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

Dear Mr. Wick:

Enclosed is a Draft Semiannual Report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-7). Comments by your staff and contractors are requested as soon as is practical, so that complete responses by NBS staff can be prepared and incorporated in a timely manner.

Please call me if you have any questions concerning this work.

Sincerely,

Charles G. Interrante

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

Enclosures

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To: Wick   To Wick
From: Interrante

[redacted]

NUREG/CR-4735
Volume 2

A-4171

EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA

DRAFT SEMIANNUAL REPORT

Covering the Period August 1986 to February 1987

C. Interrante, Editor

with E. Escalante
A. Fraker
S. Harrison
M. Linzer
R. Ricker
J. Ruspi
R. Shull

and Contributing Reviewers

U. Bertocci
D. Cronin
E. Escalante
A. Fraker
C. Interrante
M. Linzer
H. Ondik
E. Plante
R. Ricker
R. Shull

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

February 28, 1987

Prepared for:

U.S. Nuclear Regulatory Commission
Office of Nuclear Materials Safety
and Safeguards
Washington, DC 20555

EXECUTIVE SUMMARY

This document is the second semiannual progress report on the National Bureau of Standards (NBS) assessments of the Department of Energy activities related to the waste package for disposal of radioactive high-level waste. It contains NBS reviews of DOE reports and status reports on NBS assessments of DOE activities over the period September 1, 1986 to February 28, 1987. Highlights of information presented within these reports are given in this executive summary.

A major accomplishment of the NBS program for this period is the completion of the initial implementation of software required to operate a computerized database for storage and retrieval of reviews and evaluations of HLW data. On March 2, 1987, the database system was activated to serve users at the Nuclear Regulatory Commission (NRC) and the NBS. In the coming months the database system will be used to store all completed NBS reviews as well as selected reviews conducted earlier by Brookhaven National Laboratory. For the first time, this will permit rapid detailed searching of these reviews in order to locate essential technical information.

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 accellerated 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 the final barrier before radionuclide sorption can begin.
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 [1]. These areas are (1) the available data pertaining to the breakdown of packing material, (2) the groundwater chemistry, and (3) the corrosion behavior of the metallic overpack.

For the Salt Repository Project (SRP), 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. 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.

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 Savannah River Project. 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.

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1.0 INTRODUCTION

A principal objective of the NBS studies on 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 as well as that of the Materials Characterization Center (MCC). In this semiannual 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), Salt Repository Project (SRP), and Basalt Waste Isolation Project (BWIP). Recent activities of the MCC are summarized in section 6.

NBS reviews and evaluations of HLW data conducted over the period September 1, 1987 through February 28, 1987, are included as Appendix A of this report. 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 current working period is the initial implementation of a database for reviews and evaluations of HLW data. This system uses a database management system that has been 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 been completed to the level needed to begin the process of data entry. At the beginning of March, 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 related to the vitrification of high-level waste increased during this period. NBS participated in two meetings at which presentations were made giving details and a brief review of 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 aid in confirming the accuracy of DOE data and the reasonableness 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). The data to be obtained will relate the test methods and pitting potentials of low-carbon steels and 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 in accordance with the formulations and directions furnished by T.E. Jones of Rockwell Hanford Operations, Richland, Washington. These formulations are designed to represent the groundwaters of Grand Ronde -3 and -4. In addition, ASTM A27 low-carbon steel has been provided by the Bethlehem Steel Corporation for use in selected tests.

The other laboratory study, related to the durability of Zircaloy cladding on fuel rods, was prompted by recent reviews of corrosion of this material, which is present. The reviews indicate that insufficient data is available on oxidation and stress corrosion cracking (SCC) of Zircaloy, and this 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 simulates that used for cladding of fuel rods and J-13 groundwater will to 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 the areas of 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 submitted.

2.0 NNWSI- NEVADA NUCLEAR WASTE SITE ISOLATION PROJECT

The following is a summary of tuff data based on information obtained from NBS reviews of publications from DOE sponsored research. The reviews considered cover the period December 1985 to February 1987. Included with the following summaries are discussions of areas of research 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. Preceeding the discussions for each category are lists of its 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.

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) is low (200 - 400 mg/l) with one sample from the carbonate aquifer containing over 1000 mg/l TDS. The environment is oxidizing in the unsaturated zone, as expected, but the data indicate that below the proposed repository, in the unsaturated zone, the environment is reducing. 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 how representative the water samples are of 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.

The possibility of using carbon steel (A-36, AISI 1020, and gray cast iron) borehole liners for nuclear waste containers in tuff has prompted studies on the evaluation of the corrosion behavior of steel in this environment. Initial results indicate that uniform corrosion rates are low and of the order of 27 μm/yr after 5000 H in J-13 water, and 12 μm/yr in steam. However, localization of this attack can increase penetration rates by a factor of 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 a 9Cr-1Mo ferrous alloy and nodular cast iron, in a welded and stressed condition, 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.

"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.

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

"Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff", R. S. Glass, G. E. Overtuff, G. E. Garisson, and R. D. McCright, Lawrence Livermore National Laboratory, UCID-20174, June 1984.

"Laboratory Experiments Designed to Provide Limits on the Radionuclide Source Term for Radionuclide Migration from HLW or Spent Nuclear Fuel", V. M. Oversby, R. D. McCright, Lawrence Livermore National Laboratory, UCRL-91257, November 1984.

"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.

"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.

Austenitic stainless steel type 304L is the candidate canister material for the proposed tuff repository. The conclusions reached thus far indicate that this material will undergo uniform corrosion attack and, under certain conditions, may fail by stress-corrosion-cracking.

Examination of the data reveals that the corrosion rates were calculated with the assumption that corrosion was uniform. Visual examination of other specimens in similar environments, but longer term exposures, show crevice corrosion. This observation of localized corrosion raises some question of the validity of the conclusion that 304L stainless steel will undergo uniform corrosion in a simulated repository environment. This point is important because calculations based on uniform corrosion indicate a canister lifetime in excess of 1000 years, but nonuniform corrosion can lead to perforation in a fraction of this time. This issue of uniform versus localized corrosion must be investigated in order to develop a reliable data base for modeling purposes.

Stress corrosion studies 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. 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.

"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.

"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.

An important barrier to the release of radionuclides into the environment is the Zircaloy cladding that sheaths the spent fuel. Calculations based on known corrosion rates of Zircaloy in a water/steam environment show that penetration of the 800 μm Zircaloy wall will not occur in 10,000 years. However, other forms of failure, such as brittle fracture due to oxygen segregation and Zr_3O 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 release 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. Thus, it is clear that more research is needed in several areas regarding the effectiveness of Zircaloy as the final barrier before radionuclide sorption can begin.

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.

"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.

"Corrosion of Copper-Based Materials in Gamma Radiation", W. H. Yunker, Hanford Engineering Development Laboratory, HEDL-7612,

Experiments conducted in J-13 water and water vapor indicate that pure copper (CDA 101) has less tendancy to localized attack when compared with the other alloys tested, and that the 7% Al Bronze (CDA 613) and 30% Ni-Cu (CDA 715) show evidence of pit formation. 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 30% Ni-Cu alloy (CDA 715). Other work suggests that hydrogen peroxide formed during hydrolysis of water affects the oxide film that forms on copper alloys and the data indicate that there may be an effect similar to photopotential effects observed in other unrelated studies. More data are needed on this point since any changes in the structure of the oxide film may have an important influence on its protective properties. No cracking of the copper was observed.

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.

"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, Sept. 1985.

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

"Evaluation of the Potential for Spent Fuel Oxidation Under Tuff Ropository Conditions", R. E. Einziger, R. E. Woodley, Hanford Engineering Development Laboratory, HEDL-7452, March 1985.

"Test Plan for Series 2 Thermogravimetric Analyses of Spent Fuel Oxidation", R. E. Einziger, R. E. Woodley, Hanford Engineering Development Laboratory, HEDL-7577, February 1986.

"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.

"Test Plan For Series 3 NNWSI Spent Fuel Leaching/Dissolution Tests", C. N. Wilson, Hanford Engineering Development Laboratory, April 1986.

Stopping or retarding the release of radionuclides to the environment is the ultimate goal of the high-level waste packaging program. It is anticipated that 0.01% 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 there is a decrease in density of the fuel. This decrease in density results in swelling of the oxide which 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 rate of oxidation, 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.

Using data from the NRC regulations that govern the rate of release of radionuclides to the environment and using EPA regulations that govern the allowable total release evaluation of the radioactive waste inventory of a generic repository, it can be shown that americium and plutonium are two elements for which solubility and sorption data are needed. Seventeen other elements for which data are needed are the actinides, carbon and nickel.

3.0 BWIP -- BASALT WASTE ISOLATION PROJECT

In our first semiannual report in September 1986, the major deficiencies in the available data pertaining to a waste repository in basalt were noted. Briefly, these deficiencies fell in 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 effect corrosion, how are short-term corrosion data going to be extrapolated to long times?). In the last 5 months, some information on these topics has been received from DOE and been reviewed by NBS. The following discussion summarizes the main conclusions to be drawn from this data.

3.1 Packing Material Breakdown

Evidence is accumulating [2, 3] 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 Pu-239, Pu-240, Np-237, U-238, Se-76, and Ra-226 were found to be quite high when in air at 90°C. However, Tc-99 was found to be an exception. For this material the basalt/bentonite mixture was not found to be an effective absorber. 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 Se-75 onto the packing decreased significantly. It was suggested that this was caused by the loss of calcite during the thermal treatment. For the other radionuclides, no change in sorption or desorption characteristics was measured following the hyrothermal alteration.

More up-to-date thermal calculations of the basalt repository have been performed [4] using the most recent waste package and repository design. Temperature-time profiles were determined for various repository locations and for various places on the waste container and repository location. The maximum temperature of 260°C (at the center of the waste) is expected to occur 10 years following emplacement. At the waste container end, this maximum temperature is expected to be only 200°C. Following this time the temperature is expected to decrease to near 110°C in 10,000 years.

3.2 Groundwater Chemistry

No improvements have been made on determining the groundwater chemistry in the basalt repository site. However, Gray [5] found that in the presence of alpha and gamma radiation, the radiolysis of the basalt groundwater containing methane (as found in the Umtanum flow region of the Grand Ronde basalt basin) resulted in the formation of higher molecular weight organic substances. These could become complexing agents when in contact with the radionuclides and, thereby, change both the solution chemistry and the redox reactions possible in a future repository site in the Grande Ronde basalt.

3.3 Corrosion of the Metallic Container

On the corrosion behavior of the metallic waste-contanier, several pieces of information have become available. For 1025 cast steel, 1020 wrought steel, 1.2Cr - 0.5Mo cast steel, 2.5Cr - 1.0Mo cast steel, and 9Cr - 1Mo tempered steel some non-uniform corrosion behavior (after 20 months attack) has been observed in synthetic Grande Ronde basalt groundwater at 150 and 250°C [6]. Suprisingly, the 1020 and 1025 steels exhibited much higher general corrosion and pitting rates at 150°C than at 250°C. For all the above materials, it was observed that the general corrosion rate of these materials was enhanced by an order of magnitude when in the presence of gamma radiation. Slow-strain-rate tests indicated a loss in ductility in A27 and 9%Cr - 1%Mo cast and wrought steels respectively in basalt groundwater [7]. Such an effect certainly indicates the possibility of environmentally-assisted cracking occuring and further work should be performed on these materials. Fracture mechanics tests on another wrought steel, A387-9 [8, 9, 10, 11] have also indicated a loss of ductility after 2000 hours at 250°C in simulated Hanford groundwater. However, for ASTM A36 wrought steel [12] the fracture mechanics data did not indicate any crack extension in the precracked specimens after the 2000 hours immersion at 250°C.

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 (GE) 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 thoroughly characterize the salt environment and begin long-term corrosion testing on the entire range of expected environments as soon as possible.

