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August 15, 1986

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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 Reports for July 1986 (FIN-A-4171-6)

Dear Mr. Wick:

Enclosed is the 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 are curriculum vitae for Dr. A.C. Fraker who has taken a leading role in this work.

Sincerely,

Charles G. Interrante

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

1 Enclosure

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Monthly Letter Report for July, 1986

Published July 1986

(FIN-A-4171-6)

Performing Organization: National Bureau of Standards
Gaithersburg, MD

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

TASK 1 -- Review of Waste Package Data Base

WERB Reviews -- Revised procedures for obtaining NBS publication approval through the Washington Editorial Review Board (WERB) have been approved and reviews are being processed through WERB. Each NBS review will be approved separately and each will have an NBS author, who is identified internally, i.e. at NBS only. Thus, the WERB approval process will have been initiated for each of the reviews included in the Draft of the Semi-Annual Report, which will be sent to you by August 31.

Appended to this report are copies of the Draft Reviews available at this time. Comments by NRC and its contractors are solicited. These reviews are all expected to be incorporated into our Semi-Annual Report, which will be published by November 30, 1986.

BWIP -- Reviews were initiated this month on the following reports:

1. HEDL-SA-3271FP, "Low Temperature Spent Fuel Oxidation Under Tuff Repository Conditions."

2. HEDL 7452, "Evaluation of the Potential for Spent Fuel Oxidation Under Tuff Repository Conditions."

3. "Test Plan for Series 2 - Thermogravimetric Analyses of Spent Fuel Oxidation."

4. SD-BWI-TS-008, "Slow-Strain-Rate Testing of 9%Cr, 1%Mo Wrought Steel and ASTM A27 Cast Steel in Hanford Grande Rande Groundwater," S.G. Pitman (PNL), October 1984.

5. SD-BWI-TS-012, "Short-Term Stress-Corrosion - Cracking Tests for A36 and A387-9 Steels in Simulated Harford Groundwater," L.A. James (Westinghouse), June 1985.

BWIP -- The following reports are under consideration and will be reviewed as soon as a suitable reviewer has been identified.

1. SD-BWI-DP-060
   "Interim Data Document for the Advanced Conceptual Design of High Level Waste Packages for a Repository in Basalt"

2. B036177
   "Analysis of the Effects of Radiation on the Chemical Environment of a Waste Package in a Nuclear Waste Repository in Basalt"

3. RHO-BWI-ST-15
   "Corrosion Test of Canister and Overpack Materials in Simulated Basalt Groundwater"

4. 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-Bentonite Packing"

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

6. B032831
   "Analysis of the Effect of Radiation on the Chemical Environment of a Waste Package in a Nuclear Waste Repository in Basalt"

7. SD-BWI-TI-235 (B032012)
   "Corrosion Evaluation of Candidate Iron Based Nuclear Waste Package Alloys in Grande Ronde Basalt Groundwater"

8. SD-BWI-CR-005
   "Conceptual Design Requirement for Spent Fuel, HLW and TRU Waste Packages"

NNWSI -- The following report is under consideration and will be reviewed as soon as a suitable reviewer has been identified.

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

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

NBS lead workers are continuing their studies concerning the types of additional data and verification tests needed to demonstrate that the DOE waste package designs will meet the performance objectives of 10 CFR 60. The preliminary plans for laboratory tests to be conducted at the NBS under Task 3 of this contract were discussed briefly and submitted informally to NRC staff, as written preliminary proposals during a Program Review held at the NBS on June 30, 1986. It is expected that after further joint NRC/NBS consideration, the proposals will be modified as needed to permit this work to be initiated this Summer.

TASK 4 -- General Technical Assistance

During the period from July 1 through July 11, NBS responded to your recent verbal request (and the April 1st letter request) for general technical assistance related to the Department of Energy's Final Environmental Assessment for the following three sites: Hanford (BWIP); Yucca Mountain (NNWSI); Deaf Smith (SRP). The results of this work were transmitted informally to you on July 14, 1986. The letter of transmital that followed on July 18 contained an additional comment for the Hanford Reservation Site. In addition, in a memorandum, dated July 23 from R. Shull to K. Chang, additional comments were given on the Hanford Reservation FEA.

The Final Draft of the Aerospace Corporation's Demonstration of Methodology for Waste Package Performance Assessment (FIN A-4165) was presented in a meeting in Room 110 of the Willste Building on July 22, 1986. Representing the NBS at this meeting were A. Fraker, C. Interrante, and J. McFadden. Our review of this work was initiated and it will be transmitted to you on or about August 15, 1986.

On July 31, 1986, C. Interrante and M. Linzer of the NBS attended an NRC/DOE Meeting on Waste Acceptance at the Forrestal Building, Room BE069. The Waste Acceptance Process (WAP) and the Waste Acceptance Preliminary Specifications (WAPS) were discussed, and draft (for concurrence) copies of the applicable WAPS documents were made available both for the West Valley Demonstration Project High Level Waste Form (OGR/B-9) and for the Defense Waste Processing Facility High Level Waste Form (OGR/B-8).

Draft Copy

WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory, Livermore, CA 94550

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

Weiss, H., Van Konynenburg, R. A., and McCright, R. D.,"Metallurgical Analysis of a 304L Stainless Steel Canister from the Spent Fuel Test--Climax", UCID-20436, April 1985. Available from NTIS.

