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

April 15, 1987

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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

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

Dear Mr. Wick:

Enclosed is the March 1987 monthly progress report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-6). The financial information is reported separately. Also attached is curriculum vitae for Mr. David Anderson of the NBS Metallurgy Division, who will assist in the review of technical documents.

Sincerely,

Charles G. Interrante

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

Enclosures

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Monthly Letter Report for March 1987

Published April 1987

(FIN-A-4171-6)

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

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

TASK 1 -- Review of Waste Package Database and High Level Waste Data Center

A revision of the draft bi-annual report that was submitted to the NRC on February 28, 1987 is being reviewed by Washington Editorial Review Board (WERB), and their review should be completed within the time required to permit a final report to be issued by May 30, 1987.

Status of Reviews

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

1. BNL 32001, "Very Rough Preliminary Estimate of the Colloidal Sodium Induced in Rock Salt by Radioactive Waste Canister Radiation," by P.W. Levy and J.A. Kierstead, September 1984

2. HEDL-7545, "Zircaloy Spent Fuel Cladding Electrochemical Corrosion Experiment at 170 C and 120 psiA H20," April 1986

3. RHO-BW-SA-416P, "Sorption and Desorption Reactions of Radionuclides with a Crushed Basalt-Bentonite Packing Material," April 1986

SRP -- Review is continuing on the following reports.

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

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

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

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

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

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

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

BWIP -- Review is continuing on the following reports.

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

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

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

4. RHO-BW-SA-560P, "Status of Environmentally Assisted Cracking Studies by the Basalt Waste Isolation Project," March 1986

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

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

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

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

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

NNWSI -- Review is continuing on the following reports.

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

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

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

4. HEDL-7546, "C-Ring Stress Corrosion Cracking Scoping Experiment for Zircaloy Spent Fuel Cladding," March 1986

5. UCRL-5317, "Radiation Chemical Effects in Experiments to Study the Reaction of Glass in an Environment of Gamma-Irradiation Groundwater and Tuff," May 1986

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

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

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

Other Reports/Technical Papers -- Review is continuing on the following report judged to have related scientific value sufficient to warrant its review (as reference material).

1. "Aging of Cast Duplex Stainless Steels in LWR Systems", O. K. Chopra and H.M. Chung, Nuclear Engineering Design, 89, 305-318 (1985).

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

1. "Chemical Durability Studies on Glass Compositions Pertaining to Waste Immobilization at West Valley," A. Barkatt, et al., Waste Management 1986

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

3. "Leach Mechanisms of Borosilicate Glass Defense Waste Forms - Effects of Composition," A. Barkatt, et al., no date

4. "Chemical Determination of West Valley Waste Form Products," D. M. Oldman, J. R. Stimmel, and J.H. Marlow, no date

5. "Startup and Initial Experimental Results for the West Valley Vitrification Demonstration Project, Waste Management '86: Waste Isolation in the U.S. Technical Programs and Public Education Volume 2 High-Level Waste; Proceedings of the Symposium on Waste Management, Tucson, AZ, March 2-6, 1986

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

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

8. PNL-4382, "Workshop on the Leaching Mechanisms of Nuclear Waste Forms," Summary Report, May 19-20, 1982

9. "Effects of Compositon on the Leach Behavior of West Valley HLW Glasses," X. Feng, et al., no date

TASK 2 -- Identification of Additional Data Required and Identification of Tests to Generate the Data

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

DAVID B. ANDERSON

Biographical sketch

Mr. Anderson received a B.S. degree in biological sciences from Yale University and has completed graduate studies in Materials Science at North Carolina State University. Mr. Anderson joined the Bureau of Standards in 1986 after a thirty year career with International Nickel Limited. He is currently a physical scientist working in the corrosion data program. His career has been corrosion orientated throughout, with responsible positions in corrosion research, contract research marketing, alloy development and market development.

Mr. Anderson's research accomplishments cover a broad range with particular emphasis on evaluation techniques for corrosion-erosion and crevice corrosion; alloy development, principally in copper-base, nickel-base and stainless steel alloys and corrosion data interpretation and utilization. He has authored twenty-two technical publications, lectured extensively and holds four patents.