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 [1]. In addition to those points, it appears that it's weld and heat-affected zones may be susceptible to hydrogen embrittlement [13]. 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, reported non-linear corrosion kinetics (decreasing corrosion rate with increasing time) indicate that surface films do form. DOE must avoid the temptation to apply non-linear corrosion kinetics and still assume that uniform corrosion is the only form of attack. Localized corrosion, stress corrosion cracking and hydrogen embrittlement are potential failure modes that must be eliminated by mechanistic modeling and/or experimental observations and not by assumption.

The alternate alloy for container fabrication is Ti-code 12. This alloy was selected because of its corrosion resistance. The corrosion resistance of this alloy 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 with greater thermodynamic immunity to corrosion, materials 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 aid the identification of addition alternate candidates. Also, composite designs should be considered, as for examples, those where the active and sacrifical 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

The lead worker for the salt project has been changed and, as a result, all of the reports previously evaluated for review have been collected and are being reevaluated. The documents and their status in this process 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 Ettefagh
"The Effect of Stabilizers on the Heat Transfer Characteristics of a Nuclear Waste Canister"

BMI/ONWI 612 (7/86)
"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"

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

Documents under Evaluation for Review:

PNL-5157 (8/84), J. E. Mendel et al.
"Final Report on the Defense High-Level Waste Leaching Mechanisms Program"

PNL-SA-1 PREPR (4/85), J. E. Mendel et al.
"Status of the Materials Characterization Center and the Materials Review Board"

NUREG/CR-2482 Vol 8 (1/85), M. S. Davis et al.
"Review of DOE Waste Package Program Draft Biannual Report"

NUREG/CR-3427 Vol 1-3 (8/83), D. Stahl and N. E. Miller
"Long Term Performance of Materials used for High-Level Waste Packaging"

NUREG/CR-3405 (8/83), D. Stahl and N. E. Miller
"Long Term Performance of Materials used for High-Level Waste Packaging"

NUREG/CR-4134/R1 (8/86), H. C. Claiborne et al.
"Repository Environmental Parameters and Models/Methodologies Relevant to Assessing the Performance of High-Level Waste Packages in Basalt, Tuff, and Salt"

ONWI-462 (6/86), ONWI
"Conceptual Waste Package Interim Performance Specifications for Waste Forms for Geologic Isolation in Salt Repositories"

ONWI-483 (7/83), ONWI
"Results of Repository Condition Study for Commercial and Defense High Level Nuclear Waste and Spent Fuel Repositories in Salt"

BNL 29909, P. W. Levy et al.
"Radiation Damage Studies on Synthetic NACL Crystals and Natural Rock Salt for Radioactive Waste Disposal Applications"

BMI/ONWI-545 (8/84), ONWI
"Performance Assessment Plans and Methods for the Salt Repository Project"

NUREG/CR-2333 BNL-NUREG-51458 Vol. 1 (2/82), R. Dayal et al.
"Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC"

NUREG/CR-2333 BNL-NUREG-51458 vol. 2 (2/82), G. Bida
"Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC"

NUREG/CR-3427 BMI-2113 vol. 4D. (6/84), Stahl and N. E. Miller
"Long-Term Performance of Materials used for High Level Waste Packaging"

NUREG/CR-3091 BNL-NUREG-51630 Vol. 7 (1/86), P. Soo eds.
"Review of Waste Package Verification Tests"

NUREG/CR-2482 BNL-NUREG-51494 Vol. 9 (12/85), T. Sullivan et al.
"Review of DOE Waste Package Program"

BMI/ONWI-599 (4/86), R. E. Thomas
"A Feasibility Study Using Hypothesis Testing to Demonstrate Containment of Radionuclides within Waste Packages"

DOE/CH/10140-03(85-2) (3/85), ONWI
"Salt Repository Project Technical Progress Report for the Quarter 1 January-31 March, 1985"

DOE/CH/10140-03(85-3) (6/85), ONWI
"Salt Repository Project Technical Progress Report for the Quarter 1 April-31 June, 1985"

NUREG/CR-2482 Vol. 8 (12/85), C. Brewster et al.
"Review of DOE Waste Package Program, Semiannual Report 10/84-3/85"

DOE/CH/10140-3(85-1) (12/84), ONWI
"Salt Repository Project Technical Progress Report for the Quarter 1 October-31 December, 1984"

NUREG/CR-2333 BNL-NUREG-51458 Vol. 8 (2/82), G. Bida et al.
"Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC"

ONWI-438 (4/83), Westinghouse
"Engineering Waste Package Conceptual Design Defense High-Level Waste (Form 1), Commercial High-Level Waste (Form 1), and Spent Fuel (Form 2) Disposal in Salt"

PNL-4250-4, J. L. McElroy and J. A. Powell
"Nuclear Waste Management Semiannual Progress Report April 1983 through September 1983"

MRB-0488 (12/85), Ad Hoc Corrosion Panel
Reviews of the Waste Disposal Programs by the Ad Hoc Corrosion Panel

BMI/ONWI-000 (9/84), D. P. Moak
"Waste Package Materials Resting for a Salt Repository: 1983 Status Summary Report"

ONWI-463 (6/83), ONWI
"Interim Performance Specifications for Conceptual Waste Package Designs for Geologic Isolation in Salt Repositories"

ONWI-464 (6/83), ONWI
"Conceptual Waste Package Interim Product Specifications and Data Requirements for Disposal of Borosilicate Glass Defense High-Level Waste Forms in Salt Geologic Repositories"

ONWI-462 (6/83), ONWI
"Conceptual Waste Package Interim Performance Specifications for Waste Forms for Geologic Isolation in Salt Repositories"

BNL 51816 (9/83), J. Tilly et al.
"Verification of Multiple Steady-States in Localized Electrochemical Corrosion"

D. H. Alexander et al.
"Conceptual Model for Deriving the Repository Source Term"

ONWI-490, ONWI
"Waste Package Materials for Testing for a Salt Repository"

ONWI-449, ONWI
"Some Characteristics of Potential Backfill Materials"

ONWI-452, ONWI
"WAPPA: A Waste Package Performance Assessment Code"

ONWI-501, ONWI
"Methodology for Predicting the Life of Waste-Package Materials and Components Using Multifactor Accelerated Life Tests"

PNL-SA-11713 (11/83), Westerman
"Evaluation of Iron-Based Materials for Waste Package Containers in a Salt Repository"

DOE/CH/10140-03(85-4)
"SRP Technical Progress Report July 1-Sept 30, 1985"

DOE/CH/10140-03(85-4)
"SRP Technical Progress Report October 1-Dec 31, 1985"

Technical Papers for Citation in Data Base:

T. M. Ahn, H. Jain, P. Soo, "Identification of Crevice Corrosion in ASTM Grade-12 Titanium in Simulated Rock Salt Brine at 150°C", Brookhaven

T. M. Ahn, H. Jain, P. Soo, "The Effects of Gamma Radiolysis of the pH of WIPP Brine A", Brookhaven

Y. J. Kim, R. A. Oriani, "Electrochemical Behavior of Ti-30Mo in a Gamma Radiolysis Environment", Corro., 43, (1), P56, 1987.

D. R. Macfarlane, "The Dissolution Mechanism of Iron in Chloride Solutions", J. Electrochem. Soc., 133, (11), P2241, 1987.

V. Jovancicevic, J. O'M. Bockris, J. L. Carbajal, P. Zellenay and T. Mizuno, "Adsorption and Absorption of Chloride Ions on Passive Iron Systems", J. Electrochem. Soc., 133, (11), P2219, 1987.

G. P. Marsh, K. J. Taylor, G. Bryan and S. E. Worthington, "The Influence of Radiation on the Corrosion of Stainless Steel", Corro. Sci., 26, (11), P971, 1986.

5.0 VITRIFICATION OF HIGH-LEVEL WASTE

The NBS work related to the vitrification of 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 fo 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 HWL.

A presentation of the DOE waste glass program, involving both the WVDF and DWPF of the SRP, took place at the Forrestal Building, Washington, D.C., on December 8 and 9, 1986. Individuals from NRC, NBS, WVDP, SRP and others 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 adequate determinations of the efficency 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 they did furnish sufficient details to accelerate the NBS studies aimed at understanding the DOE efforts in the area of the vitrification of HLW.

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 appropriate interactions between the DOE and NRC and their contractors, the following are observed as 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? These and other points have not been adequately addressed by the West Valley Project. Moreover, DOE has not yet announced how much credit (for rate of release of radionuclides), if any, they 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 reviewed.

An intensive effort was made to engage outside consultants who were expert in the field of glass leaching and, at the same time, meet the strict NRC conflict-of-interest standards. Three individuals who satisfied these criteria were identified and placed on contract to NBS. These are Mr. J. Bruce Adams, Dr. John Wasylyk and Dr. David Cronin.

For their initial task, they were asked to review the Pacific Northwest Laboratory (PNL) document, "Final Report of the Defense High-Level Waste Leaching Mechanisms Program," published in August, 1984 [14]. (A draft review of chapter 1 is included in Appendix A). Reviews of chapters 3, "Environmental Interactions," and 4, "Dissolution of Specific Radionuclides," will be available shortly. Related documents are being identified and acquired for review.

6.0 MCC -- MATERIALS CHARACTERIZATION CENTER

6.1 Introduction

This report on the Materials Characterization Center (MCC) covers activities of the MCC May 1, 1986, through December 31, 1986. A description of the MCC and its objectives was given in NUREG/CR-4735, Vol. 1, November 30, 1986. 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 test 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 to at least thirty recipients, including the National Bureau of Standards. The MCC activities cover a wide range of activities which are grouped by the MCC under eight sections. The first part of this report summarizes MCC activities.

6.2 MCC Sections According to Programs Served

The MCC reports are presented in the following manner. Each section is presented under a separate topic for the convenience of those interested in selected portions only. The eight sections are listed as follow:

A. MCC Program Administration
B. MCC Quality Assurance
C. MCC Support to the Office of Geologic Repositories (RW-20)
D. MCC Support to the Salt Repository
E. MCC Support to the Basalt Waste Isolation Project
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

6.2.1 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 they are used in this report for the purpose of clarity and for providing the most current contacts in areas.

6.2.1.1 Program Planning

Program planning deals with a number of activities. Among these is 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 met at the Pacific Northwest Laboratory (PNL) to discuss MCC work on the West Valley Program. A walk-through safety inspection was conducted 324 and 325 buildings.

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.

6.2.1.2 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".

6.2.1.3 American Society for Testing and Materials (ASTM) Interactions

MCC interactions with ASTM have been involved 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 the development of the document, that the document should be aimed at developing a concensus 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 asociated 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 to the accelarated testing task group, 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. The December meeting was held. A revised draft was distributed to the task group. Written comments were received from BWIP. K. Chang (NRC) provided comments on the draft.

6.2.2 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. 2.

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 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. Proposed corrective action required 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 and a lack of objective evidence that a second deficiency report was issued. 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, in July. The upgrading of PNL-MA-70 and PNL-MA-60 was completed in September.

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

The subjects under this topic do not fall into specific unrelated categories.

6.2.3.1 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 (OCRD) 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; 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 and it was 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. Routing 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.

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

6.2.3.2 Reports

A Shielded Analytical Instruments Workshop was held at PNL by the MCC and 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" was prepared by J. L. Daniel and H. E. Kjarmo 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.

6.2.3.3 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 which will pass the materials on to the participants. Granite was chosen to be the host rock for the U.S. tests. Initial tests will be for 28 days and the committee will reconvene in March or April 1987 to compare results.

6.2.3.4 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.

Solid state analysis of 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 sizes were 6 nm diameter at the edge, 15 to 20 nm diameter in the center and 100 nm diameter in the grain boundaries. The metals, molybdenum (Mo), ruthenium (Ru), technicium (Tc), palladium (Pd) and ruthenium (Rh) were identified. Xe (10 wt. %) 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 Zr-93, Cs-135, and Am-235 in the fuel, for I-129 on the cladding and Ni-59, Ni-63, Nb-94, and C-14 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.