DATE REVIEWED: 10-28-85/Revised 12-2-85/1-13-86

TYPE OF DATA

Experimental study of the possible effects of a 3-year exposure of a 304L canister in Climax stock quartz mangonite and the implications for waste disposal in a tuff environment. The canister was examined by metallurgical examination and chemical analysis, and the water taken from the liner rock annulus was also analyzed for smilarities with anticipated tuff water chemistry.

MATERIALS/COMPONENTS

304L canister with 308L weld filler wire; plain C steel liner

TEST CONDITIONS

Three-year exposure; underground granite; canister containing spent fuel; water containing a significant concentration of dissolved species (including Cl) in contact with base of the welded 304L canister; maximum temperature ~ 140 °C; ambient pressure; total γ dose: 3.2×10^8 rads

METHODS OF DATA COLLECTION/ANALYSIS

Metallographic observation after exposure; thermocouple measurements during storage; chemical analysis of canister materials and well water before and after exposure

AMOUNT OF DATA

One table comparing water analysis of Climax facility with J-13 well water; one table listing results of chemical analysis of base metal, weld metal, and weld wire; one graph of temperature (20 - 120 °C) vs. time (2.4 - 3.6 yrs) for the canister and liner

UNCERTAINTIES IN DATA

Not addressed.

DEFICIENCIES/LIMITATIONS IN DATABASE

The authors discuss the validity of using the data from this environment for predicting the behavior in a tuff repository and conclude that the environment studied is more aggressive than the tuff environment indicating that the performance of 304L in tuff should be adequate for repository purposes.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

The data are considered supporting for issues 2.2 and 2.2.1 in the ISTP for NNWSI.

(b) New Licensing Issues

(c) General Comments

KEY WORDS

GENERAL COMMENTS

The authors suggest in their discussion that more severe chloride cracking occurs at the lower temperatures encountered in the spent-fuel test than at the higher temperatures expected for the repository. This is questionable based on previous work done in these alloys over a range of temperatures. For example, Kowaka and Kudo (Trans. JIM, 16 (1925) 385) show time to failure (t_f) curves for 304 as a function of temperature where there is a minimum in t_f at approximately 140 °C. Thus, it would be useful to perform similar experiments at slightly higher average temperatures (i.e., 130 - 145 °C) where SCC is most pronounced.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory, Livermore, CA 94550

(b) Authors(s), Reference, Reference Availability

Juhas, M. C., McCright, and Garrison, R. E., "Behavior of Stressed and Unstressed 304L Specimens in Tuff Repository Environmental Conditions", UCRL-91804, Corrosion 85 NACE Annual Meeting, Boston, MA, March 25-29, 1985, in press.

DATE REVIEWED: 10-28-85/Revised 12-2-85/1-13-86

TYPE OF DATA

Experimental study of the susceptibility of 304 and 304L steel to failure by general, localized, or stress corrosion

MATERIALS/COMPONENTS

AISI Types 304 and 304L stainless steels; J-13 well water

TEST CONDITIONS

(1) Irradiated corrosion tests: 304L corrosion coupons for one year at room temperature and ~ 1 x 10^5 rads/hr in J-13 water containing approximately 5 ppm oxygen gas in contact with crushed Topopah Spring tuff. Both solution-annealed and sensitized conditions were tested. Some of the specimens were in water only, while others were placed in water and tuff to determine if crevice corrosion would occur in the latter.

(2) U-bend irradiation-corrosion tests: 304 and 304L (annealed and sensitized conditions) in autoclaves at 50 and 90 °C using a ^{60}Co irradiation facility. Each autoclave contained 3 zones: water + rock (bottom), rock + vapor (middle), and vapor only (top).

(3) Slow strain rate tests: 304 and 304L (mill annealed and sensitized conditions) in autoclave containing air-sparged flowing J-13 well water with crushed tuff at the bottom.

(4) Bent beam stress corrosion tests: 304 and 304L, in 4 point loaded (ASTM G-39) configuration stressed to 90% of room temperature yield strength, were tested in tuff conditioned J-13 water at 100 °C and in saturated vapor above the water. Specimen conditions: welded, some given post-weld anneal, all sensitized (700 °C, 8 hrs) and then cold worked 20%. Also, other specimens were cold worked and welded but not heat treated for sensitization.

METHODS OF DATA COLLECTION/ANALYSIS

Irradiation corrosion tests; U-bend irradiation corrosion tests; slow strain rate tests; visual and metallographic analyses; weight loss measurements

AMOUNT OF DATA

(1) Irradiation corrosion tests: table listing corrosion rates of 24 coupons in each of two containers, one irradiated, the other nonirradiated

(2) U-bend irradiation corrosion tests: table listing No. specimens cracked/No. tested of 24 specimens tested (6 in water + rock, 6 in rock + vapor, and 6 in vapor only) for each condition (solution annealed and sensitized) and each temperature (50 °C and 90 °C)

(3) Slow strain rate tests: one table listing compositions and mechanical properties of the 304 and 304L steels; two tables listing the results of the following tests:

six 304 specimens in well annealed and six in sensitized conditions in air and J-13 water at two different strain rates (10^{4}/s and $2x10^{7}$/s)

four 304L (solution annealed) specimens tested in J-13 water at two strain rates

five 304L (sensitized) specimens: one in air and four in J-13 water at two strain rates

(4) Bent beam tests: table summarizing the results of tests on 94 specimens total of 304 and 304L in various conditions.

