) concurred on the characterization plan for ATMs-103 through 106. The status of the major characterization equipment was described in each monthly report. This equipment includes a twenty-four fuel rod storage container, a fission-gas sampler/rewelder, a fuel rod gamma scanner, a fuel rod handling strongback, fuel rod retrieval equipment and a fuel rod sectioning saw. Cold testing is complete for the twenty-four fuel rod storage container, the fission-gas-rewelder, the fuel rod gamma scanner and the fuel-rod handling strongback.

Fuel rods from ATMs-103, 104 and 106 were retrieved to begin characterization. Some time was required to repair the manipulator. The September report stated that

insertion of a fuel rod in the gamma scanning system for in-cell checkout resulted in identifying two problems: (1) the software program needed modification and debugging and (2) background radiation in the hot cell was high and the collimator needed additional lead shielding. Additional shielding was put on the collimator. The October report stated that the contractor assigned the computer specialist to another project, making it necessary to find another computer specialist to complete the software.

The fission-gas sampling system was checked out using a pressurized dummy fuel rod. This sampling-system also was used to train personnel. Comments were received from the repositories regarding fission-gas sampling procedures.

The final five ATM-101 fuel rod segments were transferred to WHC, 324 Building for gamma scanning. Gamma scanning had been completed on six segments and these were returned to the D-cell from WHC. Two full-length gamma scans were completed on ATM-103 in December. The acceptance test plan was prepared, approved and completed, and the software and control forms were completed. Route scanning can begin as soon as the technical procedure is approved. 106 g of ATM-101 were crushed to a size of -60+45 mesh and sent to BWIP in August.

An ATM cutting saw was built. The documentation for the computer software was completed and a tape back up system is being procured.

Reports

A Shielded Analytical Instruments Workshop, held at PNL by the MCC, was reported on by J. L. Daniel and H. E. Kjarmo. The summary report entitled "Examination of Radioactive Materials Using Electron Beam Instruments and X-Ray Diffraction" was completed. A list of all U.S. laboratories using these instruments was distributed to the contributors for review. A report describing MCC shielded analytical instruments entitled "Shielded Analytical Instruments for Complete Characterization of High Radioactive Materials," prepared by J. L. Daniel and H. E. Kjarmo, was cleared, approved, published and distributed (PNL-5862). "Fabrication and Characterization of MCC Approved Testing Material-ATM-9" by J. W. Wald was cleared and approved for publication (PNL-5577-9). "Fabrication and Characterization of MCC Approved Testing Material, ATM-11" by J. W. Wald and J. L. Daniel was cleared, approved and published.

Test Method Development

M. D. Merz and D. W. Shannon worked on revising TP-6 Developing MCC Test Methods. This document is in the approval process.

The October report stated that P. J. Turner attended the planning meeting in Karlsruhe, FRG, for Chairman Round Robin Repository Interactions Testing. Reference glass, vessels, etc. will be supplied to the chairman who will pass the materials on to the participants. Granite was chosen to be the host rock for the U.S. tests. After initial tests will be for 28 days, the committee will reconvene in March or April 1987 to compare results.

Analytical Methods Development

J. L. Daniel attended a meeting of the Federation of Analytical Chemistry and Spectroscopy Societies (FACSS) in St. Louis, September 29 to October 1, 1986. Capabilities of the ICP-MS analytical instrument were discussed. This instrument has been selected for the repository project and MCC programs. Also discussed at this meeting were state-of-the-art analytical instruments for analysis of solid materials such as waste glass which do not require prior solution of the sample. Laser ablation and/or ionization methods were of interest.

Spent fuel particles (N9C-C, H. B. Robinson ATM-101) were examined by L. E. Thomas (HEDL) using transmission electron microscopy. Central and edge specimens had high densities of fission product aggregates (precipitates and gas bubbles). Precipitate diameters were 6 nm at the edge, 15 to 20 nm in the center and 100 nm in the grain boundaries. The metals, molybdenum (Mo), ruthenium (Ru), technetium (Tc), palladium (Pd) and ruthenium (Rh) were identified. For Xenon (Xe), 10 wt. percent was found, suggesting the formation of solid xenon.

J. E. Mendel had discussions with J. W. Roddy, Oak Ridge National Laboratories (ORNL), regarding spark source mass spectrometry and associated quality assurance requirements for this work.

Radiochemical analyses will be conducted for ^{93}Zr , ^{135}Cs , and ^{235}Am in the fuel, for ^{129}I on the cladding and ^{59}Ni , ^{63}Ni , ^{94}Nb , and ^{14}C in the hardware. R. J. Guenther and D. E. Blahnik prepared and issued a work statement for this. J. E. Mendel and R. J. Guenther recommended ordering the examination of the first three fuel rods as follow: ATM-103 (moderate burnup, low release), ATM-106 (high burnup, high release) and ATM-104 (high burnup, low

release). R. J. Guenther presented a summary of the MCC spent fuel work at a spent fuel workshop in Wakefield, Massachusetts.

Aqueous Dissolution of Nuclear Waste Glasses

The review of published aqueous dissolution studies of nuclear waste glasses continued. Preparation of a draft report is in progress. The report summarizes literature reviewed, compositions, test conditions and dissolution properties of glass waste forms.

Data Handbook Development

This task headed by K. M. Krupka. The literature review of published aqueous dissolution studies of nuclear waste glasses was continued during May.

"Strawman" Spent Fuel ATM Selection Criteria

(J. E. Mendel and R. J. Guenther). A draft of the "Spent fuel Working Group (SFWG) Program Plan" was submitted to DOE Headquarters for comment. Selection criteria are divided into four categories.