6.2.3.5 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.

6.2.3.6 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.

6.2.3.7 "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.

6.2.3.8 Planned Work

Planned work and work accomplished are described. This work included installation of spent fuel characterization equipment in B-cell and D-cell, 324 Building, and the initiation of 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.

6.2.4 Support to the Salt Repository Project

6.2.4.1 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 and plans were 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. The other thirteen tests were handled as follow. 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 reviewing the 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 instead of titanium. 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 empahsis 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 will attempt 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.

6.2.5 MCC Support to the Basalt Waste Isolation Project

6.2.5.1 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 (Np-237) glass, basalt, and Gr-4 groundwater. Experiment number 2 ran for 613 days and contained ATM-4 (Pu-239) 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 appropriate 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 tor 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 current scheduled testing is completed in 1987.

In October, developmental testing began on the effect of temperature on the rate oa 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% 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 Np-237 and 613 day Pu-239 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.

6.2.5.2 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. This report 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. Anantatmula (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 orderly discontinue the tests and complete the records. Specimens will be archieved 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 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. Merza 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 drip-leg covers were received in June. Preliminary testing was begun in August and vessels with the dip-leg and rupture disk maintained their seal 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 and penetration rates than oven 1, chamber 2 had higher penetration and penetration rates 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 and penetration rate than did the coupons at the front of the oven, but this trend in penetration and penetration rate 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 and penetration rates than the large coupons.

6.2.5.3 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% completed, 120 day tests of 64 specimens with packing material are completed, 300 day tests with 64 specimens with packing are 40% 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. The anoxic chamber is not able to attain the required anoxic condition of <50ppm, and a new chamber was requested in the FY 1987 budget.

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

6.2.6.1 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 CO_2 free atmosphere within the teflon 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 whould 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, in October drafted a revised statement of precision and bias 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 CO_2 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 and used 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 the testing of glass, allow reuse of teflon containers, specify stainless steel or fused quartz containers for gamma emmitting 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.

Data is being collected and it is anticipated that 20 data tables will be made. The data includes 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 practive 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.

6.2.6.2 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 to remove CO_2 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.

6.2.6.3 WBS 3.4 Hanford Grout Qualification Test Methods (MCC).

a. Handford 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 regarding 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 interferring with the diffusion. In the experiment with no sediment, only scrotium (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 the soil to absorb a given amount of nitrite and then to let the nitrite pass through.

Radiocounting of Cladding Removal Waste (CRW) leachates showed no detectible Am-241, Pu-238, C-14 or I-129 but Sr-85, Cs-137, I125 and Tc-99 were present.

In September, the progress report entitled "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 and decisions were made about future levels and directions of the work.

6.2.7 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 a weight estimated to be 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 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 referenced 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 hypothetical 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.

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

<u>1. Reference Glass Chemical Durability Testing (M. R. Smith and G. B. Mellinger</u>. 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 CO_2 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) and the researcher plans to propose an MCC-3 type test in which the final forward rate of glass dissolution would be the acceptance criteria.

<u>2. Approved Testing Material ATM-10 (J. W. Wald and J. L. Daniel)</u>. 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 increas the Al_2O_3 content from 2.83% to 6.5% and to increase the SiO_2 content to the initially specified 44.9%. The Tc content was low and the Pu content was high in the new batch of glass. Tc dopping 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. SO_3 content was low and Pu ant Tc were high. It was decided in discussions with West Valley that these composition vatioations still were 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.

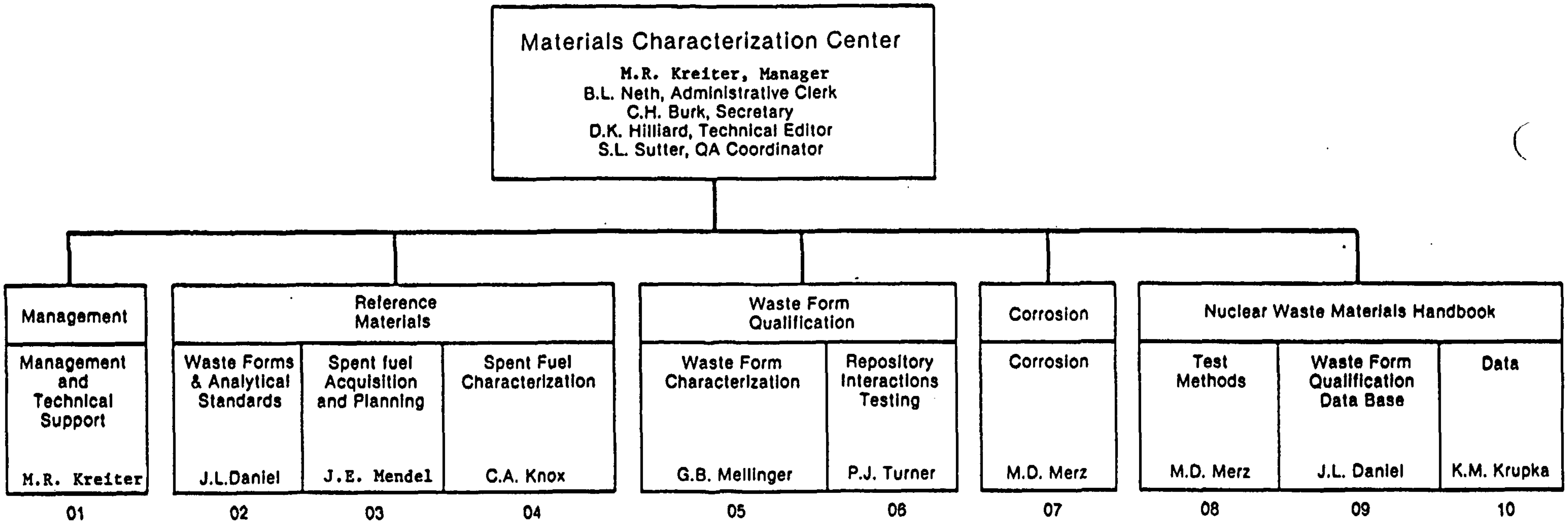

Figure 1. Organization Chart of Materials Characterization Center.

Figure 2

Waste Compliance Portion of DOE Waste Acceptance Process

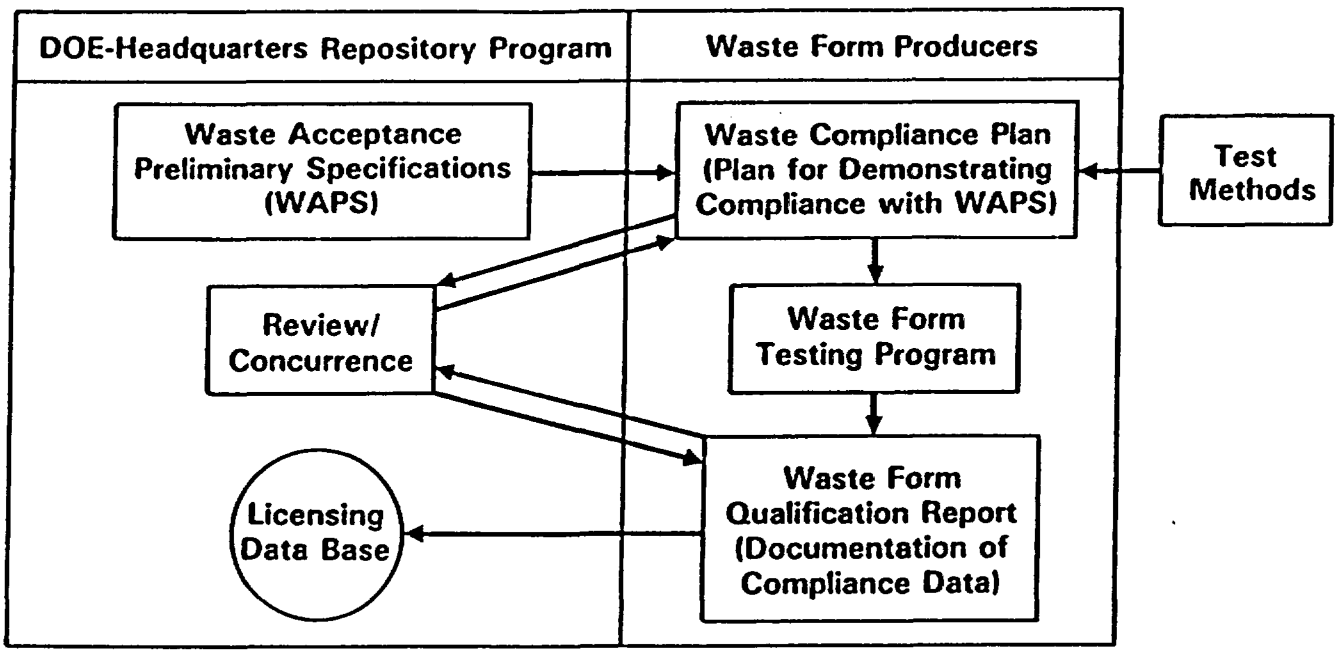

Figure 3

Advantages of the Use of Standardized Tests in Waste Compliance Testing

- Elimination of parallel test development by different organizations (develoment 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 1

Waste Acceptance Preliminary Specifications

WAPS Specification	Testing Requirements	Applicable Test Methods: Standardized Tests	Undergoing Standardization	Under Development
Waste Form Specifications				
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
Canister Specifications				
Fabrication and Closure Specification	Canister Closure Leak Tightness	--------- No Standard Test ---------		
Canistered Waste Form Specifications				
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 completing a thorough analysis of the requirements for a computer assisted database for reviews and evaluations on high level waste data, the commercial software "Revelation" was chosen as most compatible data base management system [15]. The Waste Package Database has been developed using Revelation and 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. The tracking system 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 information in 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. Key words were 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 an IBM XT and an IBM AT microcomputer, and data entry has begun.

At either a terminal screen or a printer, information records may be displayed in many different formats. The names of report formats developed for the database are PACKIT, PACK, Formatted Text Fields (FTF), and Superscripts and Subscripts in Fields (SUPER) and Superscripts and Subscripts in List (SUPERLIST). A sixth report format, LASER, also is planned. Included as Appendix B are copies of the first five report formats. 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. Report formats FTF, SUPER and SUPERLIST illustrate the output from this program. 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 SUPER, however, each new record begins on a separate page. The features from 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. LASER will give report quality output on a Hewlett Packard Laser Jet Printer.

Additional fields within the Reference File have been developed. These fields 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 pertinent 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 Hewlett Packard Laser Jet printer has been received and has been interfaced with both Revelation and Work Perfect. It and an Epson 286 printer are operational in the data center.

The overall structure of the Keyword Hierarchy has been completed. Categories for various keywords were established and an outline tree was prepared listing the various groupings. (An example is included in Appendix B). The bold face terms designate the broad topical categories to which keywords belong. First drafts of the Keyword Checklist were carefully checked to ensure that none of the keywords would be omitted on the revised list. The revised Keyword Checklist allows for the addition of new keywords as well as the addition of new categories as needed. Use of the improved Keyword hierarchy will facilitate searches for the most pertinent information on a specific topic and will result in a set of highly relevant citations.

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. The database differs from the NBS version of the Waste Package Database in that it is a subset of the total NBS database. This subset 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, new tapes will be delivered to the NRC. Documentation within the database, will include a tutorial program which contains help screens and menus. These help screens and menus train and assist the user who wishes to query the database. (Examples of help screens are included in Appendix B.) 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 will provide clear directions for searching and retrieving highly relevant citations.

A great deal of information is created and stored in computer-readable formats. It is necessary at times, however, to transfer information from one data entry format to another (i.e. 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 reduced when this software is operational.

It is anticipated that selected Brookhaven Reviews will be included in the Database for Reviews and Evaluations on HLW Data. NBS staff will identify the most highly relevant reviews [16, 17]. Once identified magnetic media encoding of the reviews will begin. Magnetic media encoding will 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 and perform other functions on the database. Because menus are very user friendly, the menus are developed, the system will be easier to learn and use, as menus are very user friendly.