Mr. Anderson is an active member of the National Association of Corrosion Engineers and ASTM and has held numerous committee assignments.

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NBS REVIEW OF TECHNICAL REPORTS ON THE HIGH LEVEL-WASTE PACKAGE FOR NUCLEAR WASTE STORAGE

DATA SOURCE

(a) Organization Producing Data:

Brookhaven National Laboratory, Upton, New York 11973

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

Levy, P. W. and Kierstad, J. A., "Very Rough Preliminary Estimate of the Colloidal Sodium Induced in Rock Salt by Radioactive Waste Canister Radiation"., Mat. Res. Soc. Symp. Proc., Vol 26 (1984) Elsevier Science Publ. Co., Inc. Pg 727-734

DATE REVIEWED: 12/31/86

TYPE OF DATA

Model calculation of interaction of gamma radiation with rock salt leading to formation of colloidal sodium.

MATERIALS/COMPONENTS

Rock salt, gamma radiation

TEST CONDITIONS

Not applicable

METHODS OF DATA COLLECTION/ANALYSIS

Not Applicable.

AMOUNT OF DATA

Figure 1. Typical Canister configuration, cocoon approximation and dimensions used for computations; Figure 2. Radiation Induced Colloid Growth Curves used for computations, absorption coefficient (cm^{-1}) vs. total dose,(Mrad); Figure 3. Salt converted to Na colloid (percent) vs. time (years) after emplacement for each layer of surrounding canister; Figure 4. Total colloid mass (kg) in each layer vs. time (yrs) after emplacement; Figure 5. Percent salt converted to colloid vs. distance (cm) into salt for 50, 100, 150 and 300 years after placement; Figure 6. Total colloid mass (kg) in the first 25 cm surrounding the canister vs. time after emplacement; Table 1. Properties of the four waste canisters used to estimate the radiation induced Na (sodium) colloid expected in rock salt adjacent to emplaced canisters (includes dimensions of canisters, air gap, thickness of carbon steel and other overpacks, active length, overpack length, distance from waste centerline to salt, initial heat load, and initial radioactivity); Table 2. Total mass in kg, of radiation induced Na

metal produced in natural rock salt from the WIPP site by gamma-ray radiation from the four canisters described in table 1 at various times after emplacement.

UNCERTAINTIES IN DATA

The authors state that the estimates are based on minimal data and require a large number of assumptions and extrapolations.

DEFICIENCIES/LIMITATIONS IN DATABASE

The authors state that "These estimates are based on minimal data and require a large number of assumptions and extrapolations. For better radiation damage estimates, particularly of the Na metal colloid formed in repository rock salt, improved data are essential."

KEY WORDS

brine, colloidal sodium in SRP, gamma radiation field, laboratory, model for radiation damage of rock salt, radiation damage of rock salt, salt, theory, gamma radiation field

GENERAL COMMENTS

This report discusses model calculations involving the formation of a sodium colloid formed in the vicinity of waste canisters in a SRP Nuclear Waste Repository. Although the authors consider the calculations as crude, the basic contention is that the formation of massive colloids of sodium in the vicinity of the canisters will eventually react with brine to form sodium hydroxide which will drastically change the pH of the local environment. The most important question involves the fate of chlorine formed as a result of radiation of the rock salt. The reviewer does not consider it likely that large amounts of a reactive gas like chlorine will escape from the local environment. More likely, reaction with brine will occur with the formation of HOCl or similar chloric acids which should react with and neutralize the sodium. But more definitive data on the fate of chlorine needs to be obtained.

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

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

This report is related to issue 2.1.3.1 involving how the chemical characteristics of the brine reaching the waste package container will be affected by radiolysis.

(b) New Licensing Issues:

(c) General Comments:

Draft Copy

NBS REVIEW OF TECHNICAL REPORTS ON THE HIGH LEVEL WASTE PACKAGE FOR NUCLEAR WASTE STORAGE

DATA SOURCE

(a) Organization Producing Data:

Westinghouse Hanford Laboratory, P. O. Box 1970, Richland, Washington for the Nevada Nuclear Waste Storage Investigations Project (NNWSI)

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

H. D. Smith, "Zircaloy Spent Fuel Cladding Electrochemical Experiment at 170°C and 120 psia H_2O," HEDL-7545, April, 1986

DATE REVIEWED: January 19, 1987

TYPE OF DATA

There is no data. This is a research plan.