UNCERTAINTIES IN DATA

DEFICIENCIES/LIMITATIONS IN DATABASE

The authors address possible limitations of their experimental results to potential repository conditions with an emphasis on low temperature sensitization and conclude that, even though no comprehensive studies have been performed to date, there is a possibility of low temperature sensitization during nuclear waste storage.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

The data in this report are considered key data relative to issues 2.2 and 2.2.1 in the ISTP for NNWSI.

(b) New Licensing Issues

(C) General Comments

KEY WORDS

GENERAL COMMENTS

The authors' statements concerning possible low temperature sensitization seem viable in view of some recent analyses by Fox, et al. (UCRL-15619). Further work is needed to elucidate these possibilities.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory

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

Fox, Michael J. and McCright, R. Daniel, "An Overview of Low Temperature Sensitization", UCRL-15619. Report by consultant (M. J. Fox) to LLNL. Availability (?).

DATE REVIEWED: 10-30-85/Revised 12-2-85/1-10-86

TYPE OF DATA

Literature review; data analysis

Purpose of work: To determine whether stainless steel waste canisters, either during the glass pouring operation of the waste material or during the burial period, are likely to develop a sensitized microstructure, thereby allowing selective corrosive attack.

MATERIALS/COMPONENTS

304, 304L, 304LN, 316, 316L, 316LN, 316ELC, 316NG, 347, CF-3, XM-19, stainless steels

TEST CONDITIONS

No tests were performed by this study. However, the investigations reviewed by this paper considered both heavily cold worked and annealed materials located in a variety of test environments (including oxalic acid, nitric acid, high purity water at 250 °C, air, and a boiling solution of sulphuric acid and copper sulphate).

METHODS OF DATA COLLECTION/ANALYSIS

No tests were performed by this study. However, the investigations reviewed by this paper used electron microscopy, Auger spectroscopy, ASTM tests No. A262-68 (Practices A, C, and E), slow strain rate test, wedge open loaded/compact tension test, crevice bent beam test, and time-temperature data from several thermometers attached to a canister into which molten glass had been poured.

AMOUNT OF DATA

This review presents four graphs of data:

(1) time ($0 < t \leq 24$ hrs) vs. temperature ($0 < T < 600$ °C) data for a waste canister during a glass pouring operation

(2) time ($0 \leq t \leq 300$ yrs) vs. temperature ($0 < T < 300$ °C) data calculated for four different types of radioactive waste packages in a tuff repository

(3) time ($0 < t < 1000$ days) vs. reciprocal temperature (250 °C $< T <$ 500 °C) line separating the t-T regimes for cold worked 304L steel sensitization

(4) time ($0 < t < 60{,}000$ yrs) vs. reciprocal temperature (100 °C $< T <$ 500 °C) extrapolation of the sensitization demarcation line of the 3rd figure to a much longer time period

UNCERTAINTIES IN DATA

With regard to commercially available materials, the sensitization line depicted in figures 3 and 4 is stated to be shifted toward higher temperatures. Therefore, data lying to the high temperature side of the line indicates a definite sensitization probability; however, data on the other side of the line does not indicate conversely the absence of any sensitization possibility.

DEFICIENCIES/LIMITATIONS IN DATABASE

There is no data on the time-temperature behavior of canisters as a function of wall thickness. There were also only a limited number of thermocouple locations on the canister monitored during the glass pouring operation; therefore, localized hot spots, such as at the very bottom of the canister, may have been missed.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

This paper is related to Issues Nos. 2.2.1 and 2.2.2. If low temperature sensitization of the container material occurs during its burial period, the chemical resistance of the material will change.

(b) New Licensing Issues

None.

(c) General Comments

KEY WORDS

Low temperature sensitization; austenitic stainless steel; cold work; molten glass pouring; canister; stress corrosion cracking; carbon depletion; grain boundary

GENERAL COMMENTS

This is a good survey of the literature on the low temperature sensitization of austenitic stainless steels. Especially useful information included in this paper are the previously unpublished time-temperature data for a waste canister during a glass pouring operation. For a 304 stainless steel canister, it is apparent that sensitization can occur both during this pouring operation as well as during the first 20 years of burial. For other stainless steels more data are required.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory (for U.S. Department of Energy), Livermore, CA 94550

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

Smith, H. D. and Oversby, V. M.,"Spent Fuel Cladding Corrosion under Tuff Repository Conditions--Initial Observations, UCID-20499", June 1985. Intended for internal or limited external distribution.

DATE REVIEWED: 10-10-85/Revised 1-16-86/1-27-86/3-11-86

TYPE OF DATA

Corrosion rate data; pH and conductivity of J-13 water
Scoping experiment to help plan future direction of corrosion tests

MATERIALS/COMPONENTS

Radioactive Zircaloy-4 (~ 700 mR/H at 2 inches)

TEST CONDITIONS

Crushed tuff in J-13 water at 90 °C

METHODS OF DATA COLLECTION/ANALYSIS

Visual examination at various magnifications (1 to 1000) (from metallurgical samples

AMOUNT OF DATA

Three observations: corrosion < 1 - 2 μm/yr; pH J-13 = 8.65 ± 0.3; conductivity = 750 ± 200 mho

UNCERTAINTIES IN DATA

Technique sensitive to "a few" microns

DEFICIENCIES/LIMITATIONS IN DATABASE

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

Related to issues 2.2.4 and 2.2.4.1 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