In conducting a query of the database using TCL in the present configuration, key fields and keyword lists are used to provide very 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 be developed to make it easier. Other examples of needed menus are those that will facilitate data entries made by either a data clerk or a report reviewer.

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 closely track the reviews in progress, for example, by furnishing lists of activities that are taking more time than that anticipated by management.

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[17] T. Sullivan, H. Jain, T. Abraham, et al., "Review of DOE Waste Package Program," Brookhaven National Laboratory, NUREG/CR-2482 Vol. 9, Dec. 1985.

Appendix A. NBS Reviews of DOE Reports Concerning the Durability of Proposed Packages for High-Level Radioactive Wastes

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 Respository Conditions", HEDL-SA-3271FP, March 1985.

DATE REVIEWED: 7/29/86

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 hrs.

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, % weight change and change in ^{16}O/Metal or ^{18}O/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.,

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 Zircaloys; 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).

UNCERTAINITIES 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.

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. I have several criticisms of this work. Athough 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.

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

The report contains a brief literature survey conducted for the Nuclear Regulatory Commission on two DoE-sponsored topics of research and testing (1) stress-corrosion behavior for candidate container materials and (2) the results of stress-corrosion cracking tests.

(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. = balan

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.

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

Alloy	Ultimate Tensile Strength (MPa)	Ultimate Tensile Strength (ksi)	0.2% Offset Yield Strength (MPa)	0.2% Offset Yield Strength (ksi)	Elongation (%)	Rockwe Hardne (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% 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% 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% 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 Instron 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% 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% 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% Elastic Limit (kg), Deflection at 100% Elastic Limit (cm), Deflection at 90% 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 $2x10^{-3}$) 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% 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 glyceregia. (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."

UNCERTAINTIES IN DATA

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

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.

Alloys with relatively low carbon contents were used and, as a result, actual container alloys with higher carbon contents would be more susceptible to sensitization and SCC.

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 of the low carbon content of the alloys, 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.

RELATED HLW REPORTS

APPLICABILITY OF DATA TO LICENSING

Ranking: key data ( ), 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 Commentsmments

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. H_2 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

[The following reports may be in the data center library already. If not, they should probably be obtained and a decision made as to whether any of them should be included in the data base. The Los Alamos reports are listed first, then U.S. Geological Survey 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
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

also, from the Office of Nuclear Waste Isolation,

16	ONWI-448

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

None

(c) General Comments

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Hanford Engineering Developement 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 (O_2) (atm), (-10 to -50) vs T°C (0 -300); Time to spallation (1-10000 hrs.) vs 1/T(K) (360-190°C), measure of oxidation rate of UO_2; 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 %/hr (10^{-5},-1) vs 1/T(K) (395-180°C) for spent fuel and UO_2 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 yr vs atm, cannister 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 yrs, showing possible pathways dependent on initial conditions; Cumulative damage fraction vs year breach occurs (used to estimate if breached cladding will split in 300 yrs); Spent fuel disposal scenarios beyond 300 years dependent on radiation field, temperature, atmosphere, moisture, and cannister 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%. There may be real differences in fuels dependent 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% 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.

<u>APPLICABILITY OF DATA TO LICENSING</u>:
[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.

(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 $3x10^{-4}$ 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 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: August 23, 1986

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 days 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°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. I have several criticisms of this plan. 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.

<u>APPLICABILITY OF DATA TO LICENSING</u>
[Ranking: key data ( ), supporting data (X)

(a) Relationship to Waste Package Performance Issues Already Identified

Related to issue 3.3 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

WASTE PACKAGE DATA REVIEW FORM

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

DATE REVIEWED: 10/7/86

TYPE OF DATA

Experimental, corrosion, SCC, crevice corrosion

MATERIALS/COMPONENTS

Pure Cu (CDA 101), 7% Al bronze (CDA 613), 30% Ni-Cu (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 10^5R/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-penetiant 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 x 10^{-5} R/h) vs. position from top of irradiation tube specimens (0 to 0.6 m)
2) Vessel gas pressure (0-300KPa) vs. time up to 115 days

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

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 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 7%-Al Bronze (CDA 613) shows lower uniform attack. The presence of liquid on the surface enhances corrosion. (CDA 715) a significant enhancement of corrosion with γ-ray exposure was found only for the 30% Ni-Cu alloy. 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. An informal report intended for internal and limited external distribution.

DATE REVIEWED: January 5, 1987

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 1 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 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 Zircaloys; 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).

UNCERTAINITIES 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

RELATED HLW REPORTS

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: August 11, 1986

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 yr, 1/100,000 of 1000 yr inventory, and level to which release rate must be controlled in Ci/MTHM per yr, 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 yr, 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 x 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 yr and EPA Table 1 limit in Ci/MTHM, 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 1% spiked release for CS, C, I, and Tc in the 1000 yr 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 10.

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 accesible 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 cite performance assessment but by far the most important elements are americium and plutonium. Note that the EPA regulation puts a limitation on release to the accesible environment in terms of Ci/MTIHM 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 which presents problems during initial handling of spent fuel has essentially decayed away by the time these regulations come into effect.
Note: In this report the term Ci/MTIHM is used interchangeably with the term Ci/MTHM.

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 and 3.3.1.2 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", March (1984).

DATE REVIEWED: October 28, 1986

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 N_2 and CH_4.

TEST CONDITIONS

The simulated basalt ground water was placed in stainless steel vessels with quartz liners and pressurized to about 0.14 MPa with gaseous N_2 and 3.30 MPa with gaseous CH_4. Gamma irradiation was done in a ^{6C}o facility at dose rates of $2x10^4$ to $5x10^6$ rad/hr. Alpha irradiation was done by spiking the solutions with ^{238}Pu at a concentration of $4x10^{-4}$ M resulting in a dose rate of $2x10^4$ rad/hr. 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 Chromatoghraphy 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 Rate of 46 Mrad (Dose rate was 1.5 Mrad/hr); Gas Concentration and Total Pressure at Total Dose of 77 Mrad (Dose rate was 1.1 Mrad/hr).

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 Sec) 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 CH_4 (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 absorbtion 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: October 31, 1986

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

TEST CONDITIONS

1. Corrosion rate by weight loss measurements

   - 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/hr. (not recycled)
- Solution passed through crushed basalt

2. Influence of Irradiation on Weight Loss

- Same as above with Co^{60} γ irradiation source (3×10^5 rad/hr)
- Test Temperature= 250°C
- Up to 16 months total exposure
- Two of the alloys were not tested in this environment:
  (a) 1.25 Cr, 0.5 Mo Cast Steel
  (b) 9% Cr, 1% Mo 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 5μm/yr.)-(0-20 months)
2) Corrosion Rate at 150°C (0-30 5μm/yr.)-(0-20 months)
3) Corrosion Rate During Irradiation at 250°C (0-30 5μm/yr.)-(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, Co^{60}, 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 corrosion rates found at 150°C are greater than those at 250°C, however, the authors make no attempt to explain this observation within this report.

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, 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 Issures

(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, June 1983.

DATE REVIEWED: October 6, 1986; November 13, 1986

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 in waste package containers, includes stress-corrosion cracking and fracture mechanics.

MATERIALS/COMPONENTS

Low-carbon and 9 Cr-1 Mo steels and carbon-steel weldments, potentially suitable for waste container materials, specifically:

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

TEST CONDITIONS

(1) State of the Material being Tested

AISI 1020 is to be tested in both hot-rolled and cast forms.
AISI 1025 is to be tested in both hot-rolled and cast forms.
ASTM A387 Grade 9 Class 2.
ASTM A36 is a hot-rolled steel.

(2) Specimen Preparation

Specimens are to be precracked. Initial K-levels are to be 30, 40, and 60 MPa x $m^{1/2}$. Specimens will be standard ASTM E647 "Compact Type" in the "1T" size (1-inch thick x 2.5-inch wide x 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 applied stress levels and flaw sizes which can be formulated through the stress-intensity factor (K). These relationships combined with the experimentally-measured fracture mechanics parameters (e.g., crack-growth rates, crack-growth thresholds, fracture toughness), allow quantitative estimates to be made of 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_{ISCC}, 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_{ISCC}, 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, for preparation of specimen load, crack length measurement, and LEFM data analysis.

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

Attached.

GENERAL COMMENTS

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 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

Draft Copy

WASTE PACKAGE DATA REVIEW

DATA SOURCE

(a) Organization Producing Data

Pacific Northwest Laboratory, Battelle Memorial Institute

(b) Authors, Reference

A. Barkatt, P. Macedo, C. Montrose; Chapter 1 : Mechanisms of Defense Waste Glass Corrosion: Dissolution of Glass Matrix from Final Report of the Defense High Level Waste Leachng Mechanisms Program, PNL-5157 August 1984

DATE REVIEWED: 2-22-87

TYPE OF DATA

Experimental data and literature review dealing with glass corrosion, including review of various types of corrosion tests.

MATERIALS/COMPONENTS

The materials studied were two simulated nuclear waste glasses, SRL TDS-131 and the Material Characterization Center defense waste reference glass(DWRG). Limited data is presented for other glasses to asssit in interpretation of results and assess general trends.

TEST CONDITIONS

The glass corrosion was studied under a variety of conditions to help elucidate the basic mechanisms involved.
Tests were conducted on both monolithic and powdered specimens of the two glasses.
Corrosion tests included static tests, continous flow and pulsed flow tests, constant medium tests and hydrothermal tests.
Variables included aqueous environment(deionized water, simulated ground water, buffered DI water), temperature, surface to volume ratio and time.

METHODS OF DATA COLLECTION/ANALYSIS

The corrosion data is presented as total mass loss per unit area per unit time or in terms elemental mass loss. Elemental mass loss is determined from measurement of elemental concentratons in the leachate. Leachate concentrations were determned by ICP spectometry, atomic absorption spectrometry and colorimetry. pH measurements are also given. Surface layer thickness and composition was determined by SEM/EDX. The corrosion data was reported as a function of time or flow rate.

AMOUNT OF DATA

Twenty one tables are given showing glass composition, corrosion data for individual glasses under specific conditions, comparison of different glasses under the same conditions and comparison of different test methods.
Twenty eight figures are given showing configurations of different test methods, corrosion data for specific glasses under specific conditions, comparison of results when different parameters are varied.
A list of all tables and figures is attached.

UNCERTAINTIES IN DATA

Error bars are included on some figures but not all. Tables of corrosion data make no reference to the precsion or accuracy of the data. The authors state that the pulsed flow experiments were performed at least in duplicate, blank tests were carried out and the analytical accuracy in the determination of the major components in solution was within +/- 10%.

DEFICIENCIES/LIMITATIONS IN DATABASE

None specifically mentioned.

GENERAL COMMENTS

The assignment for this study was to describe the leaching of the glass matrix as a function of glass compostion, leachant flow rate, temperature and leachate composition. This assignment was carried out successfully. This work provides an excellent review of glass corrosion as it pertains to nuclear waste glasses. There now appears to be a good understanding of the mechanisms involved in glass corrosion and the regimes where specific mechanisms are important. This was not the case a few years ago.
The authors have pointed out that glass corrosion rates are extremely dependent on the glass composition and the specific conditions under which corrosion takes place. They show that seemingly small changes in composition can have a very large

effect on corrosion rates. This is important because of the large compositional variatons that exist in the current holdings of defense high level waste. The understanding of the mechanisms as detailed by the authors allows meaningful site specific tests to be designed so that accurate predictions of glass corrosion in a real repository can be made. However, because of the sensitivity to compositional variations inclusion of a broader range of glass compositions would have made this study even more useful.