Planned Work

Planned work and work accomplished are described. This work included installing spent fuel characterization equipment in B-cell and D-cell, 324 Building, and the initiating characterization and distribution of spent fuel ATMs 103 through 106. The September report indicated the initiation of identifying an MCC approach to reference materials for container weldments. A revised draft of "Recommended Practice for Developing Accelerated Tests to Aid Prediction of Performance of Waste Package Materials for Geologic Disposal of High Level Nuclear Waste" will be prepared. The November report states that fuel disassembly equipment will be installed in B-cell.

Work on the ASTM version of MCC-1 continued; the ASTM version will be revised to be consistent with the Handbook version. The MCC-1 document will be ballotted again in the ASTM C26.07 subcommittee. A response was made to the ASTM C14.03 Committee on Chemical Properties of Glass, and work continued on ASTM C26.07 on accelerated testing.

Support to the Salt Repository Project

Coordination of Technical Peer Review of SRP-WPP Procedures.

(M. D. Merz, E. P. Simonen, K. Wiemers and R. A. Craig). Revisions were completed on SRP-WPP-53, Static Load Crack Growth Testing with Modified Wedge Opening Load (MWOL) Specimens and on SRP-WPP-57, Conducting U-Bend Tests. These methods were submitted to SRP in May. A draft of SRP-WPP-31, "Determination of Different Oxidation States of Plutonium and Neptunium Using Solvent Extraction Techniques," is being reviewed. The MCC submitted the Brookhaven Test Method on Salt Irradiation to SRP and gave a schedule for its development and revision which begins on May 30 and finishes on August 29. R. A. Craig revised SRP-WPP-43, "Hydrogen Ion Concentration of Brines" and began revision of SRP-WPP-44, "Standard Test Method for Total Base Determination in Natural Rock Salt" and SRP-WPP-45, "Hypochlorite Ion Determination in Solutions of Irradiated Salts." Work continued on SRP-WPP-41, "Spent Fuel Leaching", and SRP-WPP-48, "Interactive Leaching of Nuclear Waste Glass in Brines." Plans are to revise SRP-WPP-39, "Standard Practice for Preparation and Analysis of Simulated Permian Basin Brines and Dry Salts," SRP-WPP-43 and SRP-WPP-48, and to submit SRP-WPP-31 to SRP for the first time.

Fourteen SRP test methods were identified for MCC coordination. MCC recommended that SRP-WPP-40/MCC-202, "Radiolysis of Salt Brines," be suspended pending more experience with the test. Approval was requested from SRP to forward SRP-WPP-41/MCC-14.7, SRP-WPP-44/MCC-204 and SRP-WPP-45/MCC-206 to the MRB for approval. Nine of the packages await SRP approval to go to the MRB. Work continued on SRP-WPP-31/MCC-203 and SRP-WPP-39/MCC-205. SRP-WPP-18/MCC-102 was ballotted through the ASTM C26.07 committee and passed.

In September, a tentative agenda for the electrodes seminar (pH and Eh measurements in brines) to be held at PNL on October 10 was completed. This workshop assists the MCC by reviewing pH and Eh measurements, the uses of pH and Eh in modeling, and MCC plans for developing electrodes for measurements in brine. Experiments will be set up for making pH measurements in brine. The December report stated that Hastelloy C276 will be used for the autoclave. Procurement of capital items is behind schedule due to the lengthy bid and review process.

Planning for the MCC workshop on localized corrosion and stress corrosion to be held December 16-17, 1986, was started by D. Shannon and M. D. Merz. Materials emphasis was to be on low-carbon cast steel and alternate

materials. To date, the SRP test program has indicated no SCC in these steels, and the workshop attempted to recommend how these findings can be supported for licensing. In November, it was reported that four experts were invited to this workshop. The invitees were D. Duquette (Rensselaer Polytechnic Institute), W. W. Gerberich (University of Minnesota), J. Payer (Case Western Reserve University) and H. Birnbaum (University of Illinois). The workshop was listed as one on stress corrosion cracking (SCC) rather than local corrosion. The workshop was held in December and a summary letter report will be sent to SRP in March 1987.

MCC Support to the Basalt Waste Isolation Project

BWIP/MCC-14.4 Test Method Development

(P. J. Turner). The two remaining experiments were terminated. Experiment number 1 ran for 458 days and contained ATM-3 (^{237}Np) glass, basalt, and Gr-4 groundwater. Experiment number 2 ran for 613 days and contained ATM-4 (^{239}Pu) glass, basalt and Gr-4 groundwater. Residues in the two vessels were different in physical appearance. ATM-3 glass was more dissolved than ATM-4 and formed heavier deposits on the vessel cover. SEM, TEM and microprobe analysis are being performed on the residue and deposits, and ICP, IC and AA analysis are being performed on the leachate.

Work is in progress to relocate the ovens from the 3720 facility to the new leaching facility in 325 building. Laboratory facility work in preparation for the BWIP/MCC-14.4 test was completed in September.

An IBM PC XT and software were ordered for the Lab 500/504 facility for data storage and manipulation. Test condition monitoring by computer is being considered.

Operating instructions for the Parr vessel is being included as an appendix to BWIP/MCC-14.4.

W. M. Bowen prepared a letter report, "Statistical Considerations of the Adequacy of the BWIP/MCC-14.4 Test Plan" and submitted it to BWIP.

An FY 1986 run plan for BWIP/MCC-14.4 "Test Method Development using ARM-1 Reference Glass" and an FY 1987 run plan for BWIP/MCC-14.4 "Test Method Requirement Using ARM-1 Reference Glass," ATM-11 "Savannah River Glass" and ATM-10 "West Valley (Commercial) Glass" were sent to BWIP for review. MCC recommended delaying sending the test method to the MRB until currently scheduled testing is completed in 1987.