KEY WORDS

Corrosion, Zircaloy-4, J-13 water, tuff, spent fuel cladding

GENERAL COMMENTS

Little can be said about this work at this point. The authors describe a short term (one year) corrosion experiment to test the feasibility of their approach. Presently, their optical techniques detect 1 - 2 μm/yr corrosion. However, this technique has not been sufficiently sensitive to detect corrosion on the Zircaloy specimens exposed for up to one yr. They plan to improve sensitivity of these observations by starting with polished surfaces on their corrosion specimens.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory (for U.S. Department of Energy), Livermore, CA 94550

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

McCright, R. Daniel and Weiss, Haskell,"Corrosion Behavior of Carbon Steels Under Tuff Repository Environmental Conditions", UCRL-90875, October 1984, Proceedings of Materials Research Society Annual Meeting, Boston, MA, November 1984. (Preprint; intended for publication in a journal or proceeding)

DATE REVIEWED: 10-10-85/Revised 1-16-86/1-27-86/3-11-86

TYPE OF DATA

General corrosion rates, pitting corrosion rates, crevice corrosion rates, corrosion potentials, corrosion currents, number of SCC failures/time

MATERIALS/COMPONENTS

1020 steel; grey cast iron; nodular cast iron; 2-1/4 Cr-1 Mo alloy steel; 9 Cr-1 Mo alloy steel; A-36 steel

TEST CONDITIONS

J-13 water at 50 °C, 70 °C, 80 °C, 90 °C, 100 °C; J-13 steam irradiated, aerated J-13 water at 90 °C

METHODS OF DATA COLLECTION/ANALYSIS

Weight loss measurements, stress-corrosion bent-beam tests, linear polarization resistance

AMOUNT OF DATA

Limited data, e.g., one value of corrosion rate at each temperature for each alloy

UNCERTAINTIES IN DATA

Not addressed

DEFICIENCIES/LIMITATIONS IN DATABASE

Only the average values are listed, making statistical evaluation impossible.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

Related to issues 2.2.4 and 2.2.4.1 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

KEY WORDS

General corrosion, nuclear waste, canister, tuff, Nevada site, stress corrosion, pitting corrosion

GENERAL COMMENTS

This paper covers a broad range of corrosion tests, and includes a preliminary process designed to guide their future work. As such, it is useful, and the authors do recognize that more testing must be done to substantiate these preliminary results and projections.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory, Livermore, CA 94550

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

Glass, R. S., Overturf, G. E., Garrison, G. E., and McCright, R. D., "Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff", June 1984. (Informal report intended primarily for internal or limited external distribution)

DATE REVIEWED: 10-6-85/Revised 1-16-86/1-27-86/3-11-86

TYPE OF DATA

Electrochemical data (e.g., measurements of corrosion potential, corrosion current, pitting potential, and protection potential) and weight-loss.

MATERIALS/COMPONENTS

AISI 304L, 316L, 321, 347 stainless steels, and high-nickel 825, all in mill annealed condition. Water for experiments from the J-13 well at Yucca Mountain, NV, with analytical grade NaCl added in some cases

TEST CONDITIONS

Laboratory tests in 1-liter flasks; specimens immersed in static saturated water or water deaerated with high purity Argon; range of water and solution temperature: 50 to 100 °C; range of solution chloride-concentration: 0 to 30,000 ppm; 1X and 10X concentration J-13 water; crevice-corrosion disk specimens, 1 sq cm in area masked off with neoprene O-ring, or 5 sq cm cylinder of material

METHODS OF DATA COLLECTION/ANALYSIS

Potentiodynamic anodic polarization with Tafel slope extrapolation and linear polarization using three electrodes; scan rate 1 mv/s; weight loss data obtained after specimens had been immersed for 3548 or 5000 h

AMOUNT OF DATA

Limited amount of data. Corrosion rate vs. temperature for 5 temperatures for each alloy; corrosion potential, pitting potential, and protection potential vs. temperature for each alloy; and some data for corrosion and pitting potentials vs. chloride concentration (0 to 30,000 ppm)

UNCERTAINTIES IN DATA

Not addressed.

DEFICIENCIES/LIMITATIONS IN DATABASE

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

Related to issues 2.2.4 and 2.2.4.1 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

Applicable for preliminary screening of alloys.

KEY WORDS

Pitting corrosion, general corrosion, austenitic stainless steels, tuff, nuclear waste, polarization measurements, electrochemical data, J-13 spring water

GENERAL COMMENTS

On the basis of their electrochemical studies (200-day exposure) and their "sandwiched" flat metal specimen study (one year), the authors conclude that all candidate materials will meet the 300 - 1000 year containment objective. This conclusion is based on the assumption that the corrosion rates obtained over a 200-day period is applicable to the 300 to 1000 year period and that the corrosion will be uniform. However, their own data indicate that crevice corrosion does occur under some conditions of exposure. This localized attack should be taken into consideration, and explored much more fully before the analysis can be regarded as complete.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH

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

G. Jansen,"Expected Waste Package Performance for Nuclear Waste Repositories in Three Salt Formations", BMI/ONWI- , August Draft, 1984, MS-623SS.

DATE REVIEWED: 1-15-86/Revised 1-21-86/1-27-86/3-11-86

TYPE OF DATA

Modeling of corrosion performance and leaching rates of radionuclides. Calculated results of failure as a function of time, considering general corrosion and localized corrosion.