RELATED HLW REPORTS

Reports that should be added to the database include:
Hench, L. L. 1977 "Physical Chemistry of Glass Surfaces." J. Non-Cryst Solids, 25: 343-369

Plodinec, M. J., C. M. Jantzen, G.G. Wicks. " A Thermodymanic Approach to Prediction of the Stabilty of Proposed Radwaste Glasses." in Nuclear Waste Management, Advances in Ceramics, Vol. 8, eds G.G. Wicks and W.A. Ross, pp. 491-495. The American Ceramic Society, Columbus, Ohio.

LIST OF TABLES

1 Composition of Nuclear Waste Borosilicate Glasses
2 Correlation of Exchanged Fractions and Exchange Frequency, PNL 76-68 70 C
3 Comparison of Continous Flow and Pulsed Flow Leach Tests in Deionzed Water at 90 C
4 Dynamic Leach Tests on SRL TDS-131 in Deionized Water at 70 C
5 High Dilution Test at Various Temperatures SRL TDS-131 Glass, Normalized Leach Rates
6 Comparison Between TDS-131 and DWRG, Leach Data in DI Water, Modified IAEA 90 C
7 Effects of Glass and Leachant Composition On the Results of Flow Tests at 70 C
8 Solubility Tests on Defense Waste Glass, 10 g of -60 +200 Mesh Powder in DI water, 70 C 360 days
9 Dynamic Leach Test on DWRG 70 C
10 Results of Dynamic Leach Test on DWRG in DI Water at 90 C
11 Effects of Ground Water and of Ductile Iron on DWRG Leach Rates in Dynamic Leach Tests, monolithic samples, 90 C
12 Effects of Temperature on DWRG Leach Rates in Dynamic Tests, monolithic samples
13 Speciation of Leached Components of DWRG in Dynamic Tests 90 C
14 Comparison of the results of MCC-1 and Dynamic Leach Test on DWRG, 90 C monolithic samples
15 Molar Compositions of the Glass and of the Saturated Leachant (relative to boron) and Resulting Surface Composition for TDS-131 and DWRG
16 Experimental Run Conditions for the Reaction of DWRG with Water Under Hydrothermal Conditions
17 Solution Analysis for the Reaction of DWRG , Concentration in mg/kg water, 0.10 gm water, one face polished
18 Solution Analysis for the Reaction of DWRG with water, Concentration in mg/kg water, 0.01 gm water, both faces polished
19 Dynamic Leach Test on PNL 76-68 at 70 C
20 Dynamic Leach Test on PGM Glass at 70 C
21 Dynamic Leach Test on CUBS and DWRG , DI Water, 70 C

LIST OF FIGURES

1 Experimental Configuration of Continous Flow Leach Test
2 Experimental Configuration of Pulsed Flow leach Test
3 Results of Frequent Exchange Test on SRL TDS-131, DI water, 90 C
4 Effects of Flow Rate on Leachate Concentration, Measured as a Function of Corrosion Time in Continuous Flow Leach Test on TDS 131 Glass
5 Effect of Flow Rate on the Release Rate of Si and Na in Continuous Flow Leach Test on TDS131 and DWRG
6 Effect of Flow Rate on Leachate pH Measured as a Function of Corrosion Time in Continuous Flow Leach Test on TDS-131 Glass
7 Effect of Flow Rate on Total Mass Loss Measured as a Function of Corrosion Time in Continuous Flow Leach Test on TDS-131
8 SEM-EDS of 28 Day Corroded Samples Exposed to Water at Several Flow Rates
9 Thicknes of Leached Layers Observed in 28 Day Corroded Samples Exposed to Water at Several Flow Rates
10 Effects of Flow Rate and S/V on Total Mass Loss Measured as a Function of Corrosion Time in Continuous Flow leach Test on TDS-131
11 Dependence of Leachate Concentration on Total Exposure Time in a Continuous Flow Rate Leach Test
12 Dependence of Stablized Leachate Concentration on Flow Rate in a Flow Test
13 Dependence of Leach Rate on Flow Rate in a Flow Test
14 Relative Leach Rates of Various Glass Components in a Flow Test
15 Results of Dynamic Leach Test on SRL TDS-131, DI Water, 70 C, Normalized Concentration versus Tr(S/V)
16 Results of Dynamic Leach Test on SRL TDS-131, DI Water, 70 C, Leach Rate versus Contact Time
17 Results of Dynamic Leach Test on SRL TDS-131, DI Water, 70 C, Intermediate Flow Rate, Concentration Versus Total Exposure Time
18 Results of Dynamic Leach Test on SRL TDS-131, DI Water, 70 C, Slow Flow Rate, Concentration Versus Total Exposure Time
19 Results of Frequent Exchange Test on SRL TDS-131 in DI Water at Various Temperatures
20 Concentrations of Si and Na Measured as a Function of Corrosion Time in Continuous Flow Leach Test on TDS-131
21 Leacate pH Values Measured as a Function of Corrosion Time in Continuous Flow Leach Test on DWRG
22 Normalized Total Mass Loss Measured as a Function of Corrosion Time in Continuous Flow Leach Test on DWRG
23 Results of Dynamic Leach Test on DWRG, Ground Water, 70 C, Intermediate Flow Rate, Concentration Versus Total Exposure Time
24 Results of Dynamic Leach Test on DWRG, DI Water, 70 C, Intermediate Flow Rates, Concentraton Versus Total Exposure
25 Results of Dynamic Leach Test on DWRG, DI Water, 70 C, Slow

Flow Rates, Concentration Versus Total Exposure Time
26 Results of Dynamic Leach Test on DWRG, DI Water, 90 C, Normalized Concentration Versus Tr(S/V). Data Points in Parentheses Obtained in the Presence of Iron.
27 Results of Dynamic Leach Test on DWRG, DI Water, 90 C, Leach Rate(fractional loss rate) Versus Contact Time(equivalent flow rate)
28 T-T-T Plot of the Formation of Crystalline Products(DWRG)

Appendix B. Database Report Format Examples, Keyword Checklist, and Help Screens

WORK.REF.. PACKIT...

0011 Cit.no: 0011 | Report.no: UCRL-92096 | Authors: Oversby, V. M.;Wilson, C. N. | Title: Derivation of Waste Package Source Term for NNWSI from the Results of Laboratory Experiments | Type.pub: 1 | Date: 09-85 | Start.date: 01-30-87 | Change.date: 01-30-87

0012 Cit.no: 0012 | Report.no: UCRL-91464 | Authors: Wilson, C. N.;Oversby, V. M. | Title: Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater | Contractor: Lawrence Livermore National Laboratory | Sponsor: U.S. Department of Energy | Type.pub: 1 | Date: 03-85 | Contract.no: W-7405-Eng-48 | Abstract: The Nevada Nuclear Waste Storage Investigations Project (NNWSI) is studying the suitability of the welded devitrified Topopah Spring tuff at Yucca Mountain, Nye County, Nevada, for potential use as a high level nuclear waste repository. In support of the Waste Package task of NNWSI, tests have been conducted under ambient air environment to measure radionuclide release from two pressurized water reactor (PWR) spent fuels in water obtained from the J-13 well near the Yucca Mountain site. Four specimen types, representing a range of fuel physical conditions that may exist in a failed waste canister containing a limited amount of water were tested. The specimen types were: 1) fuel rod sections split open to expose bare fuel particles, 2) rod sections with water-tight end fittings with a 2.5-cm long by 150-Ìm wide slit through the cladding, 3) rod sections with water-tight end fittings and two 200-Ìm diameter holes through the cladding, and 4) undefected rod segments with water-tight end fittings. Radionuclide release results from the first 223-day runs on H.B. Robinson spent fuel specimens in J-13 water are reported and compared to results from a previous test series in which similar Turkey point reactor spent fuel specimens were tested in deionized water. Selected initial results are also given for Turkey Point fuel specimens tested in J-13 water. Results suggest that the actinides Pu, Am, Cm and Np are released congruently with U as the UO\/2/\ spent fuel matrix dissolves. Fractional release of /\137\/Cs and /\99\/Tc was greater than that measured for the actinides. Generally, lower radionuclide releases were measured for the H. B. Robinson fuel in J-13 water than for Turkey Point Fuel in deionized water. | Start.date: 01-30-87 | Change.date: 01-30-87

0013 Cit.no: 0013 | Report.no: UCRL 94222 | Authors: Oversby, V. M. | Title: Important Radionuclides in High Level Nuclear Waste Disposal: Determination Using a Comparison of the EPA and NRC Regulations | Contractor: Lawrence Livermore National Laboratory | Sponsor: U.S. Department of Energy | Type.pub: 1 | Date: 02-86 | Contract.no: W-7405-Eng-48 | Abstract: The performance objective for the engineered barrier system given in the NRC regulations (10CFR60) is used to determine a maximum release rate for each significant radionuclide for a generic repository containing PWR spent fuel. This release rate, integrated over the times during which release would occur, is then compared to the EPA requirements on limitation of total releases to the accessible environment. The amount by which the releases allowed under the NRC regulations exceeds the EPA requirements is an indication of the importance of the radionuclide for performance assessment purposes. Nuclides exceed those allowed by EPA to the accessible environment will need to be controlled either by

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limiting their release at the EBS boundary to values that are lower than the NRC requirements or by reducing the amounts of these nuclides that reach the environment by proceses that occur during transport. The simplest cast, which assumed only the minimum performance require in 10CFR60 on control of release rates, results in the identification of 17 chemical elements for which data on solubility and sorption would be needed for use in site performance assessment. Of these, americium and plutonium are by far the most important. The other actinides, carbon, and nickel are also important. If the assumption of congruent dissolution is imposed, with a 1% rapid release spike for cesium, iodine, carbon, and tchnetium, the list of elements reduces to 13, with iodine, cesium, selenium, and palladium being eliminated from the list. The importance of americium is greatly reduced in this case and plutonium becomes the most important element. A final analysis, which assumed a congruent dissolution rate of one part in 1,000,000 per year, results in a list of at most seven important elements. With reasonable assumptions the list can be narrowed to just americium and plutonium. All of the considerations point to the importance of understanding the behavior of americium and plutonium under conditions that are relevant to the waste form dissolution process and under processes that might pertain during transport from the repository to the accessible environment. | Start.date: 01-30-87 | Change.date: 01-30-87 | Filelocation: UCRL-94222

0014 Cit.no: 0014 | Report.no: HEDL-7452 | Authors: Einzinger, R. E.;Woodley, R. E. | Title: Evaluation of the Potential for Spent Fuel Oxidation under Tuff Repository Conditions | Contractor: Hanford Engineering Development Laboratory | Sponsor: U.S. Department of Energy | Type.pub: 1 | Date: 03-85 | Contract.no: DE-AC06-76FF02170 | Pages: 48 | Start.date: 01-30-87 | Change.date: 01-30-87 | Filelocation: HEDL-7452

0015 Cit.no: 0015 | Report.no: UCID-20174 | Authors: Glass, R. S.;Overturf, G. E.;Garrison, R. E.;McCright, R. D. | Title: Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff | Contractor: Lawrence Livermore National Laboratory | Sponsor: U.S. Department of Energy | Type.pub: 1 | Date: 06-84 | Contract.no: W-7405-Eng-48 | Pages: 39 | Abstract: Long-term geological disposal of nuclear waste requires corrosion-resistant canister materials for encapsulation. Several austenitic stainless steels are under consideration for such purposes for the disposal of high-level waste at the candidate repository site located at Yucca Mountain, Nevada. With regard to corrosion considerations, a worst case scenario at this prospective repository location would result from the intrusion of vadose water. This preliminary study focuses on the electrochemical and corrosion behavior of the candidate canister materials under worst-case repository environments. Electrochemical parameters related to localized attack (e.g., pitting potentials) and the electrochemical corrosion rates have been examined. | Availability: NTIS | Start.date: 01-30-87 | Change.date: 01-30-87 | Ntis.no: DE85001768/XAB