In October, developmental testing began on the effect of temperature on the rate of attaining steady-state. Tests are performed at 150°C, 175°C and 200°C and sampled at temperature using the dip-leg sampler. The November report indicated that 7 and 14 day samples were taken but the data has not been compiled. Data analyzed in December indicate a 30 percent increase in major ion concentrations in the 175°C tests as compared to the 150°C tests. The 200°C tests were discontinued due to difficulties in sampling at temperature. Preliminary results from 458 day ^{237}Np and 613 day ^{239}Pu doped tests of solids indicate that the glass was not dissolved in either case. Tests to study effects of temperature and rotation rate are planned before hot testing is started. The duration of all developmental testing is 28 days.

Development of Basalt-Specific Corrosion Tests (M. D. Merz, M. Lewis and F. Gerber).

The term "benchmark tests", indicating the workability and reproducibility of a laboratory test, was changed to "Reference Condition Tests" to avoid conflict with NRC terminology which uses benchmark as a test of computer software.

Work continued in May on BWIP/MCC-105.4, "Flowby Corrosion Reference Conditions Test Report." This work (30 and 120 day, 100°C) was completed in June and it will be submitted in early July to BWIP for review and approval to submit it to the MRB. This work will be used to support the workability of the BWIP/MCC-105.4 "Flowby Test Method."

BWIP/MCC-105.1, "Static Corrosion Reference Conditions Test Report" (300 days at 100°C) was revised after the BWIP review and will be submitted to the MRB in support of BWIP/MCC-105.1, "Static Corrosion Test Method," which was submitted to the MRB in July. The MRB review, consisting of 36 statements of needed changes, was received in October, and the response is being evaluated. Items to be resolved included the removal of a tentative groundwater composition specification and additional specifications on the packing material. Information was requested from the Reference Conditions Test Reports on precision and accuracy of the method.

Changes were made in the BWIP/MCC-105.1 "Static Corrosion Test Method" in response to the OTC/MRB review and to discussions with R. P. Anantatmula. Changes involved weighing the pressure vessel to check for water loss and to allow tolerances on allowable weight changes after specimen cleaning, proportional to the specimen surface area. The revised test method was sent to BWIP for concurrence on resubmission to the MRB.

The draft of BWIP/MCC-105.5, "Air/Steam Corrosion Test Method" was reviewed by John Mendel and revised. This test includes a statistically designed thimble arrangement by W. M. Bowen where five test coupons are embedded in packing material inside each of twelve thimbles to allow estimation of effects of ovens, test duration, coupon size and location within ovens. A run plan for air/steam test with packing was prepared. The run plan for BWIP/MCC-105.5 "Air/Steam Reference Conditions Tests" was approved in June, and tests were started. Discussions with R. P. Anantamula (BWIP) resulted in an agreement for MCC to run the tests without packing and with packing. In June, the BWIP/MCC 105.5 "Air/Steam Corrosion Test Method" was completed and submitted to BWIP for review and concurrence to send to the MRB.

In November, all experimental work on the Development of Basalt Specific Corrosion Tests was stopped as a result of reduced funding and efforts were made to discontinue the tests in an orderly manner and complete the records. Specimens will be archived and records given to RHO-BWIP.

In September, W. M. Bowen and B. A. Pulsipher conducted statistical analyses of corrosion data from two laboratories, MCC and HEDL, for BWIP/MCC-105.1, 105.4 and 105.5 tests to provide reproducibility information on interlaboratory testing. These analyses were used for preparation of a paper for the Materials Research Society.

An abstract entitled "MCC Corrosion Tests at Reference Testing Conditions for A27 Cast Steel in Hanford Ground Water" by M. D. Merz and M. Lewis was prepared for the Materials Research Society Meeting in Boston, Massachusetts, in December. The abstract was submitted for clearance through PNL and BWIP. R. Wang was added as an author and the first draft was prepared in September.

The orders for the new dip-leg covers for the vessels have been placed with the Parr Instrument Company. Parr will need eight to ten weeks to supply the covers. Testing will begin when facilities are completed in 325 Building and when the dip-leg covers are received. The dip-leg covers were received in June. Preliminary testing, begun in August, showed that vessels with the dip-leg and rupture disk maintained their seals under hydrothermal conditions. This test will determine effects of sampling at temperature versus cooling before sampling. Sampling via the dip-leg at temperature will eliminate any possible error due to "quenching effects".

Analyses of specimens from BWIP/MCC-105.4 (300 days at 100°C) and BWIP/MCC105.5 (30 and 120 days, 300°C) were completed, and reports will be made in September. A

statistical analysis of the 30-day penetration test and penetration rate showed that oven 2 specimens had higher penetration than oven 1, chamber 2 had higher penetration than chamber 1, and the position of the coupon on a rod had a significant effect on penetration and penetration rate. Coupons in the rear of one oven had higher penetration than did the coupons at the front of the oven, but this trend in penetration did not hold for both ovens. There was no effect due to the height placement of the specimen in the ovens. Smaller coupons had higher overall penetration than the large coupons.

Major Problems and Action Taken.

Initiation of BWIP/MCC-105.1 "Radiation Corrosion Tests" is behind schedule due to delays in getting heaters for the test vessels. The September report indicated that corrosion activities for 1987 would be reduced by \$157K and this reduction would not delay completion of experiments in progress. FY 1987 funding for development of Basalt Specific Corrosion Tests is \$150K with \$42K designated for management and technical support. Air/Steam tests for 300 days of 64 specimens without packing material are 79 percent completed, 120 day tests of 64 specimens with packing material are completed, 300 day tests with 64 specimens with packing are 40 percent completed. An additional \$60K would be needed to complete this work. BWIP work will be continued through April at the current spending rate of \$14.5K for experimental work and \$6K for management. Because the anoxic chamber is not able to attain the required anoxic condition of <50ppm, a new chamber was requested in the FY 1987 budget.