MATERIALS/COMPONENTS

12 to 15 cm thick overpack of AISI 1025 low-carbon steel

TEST CONDITIONS

Cast and wrought condition.
Maximum temperature range at seven sites 130-296 °C.
Low-Mg and high-Mg brines at low rate of flow (max .001 - .06 m^3/yr).

METHODS OF DATA COLLECTION/ANALYSIS

Data from modeling calculations. Temperature at overpack surface over 10K yrs. Flow rate of brine over 10K yrs. Radiation vs distance with time. Corrosion rate vs temperature. Wall thickness vs time. Effect of nonuniform (pitting) corrosion. Compressive loading and a uniform distribution of brine are assumed.

AMOUNT OF DATA

Thirteen figure numbers with subcategories (2a, 2b, ...2g)

Fig. 1 - Configuration of waste package
Fig. 2 - Transient temperature at the salt-overpack interface, 7 sites
y-axis, temp, 0 - 300 °C
x-axis, time, 0 - 1000 yrs (linear), .1 - 10,000 y (log)
Fig. 3 - Brine migration inflow rates per canister, 7 sites
y-axis, brine rate, 10^{-6} - 10 m^3/yr (log)
x-axis, time since burial, 1 - 10,000 yrs (log)

Fig. 4 - Transient radiation field profiles at waste package, one low-Mg brine, one high-Mg brine site
y-axis, dose rate, 10^{-5} - 10^5 rads/hr
x-axis, distance from centerline, 0 - 100 cm
Fig. 5 - Radiation fields at the waste package interfaces. 2 sites at low- level waste and high-level waste
y-axis, dose rate/hr, 0 - 25,000 rads/hr, 0 - 8,000 rads/hr
x-axis, distance from centerline, cm, 23 - 35 cm

y-axis, dose rate, 0 - 5 rads/hr
x-axis, distance from centerline, 30 - 90 cm

Fig. 6 - Effect of radiation field on corrosion rates of cast steel in brine, 7 sites
y-axis, penetration, 10^{-3} - 10^3 mils/yr (log)
x-axis, overpack surface temp, 0 - 400 °C
Fig. 7 - Stress boundary conditions at the waste-package midplane, 7 sites
y-axis, normal stress, 0 - 40 mpa
x-axis, time since burial, 0 - 18 yrs
Fig. 8 - Comparison of wall thickness of overpack vs time, 7 sites
y-axis, wall thickness, 0 - 20 cm
x-axis, time since burial, 2 - 20 yrs
Figs. 9 and 10 - Effect of brine rate (availability) on corrosion of the overpack
4 sites high-Mg brine at low level and high-level waste
5 sites low-Mg brine at low level and high-level waste
y-axis, wall thickness, 0 - 20 cm
x-axis, time since burial, 0 - 1,000 yrs
Fig. 11- Radiation fields at the waste package surface during corrosion, 3 sites each at low- and high-level waste
y-axis, gamma dose rate, 10^{-4} - 10^6 rads/hr
x-axis, time since burial, 0 - 10,000 yrs
Fig. 12- Net accumulated brine during corrosion of the waste package, 7 sites, 3 low-Mg brine, 4 high-Mg brine
y-axis, brine volume, 0 - 0.4 m^3 log Mg; 0 - 2 m^3, high Mg
x-axis, time since burial, 0 - 100 yrs

Fig. 13- Net accumulated brine during corrosion of the waste package, 5 sites, 2 low-Mg brines, 3 high-Mg brines
y-axis, brine volume, 0 - 0.4 m^3, low Mg; 0 - 2 m^3, high Mg
x-axis, time since burial, 0 - 10,000 yrs

Twelve tables with subsets

Table 1 - Original dimensions of overpack (for low-level waste and high-level waste)
Table 2 - Compositions of simulated salt brines used in corrosion tests (low-Mg content and high-Mg content)
Table 3 - Effect of brine availability on corrosion of waste package--uniform corrosion 0.2 - 5 cm loss.
Failure time 220 yrs to > 10,000 yrs

Table 4 - Effect of brine and corrosion distribution on failure of waste packages, 7 sites
Assume fractional coverage by brine (0.0001 → 1)
Assume pit depth ratios of 1 to 15
Failures occur from 37 to > 10,000 yrs where nonuniform conditions are considered, or where unlimited brine volume is available.
Table 5 - CHLW element inventories and other source terms per MTIHM.
Table 6 - CHLW radionuclide inventories per MTIHM.
Table 7 - SFPWR element inventories and other source terms per MTIHM.
Table 8 - SFPWR radionuclide inventories per MTIHM.

Tables 9 and 10 - CHLW package and SFPWR package -- Comparison of solubility-limited and brine-volume limited release at waste-package boundary, with EPA discharge limits to accessible environment, for 7 sites.
The ratio (of radioactivity in brine to EPA limit) is less than one and acceptable.

Tables 11 and 12 - CHLW package and SFPWR package -- comparison of release rates required to saturate incoming brine at 330 yrs at waste-package boundary with NRC release rate limits from the engineered barrier, as given in CFR 60. This comparison is made for each of the 7 sites.
Table shows that radionuclide release rates are less than the NRC limits except for the following isotopes: C-14, Sr-90, I-129, CS-135, CS-137, Ra-226, Cm-244.

UNCERTAINTIES IN DATA

Uncertainties of model output are addressed by considering various conditions and rates of moisture flow through the repository, including what they regard as outer-boundary conditions. Similarly, various degrees of moisture coverage ranging from 50 to 100% of the waste package surface are considered to be and are regarded as outer-boundary conditions.