5 Records Processed

PAGE 1 00:33:15 02 JAN 1980

0011 1CIT.NO:I 0011 | 1REPORT.NO:I UCRL-92096 | 1AUTHORS:I Oversby, V. M.;Wilson, C. N. | 1TITLE:I Derivation of Waste Package Source Term for NNWSI from the Results of Laboratory Experiments | 1TYPE.PUB:I 1 | 1DATE:I 09-85 | 1START.DATE:I 01-30-87 | 1CHANGE.DATE:I 01-30-87

0012 1CIT.NO:I 0012 | 1REPORT.NO:I UCRL-91464 | 1AUTHORS:I Wilson, C. N.;Oversby, V. M. | 1TITLE:I Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater | 1CONTRACTOR:I Lawrence Livermore National Laboratory | 1SPONSOR:I U.S. Department of Energy | 1TYPE.PUB:I 1 | 1DATE:I 03-85 | 1CONTRACT.NO:I W-7405-Eng-48 | 1ABSTRACT:I The Nevada Nuclear Waste Storage Investigations Project (NNWSI) is studying the suitability of the welded devitrified Topopah Spring tuff at Yucca Mountain, Nye County, Nevada, for potential use as a high level nuclear waste repository. In support of the Waste Package task of NNWSI, tests have been conducted under ambient air environment to measure radionuclide release from two pressurized water reactor (PWR) spent fuels in water obtained from the J-13 well near the Yucca Mountain site. Four specimen types, representing a range of fuel physical conditions that may exist in a failed waste canister containing a limited amount of water were tested. The specimen types were: 1) fuel rod sections split open to expose bare fuel particles, 2) rod sections with water-tight end fittings with a 2.5-cm long by 150-Ìm wide slit through the cladding, 3) rod sections with water-tight end fittings and two 200-Ìm diameter holes through the cladding, and 4) undefected rod segments with water-tight end fittings. Radionuclide release results from the first 223-day runs on H.B. Robinson spent fuel specimens in J-13 water are reported and compared to results from a previous test series in which similar Turkey point reactor spent fuel specimens were tested in deionized water. Selected initial results are also given for Turkey Point fuel specimens tested in J-13 water. Results suggest that the actinides Pu, Am, Cm and Np are released congruently with U as the UO\/2/\ spent fuel matrix dissolves. Fractional release of /\137\/Cs and /\99\/Tc was greater than that measured for the actinides. Generally, lower radionuclide releases were measured for the H. B. Robinson fuel in J-13 water than for Turkey Point Fuel in deionized water. | 1START.DATE:I 01-30-87 | 1CHANGE.DATE:I 01-30-87

0013 1CIT.NO:I 0013 | 1REPORT.NO:I UCRL 94222 | 1AUTHORS:I Oversby, V. M. | 1TITLE:I Important Radionuclides in High Level Nuclear Waste Disposal: Determination Using a Comparison of the EPA and NRC Regulations | 1CONTRACTOR:I Lawrence Livermore National Laboratory | 1SPONSOR:I U.S. Department of Energy | 1TYPE.PUB:I 1 | 1DATE:I 02-86 | 1CONTRACT.NO:I W-7405-Eng-48 | 1ABSTRACT:I The performance objective for the engineered barrier system given in the NRC regulations (10CFR60) is used to determine a maximum release rate for each significant radionuclide for a generic repository containing PWR spent fuel. This release rate, integrated over the times during which release would occur, is then compared to the EPA requirements on limitation of total releases to the accessible environment. The amount by which the releases allowed under the NRC regulations exceeds the EPA requirements is an indication of the importance of the radionuclide for performance assessment purposes. Nuclides exceed those allowed by EPA to the accessible environment will need to be controlled either by limiting their release at the EBS boundary to values that are lower than the NRC requirements or by reducing the amounts of these nuclides that reach the environment by proceses that occur during transport. The simplest cast, which assumed only the minimum performance require in 10CFR60 on control of release rates, results in the identification of 17 chemical elements for which data on solubility and sorption would be needed for use in site performance assessment. Of these, americium and plutonium are by far the most important. The other actinides, carbon, and nickel are also important. If the assumption of congruent dissolution is imposed, with a 1% rapid release spike for cesium, iodine, carbon, and tchnetium, the list of elements reduces to 13, with iodine, cesium, selenium, and palladium being eliminated from the list. The importance of americium is greatly reduced in this case and plutonium becomes the most important element. A final analysis, which assumed a congruent dissolution rate of one part in 1,000,000 per year, results in a list of at most seven important elements. With reasonable assumptions the list can be narrowed to just americium and plutonium. All of the considerations point to the importance of understanding the behavior of americium and plutonium under conditions that are relevant to the waste form dissolution process and under processes that might pertain during transport from the repository to the accessible environment. | 1START.DATE:I 01-30-87 | 1CHANGE.DATE:I 01-30-87 | 1FILELOCATION:I UCRL-94222

0014 1CIT.NO:I 0014 | 1REPORT.NO:I HEDL-7452 | 1AUTHORS:I Einzinger, R. E.;Woodley, R. E. | 1TITLE:I Evaluation of the Potential for Spent Fuel Oxidation under Tuff Repository Conditions | 1CONTRACTOR:I Hanford Engineering Development Laboratory | 1SPONSOR:I U.S. Department of Energy | 1TYPE.PUB:I 1 | 1DATE:I 03-85 | 1CONTRACT.NO:I DE-AC06-76FF02170 | 1PAGES:I 48 | 1START.DATE:I 01-30-87 | 1CHANGE.DATE:I 01-30-87 | 1FILELOCATION:I HEDL-7452

0015 1CIT.NO:I 0015 | 1REPORT.NO:I UCID-20174 | 1AUTHORS:I Glass, R. S.;Overturf, G. E.;Garrison, R. E.;McCright, R. D. | 1TITLE:I Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff | 1CONTRACTOR:I Lawrence Livermore National Laboratory | 1SPONSOR:I U.S. Department of Energy | 1TYPE.PUB:I 1 | 1DATE:I 06-84 | 1CONTRACT.NO:I W-7405-Eng-48 | 1PAGES:I 39 | 1ABSTRACT:I Long-term geological disposal of nuclear waste requires corrosion-resistant canister materials for encapsulation. Several austenitic stainless steels are under consideration for such purposes for the disposal of high-level waste at the candidate repository site located at Yucca Mountain, Nevada. With regard to corrosion considerations, a worst case scenario at this prospective repository location would result from the intrusion of vadose water. This preliminary study focuses on the electrochemical and corrosion behavior of the candidate canister materials under worst-case repository

WORK.REF.. FORMAT TEXT FIELDS..

CIT.NO: 0011
REPORT.NO: UCRL-92096
AUTHORS: Oversby, V. M.; Wilson, C. N.
TITLE: Derivation of Waste Package Source Term for NNWSI from the Results of Laboratory Experiments
TYPE.PUB: 1
DATE: 09-85
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
0011 RECORD END

CIT.NO: 0012
REPORT.NO: UCRL-91464
AUTHORS: Wilson, C. N.; Oversby, V. M.
TITLE: Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 03-85
CONTRACT.NO: W-7405-Eng-48
ABSTRACT: The Nevada Nuclear Waste Storage Investigations Project (NNWSI) is studying the suitability of the welded devitrified Topopah Spring tuff at Yucca Mountain, Nye County, Nevada, for potential use as a high level nuclear waste repository. In support of the Waste Package task of NNWSI, tests have been conducted under ambient air environment to measure radionuclide release from two pressurized water reactor (PWR) spent fuels in water obtained from the J-13 well near the Yucca Mountain site. Four specimen types, representing a range of fuel physical conditions that may exist in a failed waste canister containing a limited amount of water were tested. The specimen types were: 1) fuel rod sections split open to expose bare fuel particles, 2) rod sections with water-tight end fittings with a 2.5-cm long by 150-Ìm wide slit through the cladding, 3) rod sections with water-tight end fittings and two 200-Ìm diameter holes through the cladding, and 4) undefected rod segments with water-tight end fittings.

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START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
0012 RECORD END

CIT.NO: 0013

REPORT.NO: UCRL 94222
AUTHORS: Oversby, V. M.
TITLE: Important Radionuclides in High Level Nuclear Waste Disposal: Determination Using a Comparison of the EPA and NRC Regulations
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 02-86
CONTRACT.NO: W-7405-Eng-48
ABSTRACT: The performance objective for the engineered barrier system given in the NRC regulations (10CFR60) is used to determine a maximum release rate for each significant radionuclide for a generic repository containing PWR spent fuel. This release rate, integrated over the times during which release would occur, is then compared to the EPA requirements on limitation of total releases to the accessible environment. The amount by which the releases allowed under the NRC regulations exceeds the EPA requirements is an indication of the importance of the radionuclide for performance assessment purposes. Nuclides exceed those allowed by EPA to the accessible environment will need to be controlled either by limiting their release at the EBS boundary to values that are lower than the NRC requirements or by reducing the amounts of these nuclides that reach the environment by processes that occur during transport.

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START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
FILELOCATION: UCRL-94222
0013 RECORD END

CIT.NO: 0014
REPORT.NO: HEDL-7452
AUTHORS: Einzinger, R. E.; Woodley, R. E.
TITLE: Evaluation of the Potential for Spent Fuel Oxidation under Tuff Repository Conditions

WORK.REF.. FORMAT TEXT FIELDS..

CONTRACTOR: Hanford Engineering Development Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 03-85
CONTRACT.NO: DE-AC06-76FF02170
PAGES: 48
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
FILELOCATION: HEDL-7452
0014 RECORD END

CIT.NO: 0015
REPORT.NO: UCID-20174
AUTHORS: Glass, R. S.; Overturf, G. E.; Garrison, R. E.; McCright, R. D.
TITLE: Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 06-84
CONTRACT.NO: W-7405-Eng-48
PAGES: 39
ABSTRACT: Long-term geological disposal of nuclear waste requires corrosion-resistant canister materials for encapsulation. Several austenitic stainless steels are under consideration for such purposes for the disposal of high-level waste at the candidate repository site located at Yucca Mountain, Nevada. With regard to corrosion considerations, a worst case scenario at this prospective repository location would result from the intrusion of vadose water. This preliminary study focuses on the electrochemical and corrosion behavior of the candidate canister materials under worst-case repository environments. Electrochemical parameters related to localized attack (e.g., pitting potentials) and the electrochemical corrosion rates have been examined.
AVAILABILITY: NTIS
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
NTIS.NO: DE85001768/XAB
0015 RECORD END

5 Records Processed

PAGE 1 00:26:23 02 JAN 1980

CIT.NO: 0011
REPORT.NO: UCRL-92096
AUTHORS: Oversby, V. M.; Wilson, C. N.
TITLE: Derivation of Waste Package Source Term for NNWSI from the Results of Laboratory Experiments
TYPE.PUB: 1
DATE: 09-85
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0011 RECORD END

CIT.NO: 0012
REPORT.NO: UCRL-91464
AUTHORS: Wilson, C. N.; Oversby, V. M.
TITLE: Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
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DATE: 03-85
CONTRACT.NO: W-7405-Eng-48
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CIT.NO: 0013
REPORT.NO: UCRL 94222
AUTHORS: Oversby, V. M.
TITLE: Important Radionuclides in High Level Nuclear Waste Disposal: Determination Using a Comparison of the EPA and NRC Regulations
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 02-86
CONTRACT.NO: W-7405-Eng-48
ABSTRACT: The performance objective for the engineered barrier system given in the NRC regulations (10CFR60) is used to determine a maximum release rate for each significant radionuclide for a generic repository containing PWR spent fuel. This release rate, integrated over the times during which release would occur, is then compared to the EPA requirements on limitation of total releases to the accessible environment. The amount by which the releases allowed under the NRC regulations exceeds the EPA requirements is an indication of the importance of the radionuclide for performance assessment purposes. Nuclides exceed those allowed by EPA to the accessible environment will need to be controlled either by limiting their release at the EBS boundary to values that are lower than the NRC requirements or by reducing the amounts of these nuclides that reach the environment by proceses that occur during transport.