MCC Support to the Defense HLW Technology Program (DP-12) AR-05-15-10

WBS 2.5 Waste Acceptance Specifications Data Acquisition Plan (MCC)

a. Long Term Chemical Durability Testing (G. B. Mellinger). The test description drafted by Barkatt and Macedo of Catholic University of America (CUA) was reviewed and sent to CUA for their response. In this test procedure, it is important to maintain a CO₂ free atmosphere within the polytetrafluorethylene fluorethylene (PTFE) leach containers. Stainless steel chambers were fabricated for maintaining this atmosphere in the ovens. ATM-11, a doped SRL-165 glass will be tested, and ATM-18, the doped HWVP glass will be tested when it is ready.

In October, J. L. Daniel and S. O. Bates started planning for the fabrication, characterization, documentation and distribution of ATM-18.

The September report stated that the paper entitled "Standardized Test Methods for Use in Waste Compliance Testing Within the Department of Energy's Waste Acceptance Process" by G. B. Mellinger was accepted for presentation at the Dec., 1986 meeting of the Materials Research Society. This paper proposes that standardized test methods would be used when performing tests required for demonstrating compliance with the DOE's Waste Acceptance Preliminary Specifications. Figures 2, 3 and Table 1 were taken from this presentation and show the Waste Compliance Portion of the DOE Waste Acceptance Process, the Advantages of the Use of Standardized Tests in Waste Compliance Testing and the Waste Acceptance Preliminary Specifications.

G. W. Bowen drafted a revised statement of precision and bias in October for potential inclusion in the MCC-1 Static Leach Test. This will include data from the round robin tests.

The MCC-3 Agitated Powder Leach Test Method was submitted to the MRB for approval. Approval was not granted and numerous comments were received. These comments will be addressed and the test method resubmitted to the MRB. The November report stated that discussions with waste producers revealed that each producer would apply this test in a different manner, e.g. different sized fractions of crushed glass, different test periods, and CO₂ free test conditions. The different needs of these producers will be considered as the revision is made for resubmission to the MRB.

b. Handbook Data Packages (J. L. Daniel and W. M. Bowen). Two data packages, MCC-D5 and MCC-D6 based on the MCC leach tests of ARM-1 were sent to the MRB in May.

c. Comprehensive Data Base for the Nuclear Waste Materials Handbook (J. L. Daniel, C. H. Kindle and K. M. Krupka). This is a multifunded task in support of the WVDP, the OGR and the Defense HLW Technology Programs. All comprehensive data base activities are reported in this section.

The data base will have the following outline.

- I. Introduction
- II. Scope: Material Composition Ranges, etc.
- III. Conversion Constants, Symbols, etc.
- IV. Glass Waste Form Properties
 - A. Leaching

- B. Thermal Stability
- C. Thermal and Processing Properties
- D. Chemical Compatibility with Canister
- E. Bulk Physical Properties
- V. Model Availability for Waste System
- VI. Canister and Overpack Material Properties
- VII. Spent Fuel Properties
- VIII. Integrated Test Performance Data

d. Glass Durability/Hydration Energy. The application of the hydration energy model (advocated by C. M. Jantzen and M. J. Plodinec, Savannah River Laboratory) to glass durability continued using existing data on glass durability and composition. The May and June reports stated that this expanded set of N-waste glasses (95 in number including West Valley 205 and the European UK209 glasses) is consistent with the model and confirms similar behavior of N-waste glasses with other man-made and geologic glasses. The model estimated an order of magnitude change of durability for every 5.7 kcal/mole change in hydration energy. The less negative the free energy of hydration, the more durable is the glass. This model cannot be used when energy changes are less than 5 kcal/mole and would not be useful to fine tune a glass formulation to improve its durability. A letter report entitled "Glass Durability as It Related to the Free Energy of Hydration" was prepared by C. H. Kindle and G. M. Faldetta.

C. H. Kindle visited the WVDP in May to present the status of the data base.

e. MRB Approval of MCC-1 Static Leach Test. MCC received the MRB's comments on MCC-1 in May. The Procedures Panel gave full approval to MCC-1 but the number of yes votes was the bare minimum required and the yes votes were accompanied by comments indicating MRB concern. The concerns related to surface finish, oxygen fugacity and the use of the method for radioactive samples. A task force was organized to address these concerns as well as differences between this version and ASTM version. Recommendations were to have the method focus on glass testing, allow reuse of PTFE containers, specify stainless steel or fused quartz containers for gamma emitting samples, define a specific sawing procedure to result in a more reproducible surface and to work on resolving differences between the MCC and ASTM versions of the test. The test was sent to Savannah River, CUA, ANL, and Lawrence Livermore National Laboratory (LLNL) for their review. The revised version was sent to these same laboratories with a test sample in September. It is anticipated that this test will be sent back to the MRB in FY 1987.

f. Glass Durability. The literature review of aqueous dissolution studies of nuclear waste forms continued in May and June. A reference search was made and approximately 2000 citations were found relating to the solubility of nuclear waste glasses.

g. Thermal and Processing Parameters. Viscosity versus glass composition was plotted on ternary plots to pursue a correlation. However, there are TiO_2 and Li_2O composition differences in the glasses. An effort also will be made to correlate composition and processing parameters with electrical conductivity.

A thermal-properties data base is being developed. Thermal data on glass exists in the 500°C to 1000°C temperature range but most of the data is not on N-waste glasses.

It is anticipated that 20 data tables will be made from the collected data. The data include Chick's West Valley Glass study of 102 seven component glasses, 3 five component glasses and 4 three component glasses. Viscosity, composition, composition calculation, thermal properties, etc. will be included in the tables. Thermal properties being collected are thermal conductivity, specific heat, thermal expansion, viscosity and melting point.

h. Development of MCC-17 Recommended Practice for Assuring the Quality of Chemical Analyses. This recommended practice will be to assure the quality of chemical analyses associated with the development, testing and production of nuclear waste forms. This method will be developed in FY 1987.