DEFICIENCIES/LIMITATIONS IN DATABASE

Author considers conclusions for low-moisture domal salt sites to be very good (robust). Conclusions for bedded salt sites (higher moisture) are less conclusive because the brine distributions are highly variable.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

Key data relative to issues 1.1.1.1, 2.2.1, 2.2.4.1, 2.3.2.3 in the ISTP for the Salt Repository Project (SRP)

(b) New Licensing Issues

(c) General Comments

KEY WORDS

Low-carbon steel, underground corrosion, waste-package underground corrosion, radionuclide leaching, modeling of corrosion performance.

GENERAL COMMENTS

The author concludes that the two container designs that have been proposed for low-level nuclear waste (SFPWR) and high-level nuclear waste (CHLW) will exceed an expected lifetime of 10,000 yrs. These results are based on an assumption of uniform brine coverage and uniform corrosion of the containers. However, his own data shows that, if either of these two assumptions are not valid, failure of the overpack occurs within 88 y; for example, failure would occur within 88 y using a pit ratio of 2.0 (Table 4B) or within 227 y using a 54% surface coverage of the overpack by brine (Table 4E). His modeling calculations do not consider the failure time if both pitting and nonuniform coverage occur at the same time. No doubt, this would reduce the failure time even more.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory, Livermore, CA 94550

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

V. M. Oversby and R. D. McCright,"Laboratory Experiments Designed to Provide Limits on the Radionuclide Source Term for the NNWSI Project", Proceedings of Workshop on the Source Term for Radionuclide Migration from HLW or Spent Nuclear Fuel, Albuquerque, NM, November 13-15, 1984, UCRL-91257 (work performed by LLNL under auspices of U.S. Department of Energy contract W-7405-ENG-48). Preprint, available.

DATE REVIEWED: 1-21-86/Revised 1-27-86/3-11-86

TYPE OF DATA

A review of the authors' data and literature data that includes some analysis of these data. The authors assume general corrosion of stainless steels (SS), and stress corrosion cracking (SCC) of Zircaloy.

MATERIALS/COMPONENTS

Materials under conideration are 304L, 316L, 317L, 321, 347 stainless steels and high-nickel stainless alloy 825.

TEST CONDITIONS

(1) Cold-worked 304L SS. Solution-annealed 316L SS. Condition of other alloys not indicated.
(2) J-13 water at 50 - 100 °C (3548 - 7500 h exposures). 100 °C saturated steam. 150 °C unsaturated steam.

METHODS OF DATA COLLECTION/ANALYSIS

Generally not identified in this review, but results suggest mostly micrograph studies for determination.

AMOUNT OF DATA

One table--water chemistry vs temperature (25 - 150 °C)

Three figures:

(1) Temperature (100 - 310 °C) vs time (0 - 1000 Y) for waste package
(2) Time (0.1 - 1000 Y) vs temperature (100 - 400 °C) long term sensitization of stainless steel
(3) UO_2 concentration, µg/ml (10^{-4} - 10^{2}) vs time (0 - 70 d) in deionized and J-13 water

Only six corrosion rate values were given (in the text) for various conditions of exposure of the stainless steels.

UNCERTAINTIES IN DATA

Standard deviations given for corrosion rates of SS

DEFICIENCIES/LIMITATIONS IN DATABASE

None specifically mentioned.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

Related to issues 1.1.2.3, 2.1.3.1, 2.2.4, 2.2.4.1, 2.2.4.2, 2.3.2 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

(b) New Licensing Issues

(c) General Comments

KEY WORDS

Will be furnished from checklist.

GENERAL COMMENTS

This paper is an overview of past studies by LLNL in which the authors describe their approach to determining the service life of the waste package and, ultimately, the rate of radionuclide release. Their studies indicate that the containers will be above 100 °C for over 1000 yrs and are expected to undergo uniform corrosion attack resulting in a container lifetime of 67,000 yrs. However, they also state that containers in the periphery of the repository are expected to be at a temperature below 100 °C. Under this lower temperature condition, condensation of steam will occur and lead to nonuniform coverage of the container by the condensate which, in turn, could lead to localized corrosion and a reduced lifetime. The authors also state that Zircaloy will undergo SCC.

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WASTE PACKAGE DATA REVIEW FORM

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory, Livermore, CA 94550

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

R. D. McCright,"FY 1985 Status Report on Feasibility Assessment of Copper-Base Waste Package Container Materials in a Tuff Repository", UCID-20509, September 30, 1985; work performed under auspices of U.S. Department of Energy contract number W-7405-ENG-48. Available from National Technical Information Service.

DATE REVIEWED: 1-28-86/Revised 1-31-86/3-20-86

TYPE OF DATA

Experimental data and literature review. Corrosion, general and localized. Compilation on canister fabrication methods.

MATERIALS/COMPONENTS

Cu (CDA 102), Al bronze (CDA 613), 70/30 Cu-Ni (CDA 715)

TEST CONDITIONS

Experimental Studies

J-13 well water; boiled down J-13 water concentrated by a factor of 100, with and without H_2O_2 added, and with and without γ-radiation. Temperature from 23°C, 55°C, and to 80°C.

Qualitative tests for corrosion in steam.

Literature review covers higher temperatures in steam.