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FILELOCATION: UCRL-94222
0013 RECORD END

CIT.NO: 0014
REPORT.NO: HEDL-7452
AUTHORS: Einzinger, R. E.; Woodley, R. E.

TITLE: Evaluation of the Potential for Spent Fuel Oxidation under Tuff Repository Conditions
CONTRACTOR: Hanford Engineering Development Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 03-85
CONTRACT.NO: DE-AC06-76FF02170
PAGES: 48
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
FILELOCATION: HEDL-7452
0014 RECORD END

CIT.NO: 0015
REPORT.NO: UCID-20174
AUTHORS: Glass, R. S.; Overturf, G. E.; Garrison, R. E.; McCright, R. D.

TITLE: Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 06-84
CONTRACT.NO: W-7405-Eng-48
PAGES: 39
ABSTRACT: Long-term geological disposal of nuclear waste requires corrosion-resistant canister materials for encapsulation. Several austenitic stainless steels are under consideration for such purposes for the disposal of high-level waste at the candidate repository site located at Yucca Mountain, Nevada. With regard to corrosion considerations, a worst case scenario at this prospective repository location would result from the intrusion of vadose water. This preliminary study focuses on the electrochemical and corrosion behavior of the candidate canister materials under worst-case repository environments. Electrochemical parameters related to localized attack (e.g., pitting potentials) and the electrochemical corrosion rates have been examined.
AVAILABILITY: NTIS
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
NTIS.NO: DE85001768/XAB
0015 RECORD END

5 Records Processed

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WORK.REF.. ....

CIT.NO: 0011
REPORT.NO: UCRL-92096
AUTHORS: Oversby, V. M.; Wilson, C. N.
TITLE: Derivation of Waste Package Source Term for NNWSI from the Results of Laboratory Experiments
TYPE.PUB: 1
DATE: 09-85
START.DATE: 01-30-87
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0011 RECORD END

WORK.REF.. ....

CIT.NO: 0012
REPORT.NO: UCRL-91464
AUTHORS: Wilson, C. N.; Oversby, V. M.
TITLE: Radionuclide Release from PWR Fuels in a Reference Tuff Repository Groundwater
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 03-85
CONTRACT.NO: W-7405-Eng-48

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REPORT.NO: UCRL 94222
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SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
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START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
FILELOCATION: UCRL-94222
0013 RECORD END

WORK.REF.. ....

CIT.NO: 0014
REPORT.NO: HEDL-7452
AUTHORS: Einzinger, R. E.; Woodley, R. E.
TITLE: Evaluation of the Potential for Spent Fuel Oxidation under Tuff Repository Conditions
CONTRACTOR: Hanford Engineering Development Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 03-85
CONTRACT.NO: DE-AC06-76FF02170
PAGES: 48
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
FILELOCATION: HEDL-7452

0014 RECORD END

WORK.REF.. ....

CIT.NO: 0015
REPORT.NO: UCID-20174
AUTHORS: Glass, R. S.; Overturf, G. E.; Garrison, R. E.; McCright, R. D.

TITLE: Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff
CONTRACTOR: Lawrence Livermore National Laboratory
SPONSOR: U.S. Department of Energy
TYPE.PUB: 1
DATE: 06-84
CONTRACT.NO: W-7405-Eng-48
PAGES: 39
ABSTRACT: Long-term geological disposal of nuclear waste requires corrosion-resistant canister materials for encapsulation. Several austenitic stainless steels are under consideration for such purposes for the disposal of high-level waste at the candidate repository site located at Yucca Mountain, Nevada. With regard to corrosion considerations, a worst case scenario at this prospective repository location would result from the intrusion of vadose water. This preliminary study focuses on the electrochemical and corrosion behavior of the candidate canister materials under worst-case repository environments. Electrochemical parameters related to localized attack (e.g., pitting potentials) and the electrochemical corrosion rates have been examined.
AVAILABILITY: NTIS
START.DATE: 01-30-87
CHANGE.DATE: 01-30-87
NTIS.NO: DE85001768/XAB
0015 RECORD END

5 Records Processed

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Keyword Checklist Tree 3/2/87

KEYWORDS	{FIELD NAME}	
A. TECHNICAL DESCRIPTION OF REPORT		
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2. applicability of data to licensing	{LIC.APP}	LIST2
3. model/methodology	{MODEL}	LIST3
4. measurement	{MEASURE}	LIST5
5. computer programs	{PROGRAM}	LIST4
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2. geographic location of field site	{ENV.LOC}	LIST7
3. gaseous environment	{ENV.GAS}	LIST8
4. aqueous environment	{ENV.H2O}	LIST9
5. ionic or other chemical species present	{ENV.ION}	LIST10
6. solid materials environment	{ENV.SOLID}	LIST11
7. radiation environment	{ENV.RAD}	LIST12
8. experimental/test conditions	{ENV.COND}	LIST13
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2. specific material designation	{MAT.DES}	LIST15
3. condition prior to test	{MAT.COND}	LIST16
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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. condition prior to test {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. test specimen specifications {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. waste form {MAT.WASTE}

__commercial high level waste (CHLW)
__defense high level waste (DHLW)
__spent fuel
__spent fuel (PWR reactor)
__spent fuel (BWR reactor)
__other____

2. radionuclides {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____

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:HELP SELECT

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¤ The SELECT command is used to create a list of record ids that meet          ¤
¤ specific criteria. The id list can then be "passed" to another REVELATION   ¤
¤ command.  This enables the user to create sublists from larger lists.        ¤
¤                                                                              ¤
¤ Example:                                                                     ¤
¤                                                                              ¤
¤ :SELECT NRC.ABS WITH ABSTRACT CONTAINING "Yucca Mountain"                    ¤
¤                                                                              ¤
¤ :SELECT NW.REF WITH PUB.TITLE CONTAINING 'Methodologies' BY PUB.TITLE        ¤
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Proper Syntax of the command :
:SELECT {DICT} {filename} {with.clause} {by.clause}

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:HELP DELETE-LIST

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¤ The DELETE-LIST command is used to remove a saved list from the LISTS file. ¤
¤ A saved list was created using the SAVE-LIST command.                        ¤
¤                                                                              ¤
¤ Example:                                                                     ¤
¤                                                                              ¤
¤ :DELETE-LIST NRC-LIST1A                                                      ¤
¤ 'NRC-LIST1A' deleted from the 'LISTS' file                                   ¤
¤                                                                              ¤
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Proper Syntax of the command :
:DELETE-LIST (listname)

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:HELP SAVE-LIST

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¤ The SAVE-LIST command is used to save a list of record ids after a           ¤
¤ SELECT or a SSELECT. The list can then be edited using EDIT or               ¤
¤ recalled using GET-LIST.                                                     ¤
¤                                                                              ¤
¤ Example:                                                                     ¤
¤                                                                              ¤
¤ :SAVE-LIST AUTHORS-LIST                                                      ¤
¤                                                                              ¤
¤ 'AUTHORS-LIST' saved on 'LISTS' file                                         ¤
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Proper Syntax of the command :
:SAVE-LIST (listname)

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:HELP GET-LIST

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¤ The GET-LIST command is used to retrieve a saved list of record ids from    ¤
¤ the LISTS file for processing with a SELECT, LIST, or EDIT command.          ¤
¤ The list of record ids was created with the SAVE-LIST command.               ¤
¤                                                                              ¤
¤ Example:                                                                     ¤
¤                                                                              ¤
¤ :GET-LIST NRC-LIST1A                                                         ¤
¤ 25 records selected                                                          ¤
¤                                                                              ¤
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Proper Syntax of the command :
:GET-LIST (listname)

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:HELP LIST

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¤ The LIST command is used to produce a columnar report from a file.          ¤
¤ If you want to produce sublists from larger lists use the SELECT command.   ¤
¤                                                                             ¤
¤ Example:                                                                    ¤
¤                                                                             ¤
¤ :LIST NRC.ABS WITH CIT.NO = "0005" PACK                                     ¤
¤                                                                             ¤
¤ :LIST NRC.REF WITH PUB.NOS STARTING WITH 'UCID' FTF                         ¤
¤                                                                             ¤
¤ :LIST NRC.REV BY SOURCE WITH DATA.TYPE CONTAINING 'radionuclides' FTF (P)   ¤
¤                                                                             ¤
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Proper Syntax of the command :
:LIST (DICT) (filename) (with.clause) (by.clause) (format) (heading/footing)

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:HELP COUNT

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¤ The COUNT command allows the user to count the number of records that       ¤
¤ meet a certain selection criteria.  Examples:                               ¤
¤                                                                             ¤
¤ :COUNT NRC.ABS WITH ABSTRACT CONTAINING "copper-base alloys"                ¤
¤ 38 RECORDS COUNTED                                                          ¤
¤                                                                             ¤
¤ :COUNT NRC.REV                                                              ¤
¤ 154 RECORDS COUNTED                                                         ¤
¤                                                                             ¤
¤ :COUNT NRC.REF WITH PUB.ORG CONTAINING "OAK RIDGE NATIONAL LABORATORY"      ¤
¤ 25 RECORDS COUNTED                                                          ¤
¤                                                                             ¤
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Proper Syntax of the command :
:COUNT (filename) (with.clause)