WBS 2.6 Canister Qualification Test Methods (MCC).

a. Canister Qualification Test Methods (R. K. Farnsworth and P. A. Scott). The MCC-18 "Waste Canister Thermal Test Method" was completed and demonstrated in May. MCC-18 assumes that the canister is filled with glass, and the procedure's format is similar to MCC-15, "Waste/Canister Accident Testing and Analysis Method". The furnace and furnace sleeve were heated to 850°C prior to canister insertion. The test requirement is that the canister must be exposed to temperatures in excess of 800°C for 30 minutes. 24 thermocouples were used and two of these failed, indicating the need for multiple thermocouples. The canister's surface was covered with a layer of powder when withdrawn after the test, indicating oxidation. A draft of MCC-18 was circulated for comments and returned in July.

In October, K. M. Olson reported that canisters would be modified to make purging of CO₂ possible. This modification involves welding valved inlet and outlet tubes into holes drilled through the canister lid. During November, testing was modified to determine how glass leaching changes as a function of the silica concentration in the leachate and to determine the rate of silica precipitation from the leachate.

WBS 3.4 Hanford Grout Qualification Test Methods (MCC).

a. Hanford Grout Test Methods (R. J. Serne). Hanford Facilities Waste (HFW) grout leaching and leachate Hanford sediment adsorption data were assembled and analyzed in May.

Discussions were held with PNL regarding the Performance Assessment Scientific Support (PASS) program on mathematical models used for analyzing combined leach/adsorption data. Constant release rates or first-order decay leach rates and constant linear adsorption are required by most models including that of Pigford of the University of California. These leach rate restrictions may not be met with grout, and an experiment will be started to check the Pigford model.

The June report stated that the ANSI 16.1 and static tests with HFW grout show leach rates continuing to drop even after 400 days, and a simple diffusion mechanism does not describe the data. It appears that precipitation (calcium carbonate, magnesium hydroxide, etc.) and adsorption in the grout must be interfering with the diffusion. In the experiment with no sediment, only strontium (Sr) and cesium (Cs) leached from the block in measurable amounts. No radionuclides were found in the experiment with sediment, and adsorption of Sr and Cs must have occurred. The nitrite leached from the grout was absorbed by the soil. Further tests showed that the soil absorbed a given amount of nitrite and then let additional nitrite pass through.

Radiocounting of Cladding Removal Waste (CRW) leachates showed no detectible ²⁴¹Am, ²³⁸Pu, ¹⁴C or ¹²⁹I but ⁸⁵Sr, ¹³⁷Cs, ¹²⁵I and ⁹⁹Tc were present.

In September, the report "Progress Report on MCC-14.9 Grout Performance Test for Hanford Site Disposal" was prepared and submitted to the DHLWTP Office.

In November, reduced funding from the DHLW/TPO was reported. A work stoppage was put into effect until decisions were made about future levels and directions of the work.

MCC Support to the Transportation
Technology Center AR-05-15-30

1. Impact Test Pad Design and Fabrication (J. A. Glissmeter and P. A. Scott). The pad design was revised and approved and construction was begun in May. The impact pad was constructed in June with an estimated weight 128,000 kg. This pad should withstand an impact from a 12,800 kg test specimen, and the mass of the Savannah River canisters is about 1,936 kg. This pad should be sufficiently rigid for these tests. Testing procedures will include simulated flaws of two sizes; a 3/32 inch diameter hole for normal transportation conditions and a 3/8 inch diameter hole for accident conditions. An audit of the MCC work for the Safety Analysis Report for the shipping cask was conducted by Sandia National Laboratories in August.

2. Full Scale Canister Impact Testing (P. A. Scott). A cost estimate was prepared for the Defense Waste Processing Facility (DWPF) for impact testing of canisters dropped on end from a height of one foot. A procedure was devised to capture fines escaping through a reference sized hole when the canister is pressurized. The method requires that the canister be pressurized with a water soluble gas.

Two canisters were tested during September. MCC-15 procedures will be used to determine leak tightness and impact area strain. A simulated flaw (hole) will be made in the canister and the waste leakage from this flaw determined. Two canisters were impacted in October; one, designated A-27, from a height of 30 feet and the other, designated A-10, from a height of 1 foot. Sampling cups will be placed over the canister flaws and the canisters will be transported 1000 miles by truck to determine the amount of glass leakage under these conditions. Helium leak and dye penetrant tests of the bottom weld and impact area in November showed no canister breaching.

The report in December stated that work on this project, MCC support to the Transportation Technology Center, had stopped until funding issues were resolved.

MCC Support to the West Valley
Demonstration Project AH-10-30

1. Reference Glass Chemical Durability Testing (M. R. Smith and G. B. Mellinger). 91 days of MCC-1 testing with ARM-1 glass were reported completed in May. Leachates through 56 days were analyzed and compared with ARM-1 data in the MCC-D5 data package and showed good agreement. Preparations are complete for tests with the CUA pulsed

flow test except for the installation of the anoxic chamber needed to maintain a CO₂-free atmosphere. Tests for the SRP will be in 90°C brine with excess salt and samples of glass and low-carbon steel present. These tests will not begin until West Valley gets a confirmation from SRP regarding the type of salt and other conditions needed.

Testing of the CTS glass using MCC-1 was completed in August. Some Ca, Mg and possible Ba contamination had occurred with the blank tests. CTS glass tests for the SRP began in September. ATM-10 will be tested in the same manner.

The June report stated that a WPP researcher will help the SRP develop a radionuclide release test for inclusion in the Waste Acceptance Preliminary Specifications (WAPS). The researcher plans to propose an MCC-3 type test in which the final forward rate of glass dissolution would be the acceptance criterion.