METHODS OF DATA COLLECTION/ANALYSIS

Electrode potentials and potential dynamic scans. Some visual observations.

AMOUNT OF DATA

Experimental Results

8 Electrode potential vs. log of current density (polarization curves) without radiation and one curve with radiation

Corrosion rates vs temperature (23°, 55°, and 80°C).
Corrosion potentials vs temperature (23°, 55°, and 80°C).
Passive current vs temperature (23°, 55°, and 80°C).
Differences between pitting and corrosion potentials vs temperature (23°, 55°, and 80°C).
Effect of irradiation and H_2O_2 on corrosion potential (23°, 55°, and 80°C)

Calculated Results

2 potential-pH diagrams-Cu/J-13 water with and without irradiation.

Literature Findings

Plots--Effect of temperature and oxygen partial pressure, P_{O_2}, on stability of Cu oxides.
Effect of Ni and temperature on corrosion of Cu-Ni alloys
Corrosion rates of canister tube materials

Tables--Corrosion rates of Cu in various waters
Corrosion rates of Cu-base alloys
Thermophysical and mechanical properties of Cu-base alloys

Compilation of methods of fabrication that might be suitable.

UNCERTAINTIES IN DATA

Experimental Results-Bars show the range of values for replicate tests (in general,3 tests).

DEFICIENCIES/LIMITATIONS IN DATABASE

None specifically mentioned.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

Related to issues 2.2.4, 2.2.4.1, 2.2.4.2.2 in the ISTP for the Nevada Nuclear Waste Storage Investigation (NNWSI) Project.

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(b) New Licensing Issues

(c) General Comments

KEY WORDS

Corrosion, Cu, Cu alloys, tuff

GENERAL COMMENTS

This report gives a comprehensive description of the reasons and purposes of a study to see if Cu-base materials can be used in a tuff environment. It includes a literature search of extant data concerning some corrosion aspects of Cu-base materials, but this literature review is far from exhaustive. Some physical and mechanical properties of these materials are also included in tables.

The experimental part is rather limited, particularly concerning the effect of γ-radiation.

The authors, rightly, emphasize the fact that the major source of uncertainty in the corrosion behavior of Cu-base alloys in the Nevada repository is the lack of information concerning the effect of γ-radiation. Some of the experimental results indicate that one of the major consequences of radiation is H_2O_2 production, so that the addition of H_2O_2 mimics reasonably well the radiation effects.

However, a major concern on the part of the reviewer is that the possible effects of radiation on solid-state transport in the protective oxide films are not taken into consideration. Cu oxides have shown large photoeffects, and some of the corrosion potential curves (such as Figs. 24, 26, and 27) have striking similarities with photopotential curves. See, for instance, J. Elec. Soc. 125, 1598 (1978) and "Encyclopdia of the Electrochemistry of the Elements,: Vol. II, p. 461. L. Mancel Dekker, NY (1974). Enhancement of the number of charge carriers in the film might modify substantially the oxidation rates of the metal, particularly in wet steam environments.

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

DATA SOURCE

(a) Organization Producing Data: Rockwell Science Center

(b) Author(s), Reference, Reference Availability: J. B. Lumsden, Pitting Behavior of Low Carbon Steel, BWI-TS-014, August, 1985.

DATE REVIEWED: 2/28/86

TYPE OF DATA: Experimental, pitting corrosion, general corrosion, theoretical discussion

MATERIALS/COMPONENTS: Low carbon steel of type ASTM A27, Grade 60-30

TEST CONDITIONS: Synthetic ground water (Grande Ronde No. 4, low oxygen) plus packing material (75% basalt and 25% bentonite) at 50C - 200C for 1 hr., 1 day, 1 week for pitting and passivation studies at 200C, 150C and 100C.

METHODS OF DATA COLLECTION/ANALYSIS: Potentiostatically produced electrochemical data were recorded on an x-y recorder. Pitting data was based on electrode potentials, polarization curves, and light microscopy observation of specimens. Computer controlled polarization measurements and polarization resistance determinations were used to determine corrosion rates.

AMOUNT OF DATA: 4 graphs of current, uAmps(10^{-1} to 10^{4}), vs. potential (-2.0 to 0.8 volts vs. Standard Hydrogen Electrode,SHE) showing anodic and cathodic regions, 1 graph of corrosion rate (0 to 250 um/yr) vs. time (2400 hr.), 1 graph of potential (-700 to -300 volts vs. SHE) vs. time (2400 hr.) at 100C, 150C and 200C, 1 table of corrosion potentials and currents.

UNCERTAINITIES IN DATA: Not given by authors.

DEFICIENCIES/LIMITATIONS IN DATABASE: Not given by authors.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

(b) New Licensing Issues

(c) General Comments: Data insufficient to address licensing issue

KEY WORDS: steel, corrosion (pitting), corrosion (general), electrode potential, polarization, basalt, experimental data, data analysis, visual examination

GENERAL COMMENTS: The experimental apparatus for exposing specimens and making electrochemical measurements and the experimental approach appear to be good. Effects and values of testing solution pH should be noted. Pitting potentials given in the paper may be uncertain. The conclusion that pitting will not occur under expected repository conditions should not be made based solely on the type of data presented in the paper. Additional measurements are needed which would give more information on the pitting potential and on pit propogation. Information on pitting potentials or pitting tendencies determined by using cyclic polarization techniques, and stimulation techniques would supplement these data. Additional corrosion rate studies with increased time and temperature changes and effects of alternate wetting and drying are needed for use in projecting corrosion rate. Other aspects of corrosion, including those involving cracking, should be considered with this material.