MATERIALS/COMPONENTS

Zircaloy spent fuel cladding from the H. B. Robinson fuel assembly B-05 (removed from service, May 6, 1974 after 799 full power days, av. burnup of 28.0 MWd/kgM and peak burnup, 31.4 MWd/kgM), identified as MCC Approved Testing Material ATM-101; cladding with surface oxide and cladding with the oxide removed to expose the bare metal.

TEST CONDITIONS

Two environments, considered to be critical, are used. The first occurs after 90 to 100 years with an unbreached container but with breached fuel containing water. This test is designed to expose cladding in 120 psi, 170°C deionized water in an autoclave. The second environment occurs after 1000 years and the canister is breached. Cladding is exposed to 90°C tuff-equilibrated J-13 water. Six experiments are planned. The first two tests are scoping tests and results will be used to refine future tests. Bundles of spent fuel cladding mimicing the configuration in repository storage will be placed in Hastelloy-276 autoclave for periods of 2, 6 and 12 months. The fuel with Zircaloy-4 plugged ends will be bundled in a hexagonal array, wrapped in 304 stainless steel sheet, clamped with stainless steel, and will sit on alumina plates. Alumina spacers will be used to electrically isolate the cladding from the autoclave. Some C-clamp specimens will be loaded to test for stress corrosion cracking.

METHODS OF DATA COLLECTION/ANALYSIS

A thermocouple will be used to monitor the autoclave temperature, the pressure will be monitored, and the amount of water at the beginning and end of the test and any added water will be measured. The water level will be about one-half the distance to the top of the autoclave. Records will be kept in a dedicated HEDL notebook. At intervals in the test, 5 ml of water will be drawn to analyze for pH, Zr, ^{14}C, and ^{137}Cs. Evaluation after testing will use photography to document exposed specimens. Selected specimens will be prepared metallographically and observed with light

microscopy and with the scanning electron microscopy. Auger analysis in conjunction with ion milling will be used to analyze thin film growth.

AMOUNT OF DATA

This report gives no data from experimental tests. There are five figures. Four of these figures are "Section Diagrams" and "Sample Identifiers Correlated to the Cs-137 Gamma Scan of H.B. Robinson Fuel Rod Segments (J12C, N4C, C5T, and C5B)"; these show activity counts per 300 sec. vs. distance from the bottom in cm. The fifth figure is a "Schematic Cross Section of the Autoclave with the Spent Fuel Cladding Bundle in Place." There is one table which gives the "Distribution of Spent Fuel Specimens (from Figures 1, 2, 3, and 4) Between the Three Experimental Bundles."

UNCERTAINTIES IN DATA

Corrosion rates at 180°C on the order of tens of angstroms may be detectable by profiling thin films with Auger analysis/ion milling techniques. Photographic and microscopic evaluation rely on comparative observations to identify and measure corrosion.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

KEYWORDS

Planned work, corrosion, surface film, pitting, visual examination radioactivity level simulated field, Yucca Mountain, J-13 water, deionized water, high temperature high pressure, cladding, spent fuel, Zircaloy, Zircaloy-4, ^{14}C, ^{137}Cs.

COMMENTS

This seems to be a good experiment for scoping corrosion behavior under the conditions given. The sampling, apparatus, testing procedures, and specimen analyses appear to be planned well. The data from these tests will show whether corrosion does or does not occur within the testing time. There are no electrical measurements, and electrochemical measurements are the type of data needed to describe corrosion reactions and to predict long-term behavior. Tests with electrochemical connections should be considered for future planned tests. In the meantime, the data, as obtained from the tests as planned, should be useful.

RELATED HLW REPORTS

Five reports are cited in the references; HEDL-7455, HEDL-TC-2562, PNL-5109, UCID-20172 (reviewed at NBS) and HEDL-7455 Rev. 1.

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 mechanism for spent fuel cladding

(b) New Licensing Issues: None

PDR

Dr. John T. Greeves, Chief
Engineering Branch, Division of Waste Management
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Washington, DC 20555