2. Approved Testing Material ATM-10 (J. W. Wald and J. L. Daniel). Preparation of ATM-10 was continuing in May. The preliminary batch was prepared and analyzed by two PNL laboratories and showed satisfactory agreement except for aluminum concentrations. The full frit batch was prepared and is undergoing analysis at CUA, LLNL, West Valley and two PNL laboratories.

The July report indicated compositional changes in ATM-10 to increase the Al₂O₃ content from 2.83 to 6.5 percent and to increase the SiO₂ content to the initially specified 44.9 percent. The Tc content was low and the Pu content was high in the new batch of glass. Tc doping was applied to modify the final composition. Preparation of ATM-10 was completed in August. The mean concentration of the 25 nonradioactive elements was within the required limits. SO₃ content was low and Pu and Tc were high. It was decided in discussions with West Valley that the composition was still within reasonable limits. Plans for characterization of ATM-10 were submitted to West Valley for approval in September.

The October report announced the fabrication and characterization of ATM-MV/205 by J. L. Daniel and G. D. Maupin. The composition of this material is based on ATM-10 with nonradioactive constituents being substituted for the radioisotopes.

J. L. Daniel reported on analytical methods validation and correlation in December. An examination was conducted of some of the analytical methods in use by MCC for nuclear waste analysis and potential sources of inconsistencies which might be typical of the industry. This activity will continue in January 1987.

Appendix C. Database Report Format Example, Keyword
Checklist Tree and Keyword Checklist

Sample of Database Output in Report Format PACKIT

WORK.REF.. Print of Text Files in PACKIT Form

0005 Cit.no: 0005 | Report.no: UCRL-90875 | Authors: McCright, R.
D.; Weiss, H. | Title: Corrosion Behavior of Carbon Steels under
Tuff Repository-Environmental Conditions | Contractor: Lawrence
Livermore National Laboratory | Sponsor: U.S. Department of
Energy, Washington, DC | Date: 10-84 | Contract.no: W-7405-Eng-48
| Pages: 9 | Abstract: Carbon steels may be used for borehole
liners in a potential high-level nuclear waste repository in tuff
in Nevada. Borehole liners are needed to facilitate emplacement
of the waste packages and to facilitate retrieval of the
packages, if required. Corrosion rates of low carbon structural
steels AISI 1020 and ASTM A-36 were determined in J-13 well water
and in saturated steam at 100 C. Tests were conducted in
air-sparged J-13 water to attain more oxidizing conditions
representative of irradiated aqueous environments. A limited
number of irradiation corrosion and stress corrosion tests were
performed. Chromium-molybdenum alloy steels and cast irons were
also tested. These materials showed lower general corrosion but
were susceptible to stress corrosion cracking when welded. |
Availability: NTIS | Ntis.no: DE85002247/XAB

0139 Cit.no: 0139 | Report.no: HEDL-7546 | Authors: Smith, H. D. |
Title: C-Ring Stress Corrosion Cracking Scoping Experiment for
Zircaloy Spent Fuel Cladding | Contractor: Westinghouse Hanford
Company, Richland, WA, Hanford Engineering Development Laboratory
| Sponsor: Department of Energy, Washington, DC, Office of
Geologic Repositories | Date: 03-86 | Contract.no:
DE-AC06-76FF02170 | Pages: 1 VOL. | Abstract: This document
describes the purpose and execution of the stress corrosion
cracking scoping experiment using "C-ring" cladding specimens.
The design and operation of the "C-ring" stressing apparatus is
described and discussed. The experimental procedures and
post-experiment sample evaluation are described. | Filelocation:
HEDL-7546

2 Records Processed

Sample of Database Output in Report Form SUPERLIST

WORK.REF.. Print of Text Files in SUPERLIST Form

CIT.NO: 0209

REPORT.NO: UCRL-53735

AUTHORS: Bazan, F.; Rego, J. H.

TITLE: The Tuff Reaction Vessel Experiment

CONTRACTOR: University of California, Lawrence Livermore National
Laboratory

DATE: 06-86

PAGES: 35

TYPE.REVIEW: Glass

ABSTRACT: A laboratory leaching test has been performed as part of a project to evaluate the suitability of tuff rocks at Yucca Mountain, Nevada, as a site for a high-level nuclear waste repository. Glass samples of the kind that will be used to store nuclear waste were placed in water inside tuff vessels, and then the tuff vessels were placed in water inside Teflon containers. Glass-component leach rates and migration through the tuff were measured for samples of the ATM-8 actinide glass, which is a PNL 76-68 based glass doped with low levels

v 99 237 238 239
of Tc, Np, U, and Pu to simulate wastes. Disc samples of this glass were leached at 90°C for 30, 90, and 183 days inside tuff vessels using a natural groundwater (J-13 well-water) as the leachant. Some samples were held by 304L stainless steel supports to evaluate the effect of this metal on the release rate of glass constituents. At the end of each leaching interval, the J-13 water present inside and outside the rock vessel was analyzed for glass components in solution.

On the basis of these analyses, boron, molybdenum, and technetium appear to migrate through the rock at rates that depend on the porosity of each vessel and the time of reaction. The actinide elements (uranium, neptunium, and plutonium) were found only in the linear leachate. Sodium, silicon, and strontium are present in the rock as well as in the J-13 water, and the addition of these elements from the glass could not be determined. Normalized elemental mass loss values for boron, molybdenum, and technetium were calculated using the combined concentrations of the inner and outer leachates and assuming a negligible retention on the rock. The maximum normalized release was

v 2
2.3 g/m for technetium. Boron, molybdenum, technetium, and neptunium were released linearly with respect to each other, with boron and molybdenum released at about 85% of the technetium rate, and neptunium at 5-10% of the technetium rate. Plutonium was found at low levels in the inner leachate but was strongly sorbed on the steel and Teflon supports. Neptunium was sorbed to a lesser extent. Future analysis of the tuff vessels will determine whether the actinides were strongly sorbed on the surface of the tuff rock.