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WASTE PACKAGE REVIEW FORM

DATA SOURCE: Materials Research Society 1985 Symposium on the Scientific Basis for Nuclear Waste Management, Stockholm, Sweeden, Sept. 9-11, 1985.

(a) Organization Producing Data: Lawrence Livermore National Laboratory

(b) Author(s), Reference, Reference Availability: Derivation of a Waste Package Source Term for NNWSI From the Results of Laboratory Experiments, UCRL-92096, Preprint. Virginia M. Oversby, Lawrence Livermore National Laboratory, P. O. Box 808, Livermore, CA 94550, and Charles N. Wilson, Westinghouse Hanford Company, P. O. Box 1970, Richland, WA 99352.

DATE REVIEWED: 4/10/86

TYPE OF DATA: Experimental data and model for dissolution of spent fuel

MATERIALS/COMPONENTS: Uranium, spent fuel from Turkey Point PWR, H. B. Robinson PWR, various other fuel

TEST CONDITIONS: Three tests with 4 different fuel rod configurations each were discussed; (1)Turkey Point fuel in dionized water and (2)Turkey Point fuel in J-13 water and (3) H. B. Robinson fuel in J-13 water. Test conditions and environment include J-13 water, deionized water, dilute bicarbonate ground waters, under air at ambient hot cell temperatures, Tuff, Paintbrush Tuff, Tonapah, Nevada, Yucca Mt., fuel rod split, fuel rod with slits, fuel rod with holes, undefected fuel rods. Model uses data from these tests and allows 45 yrs. for the borehole to fill with water.

METHODS OF DATA COLLECTION/ANALYSIS: Uranium and radionuclides from spent Turkey Point PRW fuel in deionized water were monitored for 8 mos., terminated, restarted and monitored for 4 mos. Spent fuel from Turkey Point PWR and H. B. Robinson PWR was tested for uranium and radionuclide release in J-13 well water which is a dilute bicarbonate ground water. Test specimens and vessels were washed and all U and radionuclides in each phase of testing were counted. Model is given to show maximum dissolution of spent fuel at Yucca Mt., NV. A U solubility of 5 mg/l in J-13 water is used, and it is assumed that all water leaving carries away this maximum amount of U, resulting in a fractional release rate of 6.4 x 10^{-8}/yr.

AMOUNT OF DATA: 2 graphs, 3 tables and 1 figure - Graph showing uranium concentration (unfiltered) in ug/ml (0.001 to 10 log. range) vs. time for 180 days for tests using Turkey Point fuel in J-13 water for each of the 4 specimen types; Graph showing uranium concentration in ug/ml (0 to 5 range) vs. time for up to 230 days for Turkey Point and H. B. Robinson bare fuels tested in J-13 water; Table showing Release Data for Turkey Point Bare Fuel in J-13 Water with U in ug, $^{239+240}Pu$, ^{241}Am, ^{244}Cm, ^{99}Tc in nCi and ^{137}Cs in uCi; Table showing total measured fractional release data in parts per 100,000 of the test specimen inventory for the 3 test conditions and for the 4 types of specimens with numbers ranging from .003 to 739, ^{237}Np in addition to the nuclides given above are in the table; Table showing Concentration of Tc in Solution for Turkey Point and H. B. Robinson Bare Fuel in J-13 Water for 63 to 223 days with amounts ranging from 198 to 468 pCi/ml; Figure showing schematic of fuel container in a vertical emplacement hole.

UNCERTAINTIES IN DATA: Release data showing higher total release in deionized water(DIW) than in J-13 water may be due to fine particles dispersed in DIW at the beginning of the test.

DEFICIENCIES/LIMITATIONS IN DATABASE: Authors state that the following is needed; dependency of uranium solubility on temperature, rate of failure of the containment barrier, rate of breach of cladding and rate of oxidation of the UO_2 fuel matrix. Model assumes that oxide matrix fuel is not degraded by oxidation or other mechanisms. Model does not account for ^{14}C release from container, activation products from container, initial rapid release of gap/grain boundary components or oxidation of the oxide fuel matrix.

APPLICABILITY OF DATA TO LICENSING

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

(a) Relationship to Waste Package Performance Issues Already Identified

(b) New Licensing Issues

(c) General Comments : These data are pertinent to licensing. More data and refinement of model are needed.

KEY WORDS: data analysis, experimental data, model, J-13 water, field site, tuff, Yucca Mt., Tonapah, NV, Paintbrush Tuff, radionuclide, uranium, pressurized water reactor(PWR), spent fuel, fuel rods, ^{244}Cm, ^{241}Am, ^{240}Pu, ^{237}Np, ^{239}Pu, ^{137}Cs, ^{99}Tc, ^{14}C, Turkey Point Reactor, H. B. Robinson Reactor, deionized water(DIW).

GENERAL COMMENTS: The results and work appear to be of good quality. The results are specific to waters, locations, specimens, etc. More information on experimental and analytical methods would be helpful. The assumption that a stainless steel or low carbon steel will remain intact for 10,000 yrs. is incorrect due to the presence of moisture and other possible corrosives . The idea of presenting a model which the authors do for these conditions is a good approach. Additional data on U solubility, containment barrier failure, breach of cladding, and UO_2 oxidation, as suggested by the authors, needs to be obtained.