:
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:DISPLAY.FIELDS

TYPE IN THE FILE NAME CONTAINING THE FIELDS TO BE QUERIED ON:
?NW.REF
```

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:DISPLAY.FIELDS

TYPE IN THE FILE NAME CONTAINING THE FIELDS TO BE QUERIED ON:
?NW.REF

===========================================================================
                        FIELDS CONTAINED IN FILE: NW.REF
---------------------------------------------------------------------------

AUTHORS        CHANGE.DATE    CHAP.NO        CIT.NO         CIT.REL
CONTRACT.NO    EDITORS        OTHER.DATE     PAT.NO         PUB.AVAIL
PUB.CODEN      PUB.DATE       PUB.ISSN       PUB.ISSUE      PUB.NOS
PUB.ORG        PUB.PAGE       PUB.SPONSOR    PUB.TITLE      PUB.TYPE
PUB.VOL        SITE.REL       START.DATE     SUB.TITLE

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Appendix C. Visit by Dr. R. Ricker to Battelle Columbus Laboratories, September 5, 1986

Subject: Trip Report by R.E. Ricker on Visit to Battelle Columbus Laboratories (BCL) September 5, 1986

Purpose: To Obtain Background Information on the SRP by Attending Talks on Current NRC Office of Waste Management Sponsored Research at BCL

Four presentations on the status of NRC Office of Waste Management research programs on the long term performance of candidate materials for high-level waste packages at Battelle Columbus Laboratories were given at BCL on September 5, 1986:

(A) "Long-Term Performance of Materials Used for High-Level Waste Packaging, Project Overview" by D. Stahl, presentation by D. Stahl

This presentation was a basic overview of the entire NRC project at Battelle Columbus Laboratories (16 overheads).

(B) "Container Materials, Overpack Corrosion" by J.A. Beavers, N.G. Thompson and R.N. Parkins (consultant), presentation by John Beavers

This presentation reviewed the accomplishments of the first two years and the ongoing work in the third and fourth year of the overpack corrosion part of this program (71 Overheads).

(C) "Hydrogen Embrittlement of Containers for High-Level Waste-Containment" by H.J. Cialone, presentation by Henry Cialone

This presentation reviewed the results at Battelle on the effects of hydrogen on the properties of cast steels and Armco CP iron. Then, the plans to complete the work on the cast commercial-purity iron and to study the influence of welding on the sensitivity of cast steel were discussed (37 overheads)

(D) "Analysis of Container Corrosion" by A.J. Markworth, J.E. McCoy and D.D. Macdonald (consultant), presentation by Alan J. Markworth

This presentation briefly reviewed the work on general corrosion and then focused on the pit modeling work. Discussion on the relative influence of active vs. inert walls on the pitting model provoked numerous questions. (50 overheads in handout, 21 covered in presentation)

This visit was informative and the overall quality of effort at BCL is good. The overheads used in the presentations were handed out and copies (with my notes) are available (NBS-SRP Shelf list No. 72a-d) for anyone interested.

Appendix D. Fall Meeting of the Materials Research Society, Boston, MA, December 1 through December 4, 1986.

Report - 1986 Fall Meeting of the Materials Research Society

Symposium L: Scientific Basis for Nuclear Waste Management X

This symposium was held on Dec. 1 through Dec. 4, 1986 in Boston MA, and proceedings will be published as Volume 84 of the Materials Research Society Symposia Proceeding Series. The symposium received support from both the Department of Energy and the Nuclear Regulatory Commission. An estimate of the cumulative attendance over the four days would be 200 to 300 attendees. The Symposium consisted of eight sessions of oral presentations and eight poster sessions. The papers covered many areas of nuclear waste disposal which currently are thought to be important. NBS participated in the meeting as attendees, A. Fraker, E. Plant, session chairman, C. Interrante, author and presenter of a paper, U. Bertocci. It is anticipated that the publication of the proceedings of the MRS-87 Symposium on Scientific Bases for Nuclear Waste Management would be available in March 1987. This tentative date may be met due to th fine efforts of J. Bates and We. Seefeld of Argonne National Laboratory, who planned to have edited and sent camera-ready copy to the publisher by January 1, 1987.

The symposium covered a wide area from statistics to corrosion science, and from inorganic chemistry and geochemistry to mechanical engineering, so that it was impossible for the typical attendee to be conversant with more than one or two of the subjects. A significant part of the content of the papers was hardly new; for instance the work presented on corrosion of copper-base alloys was essentially identical to that presented in Houston last spring.

It was interesting to hear some of the work being carried out in foreign countries, notably Germany, Sweden, and the United Kingdom. For example, the modeling work done at Harwell by G.P. Marsh, S. M. Sharland and coworkers is raising some important points concerning the extent of localized corrosion which can be expected in repository conditions. The session topics and brief discussions are presented as follow. Meeting with the authors and hearing their presentations was very informative. Much work has been done and much much more seems needed to solve the problems associated with understanding the waste package.

Session L1 - Plenary Session on Long Term Projection of Materials Interactions

Considerations for licensing and for extrapolating test for long term projections were presented by T. C. Johnson,

K. Chang, T. L. Jungling, L. S. Person, C. H. Peterson, J. Vogelwede and E. Wick of the Div. of Waste Management, Nuclear Regulatory Commission, Washington, DC. Other aspects of long term waste disposal, including materials performance, validation of models, nuclear waste glass, repository material interactions, natural analogues for predicting corrosion of borosilicate glass and radiation effects in ceramics, were discussed in four other talks of this session, and Jeffrey Fong of the National Bureau of Standards discussed database strategy for engineering decision making. He defined a global database and spoke to the importance of judgement on the part of the engineers who must use the database. As the database is insufficient, due to limitations in time and money, good engineering judgement must be used in most important decisions affecting the public.

Session L2 - Waste Form Performance: Spent Fuel

V. M. Oversby talked about the requirement of the EPA and NRC and the calculations needed to assure required radionuclide containment for 10,000 years. Knowledge of waste form performance and of the acting physical and chemical processes are needed to make these calculations. Much of her talk was collected from previous NNWSI publications. Another paper, by J. J. Mahoney, reported on behavior of spent fuel in a simulated basalt repository environment and found that actinide release was reduced in the presence of basalt (This could be due to experimental design.) and that U concentration may be controlled by solubility. Other papers (S. Sunder, et.al. and H. Christensen) reported on the oxidation of UO_2 by alpha-radiolysis products and the retardation of this oxidation by reducing ions such as H_2 or Fe^{2+}. C. N. Wilson reported on radionuclide release from spent fuel with various cladding defects in J-13 water. Transport analysis of radionuclide transport from holes in the canister was presented by P. L. Chambre, et.al.

Session L3 - Metal Corrosion

The papers in this session dealt with local and general corrosion. M. McNeil reviewed statistics of pitting, pit depths and distributions in carbon steel and related findings to repository conditions and needs for future experimental data. U. Bertocci, et.al. reported on the statistical analyses of passive current breakdown events decay rates to predict pitting initiation. D. Merz and M. Lewis presented three BWIP/MCC reference test procedures along with statistical treatment of data and of interlaboratory variation. The three tests discussed were the BWIP/MCC-105.5 Air/Steam Test, the BWIP/MCC-105.1 Static Pressure Vessel Test and the BWIP/MCC-105.4 Flowby Test. G. P. Marsh, et.al. of Harwell discussed localized and general corrosion of carbon steel containers for nuclear waste and concluded that localized corrosion would not occur if oxidizing conditions were not present. Marsh, et al. also consider the long term corrosion of carbon steel containers for nuclear waste in a granitic repository. Under such conditions carbon steel may exhibit general, and localized corrosion or passive behavior depending on the exact composition and redox potential of the groundwater contacting the containers, localized corrosion being of most concern because it has the fastest propagation rate. It is well established, however, that such localized corrosion is only possible when the environment is sufficiently oxidizing to maintain a positive potential gradient between the cathodic surface and the corrosion sites, which requires that species with oxidizing potentials greater than water need to be present. This fact provides a basis for estimating the periods during which containers may be subject to localized and subsequently to general or passive corrosion, and hence for making an overall assessment of the metal allowance required for a specified container life. Such an analysis has been made in which the rate of consumption of oxygen and oxidising radiolysis products is equated to the leakage current from a passive steel surface. It is concluded that localized corrosion is only possible for a small proportion of the required container life of 500-1000 years. Thus, this work stresses that the form and rate of corrosion must be known over long times, using firm mechanistic understanding, without which local corrosion must be assumed and when this is done, an unacceptable conclusion is reached: unrealisticly thick containers must be used. Hence, the kind of corrosion and its form are being investigated to permit general corrosion modelling. S. G. Pitman's examination of stress corrosion cracking (SCC) susceptibility of cast mild steel found no tendency for SCC. However, there was some loss of ductility which indicates that this material and its resistance to SCC or other cracking mechanisms needs further study. Under fatigue loading, he showed da/dn values could be higher (faster rate of crack growth) in brine than in air, indicating that corrosion fatigue values are conservative measures of crack extension rate. Metal matrix studies reported by P. M. Mathew and P. A. Krueger were conducted on lead, zinc and aluminum-7%silicon. Lead showed almost no corrosion while the zinc and aluminum-7%silicon corroded and pitted.

Session L4 - Low Level Waste Materials

Papers in this session dealt with chemical aspects such as leaching (S. Hoyle and M. W. Grutzeck, and A. T. Jakubick, et. al.) and solubility modelling (Urs. R. Berner, E. O. Glasser, et. al.) of cement containment of nuclear waste, and with a report of a preliminary evaluation of near field radioactivity release from neutron activated reactor parts buried in a shallow site at Oak Ridge, Tennessee. B. G. J. Thompson, et. al., reported on independent efforts of the United Kingdom's Dept. of the Environment to evaluate proposals from the nuclear industry for the burial of nuclear waste and outlined their program.

Poster Sessions L5 through 12

Poster sessions were held in the evening. These sessions provided a good opportunity to meet the authors and to discuss the work. Papers from the poster sessions will be included in the published proceedings of the meeting and cover essentially the same general areas as the oral sessions.

Session L5 - Waste Form Performance: Spent Fuel

Session L6 - Metal Corrosion

A poster by Sharland discussed several techniques of modelling long-term pit propagation in waste canisters. The complexity of the problem has lead to the necessity for a number of physical and chemical approximations in the modelling. The applicability and ranges of validity of several of the more common approximations investigated, and the predictions with experimental pit growth rates compared. An investigation of the sensitivity of the models to the various empirical input parameters indicates which ones need to be determined most accurately. Finally, a steady-state model of cavity propagation and a more sophisticated numerical model which involves a finite element technique and yields both steady-state and time-dependent solutions was discussed.

Another model for predicting corrosion of nuclear waste containers has been presented by J. C. Walton and coworkers. This mechanistic model for corrosion of high-level waste containers in an aqueous environment is being formulated by the Basalt Waste Isolation Project. Electrochemical charge-transfer reaction kinetics are used in conjunction with transport of reactants and products due to Fickian diffusion, hydrodynamic dispersion and fluid advection. The model provides for a choice of oxidants from among water, oxygen, sulfate, and radiolysis products. The model user may elect to study effects of any combination of these oxidants.

Session L7 - Low-Level Waste and Materials
Session L8 - Materials Interactions
Session L9 - Waste Form Performance: Glass
Session L10 - Radiation Effects
Session L11 - Groundwater Chemistry and Interactions
Session L12 - Rock/Backfill Performance

Session L13 - Materials Interactions

This session had papers dealing with degradation and leaching of nuclear waste glass (R. B. Heimann, F. J. Ryerson, et al. and R. Bazan, et al.) and effects of steel and iron corrosion products on the nuclear waste glass performance. G. Bart, et al. and B. Grambow, et al. discussed the effects of steel corrosion products on the leaching process. There appears to be some evidence that the rate of leaching is increased in the presence of steel corrosion products. One of the papers was by T. E. Jones, et. al. and described development of a tracer injection system for characterizing reactions of waste materials under hydrothermal conditions.

Session L14 - Waste Performance: Glass

The papers in this session dealt with waste glass leaching. Chemistry and kinetics of waste glass leaching and most aspects of modelling of glass were discussed by B. C. Bunker who stated that his approach was objective in terms of his not working with nuclear waste disposal. He talked about complex glass compositions, complex solution chemistry and leached layer effects. He discussed network hydrolysis, interdiffusion or chemically controlled diffusion, chemical diffusion in terms of hydrolysis, water diffusion, and free energy of hydration, solution chemistry models and dissolution rate vs. saturation. He concluded that the greatest uncertainity in leach rate predictions are due to our lack of understanding of the surface layer and that an understanding of network hydrolysis is needed. Papers specifically dealing with leaching mechanisms included those by X. Feng and A. Barkatt, T. A. Abrajano, et al., R. D. Aines, et al., and F. Lanza, et al. Some of the results presented at the meeting were from reports recently reviewed under the NBS/NRC program. There was more emphasis on mechanisms of the leaching process, which investigators are becoming more familiar with, than was evident in previous reports. The mechanism involves diffusion of an ionic acid species (possibly H_3O^+) into the glass and the simultaneous outward diffusion of alkali ions. Network forming species (for example Si, B or oxide ions containing these elements such as $HSiO_4$-tend to be removed at a much slower rate. This process leads to the formation of a gel covering the glass which consists of remaining network forming materials. Most aspects of the modelling of the glass leaching process were summarized. Progress in modelling efforts and the significant problems concerning the chemistry and kinetics of waste glass leaching were outlined. Also of interest were papers by D. F. Bickford and D. J. Pellarin describing results of large scale leach studies.

Session L15 - Radiation Effects

The papers dealt with radiation effects on copper corrosion (W. H. Yunker and R. S. Glass, W. H. Smyrl, et. al.), oxidation of actinides in salt solutions (J. I. Kim, et. al.), effects of radiation on groundwater chemistry and the waste package (W. L. Ebert, et. al. and M. A. Lewis and D. Reed). Smyrl, et al.'s found that kinetics of Cu corrosion in low concentrations of H_2O_2 are identical to kinetics for dissolved oxygen. At higher concentrations of H_2O_2, this similarity disappears and Cu oxidizes to Cu^+ (not Cu^{++}) producing a CuCl film on the surface.

Session L16 - Groundwater Chemistry and Interactions

M. I. Wood, et al. and C. A. Cederberg, et al. focussed on the adsorption process as species migrated into surrounding packing material. Some of the important leachate species such as Pu and Tc have a tendency to undergo reduction and be adsorbed or precipitate from the groundwater. There appears to be good evidence that reduced actinide species have a strong affinity for basalt or bentonite so that a large fraction of will be immobilized, or the rate of migration will be impeded.