AVAILABILITY: NTIS

0209 RECORD END

1 Records Processed

Keyword Checklist Tree

KEYWORDS	(FIELD NAME)	
A. TECHNICAL DESCRIPTION OF REPORT		
1. <u>scope of work</u>	(SCOPE)	LIST1
2. <u>applicability of data to licensing</u>	(LIC.APP)	LIST2
3. <u>model/methodology</u>	(MODEL)	LIST3
4. <u>measurement</u>	(MEASURE)	LIST5
5. <u>computer programs</u>	(PROGRAM)	LIST4
B. ENVIRONMENTAL FACTORS		
1. <u>testing laboratory site</u>	(ENV.LAB)	LIST6
2. <u>geographic location of field site</u>	(ENV.LOC)	LIST7
3. <u>gaseous environment</u>	(ENV.GAS)	LIST8
4. <u>aqueous environment</u>	(ENV.H2O)	LIST9
5. <u>ionic or other chemical species present</u>	(ENV.ION)	LIST10
6. <u>solid materials environment</u>	(ENV.SOLID)	LIST11
7. <u>radiation environment</u>	(ENV.RAD)	LIST12
8. <u>experimental/test conditions</u>	(ENV.COND)	LIST13
C. MATERIALS TESTED		
a. waste container and component materials		
1. <u>general material type</u>	(MAT.GEN)	LIST14
2. <u>specific material designation</u>	(MAT.DES)	LIST15
3. <u>condition prior to test</u>	(MAT.COND)	LIST16
4. <u>test specimen specifications</u>	(MAT.SPECIMEN)	LIST17
b. radioactive waste (radionuclides and fuel) materials		
1. <u>waste form</u>	(MAT.WASTE)	LIST18
2. <u>radionuclides</u>	(MAT.RAD)	LIST19
c. environmental materials		
1. <u>kind of water present</u>	(MAT.H2O)	LIST20
2. <u>electrolytes present</u>	(MAT.ION)	LIST21
3. <u>solids</u>	(MAT.SOLID)	LIST22
D. PROPERTIES AND FAILURE MODES STUDIED		
1. <u>physical properties</u>	(PROP)	LIST23
2. <u>failure modes</u>	(FAIL)	LIST24

Keywords

(Field Name)

A. TECHNICAL DESCRIPTION OF REPORT

1. scope of work (SCOPE)
 __ data analysis
 __ design
 __ experimental data
 __ literature review
 __ planned work
 __ theory
 __ other _____
2. applicability of data to licensing (LIC.APP)
 __ key data
 __ supporting data
3. model/methodology (MODEL)
 __ Latin hypercube
 __ Monte Carlo
 __ PDF (probability distribution functions)
 __ sampling
 __ scoping test
 __ other _____
4. measurement methods (MEASURE)
 __ adsorption
 __ corrosion
 __ electrochemical
 __ linear-elastic fracture mechanics (LEFM)
 __ microscopy
 __ neutron diffraction
 __ x-ray diffraction
 __ sorption
 __ spectroscopy
 __ surface film
 __ tensile testing
 __ visual examination
 __ weight change
 __ other _____
5. computer programs (PROGRAM)
 __ WAPPA
 __ other _____

B. ENVIRONMENTAL FACTORS

1. testing laboratory site (ENV.LAB)
 __ field
 __ simulated field
 __ laboratory

Keywords

(Field Name)

2. geographic location of field site

(ENV.LOC)

☐ Deaf Smith County
☐ Hanford Reservation
☐ Yucca Mountain
☐ other _____

3. gaseous environment

(ENV.GAS)

☐ air
☐ carbon dioxide
☐ other _____

4. aqueous environment

(ENV.H2O)

☐ J-13 water
☐ brine
☐ brine (high ionic content)
☐ brine (low ionic content)
☐ deionized
☐ basalt composition
☐ granite composition
☐ tuff composition
☐ groundwater
☐ simulated groundwater
☐ deaerated distilled water
☐ distilled water
☐ aerated water
☐ other _____

5. ionic or other chemical species present

(ENV.ION)

☐ Cl
☐ Cu
☐ Fe
☐ Ni
☐ S
☐ Keller's reagent
☐ other _____

6. solid materials environment

(ENV.SOLID)

☐ basalt
☐ granite
☐ salt
☐ tuff
☐ bentonite
☐ other _____

7. radiation environment

(ENV.RAD)

☐ alpha radiation field
☐ gamma radiation field
☐ cobalt 60
☐ other _____

Keywords

(Field Name)

8. experimental/test conditions

(ENV.COND)

- ☐ ambient temperature
- ☐ high temperature
- ☐ ambient pressure
- ☐ high pressure
- ☐ hydrostatic head
- ☐ lithostatic pressure
- ☐ acidic solution (pH <7)
- ☐ basic (alkaline) solution (pH >7)
- ☐ neutral solution (pH = 7)
- ☐ redox condition
- ☐ static (no flow)
- ☐ dynamic (flow rate given)
- ☐ other _____

C. MATERIALS TESTED

a. waste container and component materials

1. general material type

(MAT.GEN)

- ☐ brass
- ☐ bronze
- ☐ cast iron
- ☐ cast iron (gray)
- ☐ cast iron (nodular)
- ☐ cladding
- ☐ copper base
- ☐ nickel base
- ☐ packing
- ☐ stainless steel
- ☐ steel
- ☐ carbon steel
- ☐ titanium base
- ☐ weld
- ☐ zircaloy
- ☐ zirconium base
- ☐ other _____

2. specific material designation

(MAT.DES)

- ☐ 304 stainless steel
- ☐ 304L stainless steel
- ☐ 308L weld filler wire
- ☐ 316L stainless steel
- ☐ 317L stainless steel
- ☐ 321 stainless steel
- ☐ 347 stainless steel
- ☐ 1020 carbon steel
- ☐ 1025 carbon steel
- ☐ high-nickel alloy 825
- ☐ zircaloy-4
- ☐ other _____

Keywords	(Field Name)
3. <u>condition prior to test</u>	(MAT.COND)
__annealed	
__annealed (austenitized and transformed)	
__case hardened	
__cast	
__cold worked	
__irradiated	
__magnetized	
__mill annealed	
__sensitized	
__sintered	
__solution treated	
__stress relieved	
__textured	
__welded	
__wrought	
__other _____	
4. <u>test specimen specifications</u>	(MAT.SPECIMEN)
[loading, geometry, and size designated by major category]	
__standard tensile (round type)	
__standard tensile (strip or strap type)	
__slow strain rate	
__standard compact	
__modified compact	
__size (circle): 1/4T, 3/8T, 1/2T, 3/4T, 1T, 2T, 3T, _____	
__bolt or wedge loading	
__tensile loading	
__precracked	
__prestressed (before exposure)	
__prestressed (during exposure)	
__other _____	
b. radioactive waste (radionuclides and fuel) materials	
1. <u>waste form</u>	(MAT.WASTE)
__commercial high level waste (CHLW)	
__defense high level waste (DHLW)	
__spent fuel	
__spent fuel (PWR reactor)	
__spent fuel (BWR reactor)	
__other _____	
2. <u>radionuclides</u>	(MAT.RAD)
__Co60	
__Np237	
__Pu239	
__other _____	

Keywords

(Field Name)

c. environmental materials

1. kind of water present

(MAT.H2O)

☐ J-13 water☐ J-13 steam☐ groundwater☐ other _____2. electrolytes present

(MAT.ION)

☐ acetic☐ chloride☐ chloride (high ionic content)☐ chloride (low ionic content)☐ other _____3. solids

(MAT.SOLID)

☐ granite☐ basalt☐ salt☐ tuff☐ bentonite☐ other _____

D. PROPERTIES AND FAILURE MODES STUDIED

1. physical properties

(PROP)

☐ bent beam tests☐ creep strength☐ density☐ elongation☐ heat capacity☐ modulus of elasticity☐ stress or strain☐ tensile strength☐ thermal conductivity☐ thermal expansion☐ yield strength☐ other _____

Keywords

(Field Name)

2. failure modes

(FAIL)

☐ buckling
☐ corrosion (crevice)
☐ corrosion (general)
☐ corrosion (intergranular)
☐ corrosion (local)
☐ corrosion (microbial)
☐ corrosion (pitting)
☐ corrosion (stray current)
☐ corrosion (galvanic)
☐ corrosion (stress cracking) SCC
☐ creep
☐ creep buckling
☐ dealloying
☐ debonding
☐ deformation (elastic)
☐ deformation (plastic)
☐ degradation (spent fuel)
☐ devitrification (glass)
☐ diagenetic-like changes
☐ fatigue (corrosion)
☐ fatigue (high cycle)
☐ fatigue (low cycle)
☐ fatigue (thermal)
☐ fracture (brittle)
☐ fretting
☐ hydration (glass)
☐ hydrogen attack
☐ hydrogen embrittlement
☐ leaching (radiation enhancement)
☐ leaching (spent fuel)
☐ matrix dissolution (glass)
☐ passivity
☐ poisoning (chemical)
☐ radiation effects
☐ rupture (ductile)
☐ rupture (stress)
☐ sensitization
☐ spalling
☐ thermal instability
☐ cracking (stress corrosion) SCC
☐ cracking (environmentally assisted)
☐ cracking
☐ other _____

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)

NUREG/CR-4735
Volume 2

2. TITLE AND SUBTITLE

Evaluation and Compilation of DOE Waste Package Test Data
Biannual Report: August 1986 - January 1987

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH

YEAR

May

1987

6. DATE REPORT ISSUED

MONTH

YEAR

October

1987

5. AUTHOR(S)

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M. Linzer, R. Ricker, J. Ruspi, and R. Shull

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Metallurgy Division - Corrosion Section
Center for Materials Science and Engineering
National Bureau of Standards
U.S. Department of Commerce
Gaithersburg, MD 20899

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

A-4171

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of High-Level Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

11a. TYPE OF REPORT

Technical

b. PERIOD COVERED (Inclusive dates)

August 1986 - January 1987

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This report summarizes results of the National Bureau of Standards (NBS) evaluations of Department of Energy (DOE) activities on waste packages designed for containment of radioactive high-level nuclear waste (HLW). The waste package is a proposed engineered barrier that is part of a permanent repository for HLW. Metal alloys are the principal barriers within the engineered system. Technical discussions are given for the corrosion of metals proposed for the canister, particularly carbon and stainless steels, and copper. In the section on tuff, the current level of understanding of several canister materials is questioned. Within the Basalt Waste Isolation Project (BWIP) section, discussions are given on problems concerning groundwater, materials for use in the metallic overpack, and diffusion through the packing. For the proposed salt site, questions are raised on the work on both the primary candidate material (ASTM A216 Steel) and the alternate alloy (Ti-Code 12). NBS work related to the vitrification of HLW borosilicate glass at the West Valley Demonstration Project (WVDP) and the Defense Waste Processing Facility (DWPF) is covered. NBS reviews of selected DOE technical reports and a summary of current waste-package activities of the Materials Characterization Center (MCC) is presented. Using a database management system, a computerized database for storage and retrieval of reviews and evaluations of HLW data has been developed and is described.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

Borosilicate glass; corrosion; evaluations; metal alloys; nuclear waste;
repository; reviews; vitrification.

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY
STATEMENT

Unlimited

16. SECURITY CLASSIFICATION

(This page)

Unclassified

(This report)

Unclassified

17. NUMBER OF PAGES

18. PRICE