UNITED STATES DEPARTMENT OF COMMERCE
National Institute of Standards and Technology
(formerly National Bureau of Standards)
Gaithersburg, Maryland 20899

August 11, 1989

Mr. Charles Peterson
Technical Review Branch
Division of High-Level Waste Management
Office of Nuclear Materials Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Monthly Letter Status Report for March 1989 (FIN-A-4171-9)

Dear Mr. Peterson:

Enclosed is the March 1989 monthly progress report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-9).

Sincerely,

Charles G. Interrante
Program Manager
Corrosion Group
Metallurgy Division

Enclosures

Distribution:
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Monthly Letter Report for March 1989

Published August 1989

(FIN-A-4171-9)

Performing Organization: National Institute for Standards and Technology (NIST)
Gaithersburg, MD 20899

Sponsor: Nuclear Regulatory Commission (NRC)
Office of Nuclear Materials Safety and Safeguards
Washington, DC 20555

## TASK 1 -- REVIEW OF WASTE PACKAGE DATA BASE

STATUS OF DATABASE

| | Current Month | Previous Month |
|---|---|---|
| Number of citations | 1086 | 1040 |
| Number of completed reviews | 83 | 78 |

### Status of Recently Listed Reviewable Documents

Reviewable documents are classified as follows: Category 1 documents are currently being reviewed. Categories 2 and 3 are documents that will be entered into the database with citation information and authors abstracts, with the Category 2 documents being flagged for review when time permits.

Yucca Mountain Project

5 Reports currently under review (Category 1).
28 Reports to review when time permits (Category 2).
1 Reports to file with cross reference(s) to other reports (Category 3).
5 Reports identified and not yet categorized.
10 Reports received and not yet categorized.

GLASS -- VITRIFIED WASTE FORM

1 Reports currently under review (Category 1).
4 Reports to review when time permits (Category 2).
0 Reports to file with cross reference(s) to other reports (Category 3).
0 Reports identified and not yet categorized.

Database searches for the month of March 1989 include Metadex and Engineering Abstract. Examples of the search conducted for each of these databases are in this report (see p. 11 to 12).

STATUS OF REVIEWS OF NNWSI REPORTS

Yucca Mountain Project -- Reports recently identified for review

Five reports have been identified for review. Two are on water chemistry, two are on the subject of the container, and one is on the performance of cladding.

The effects of temperature on the chemistry of the water in tuff, are addressed in this study. Experiments, in which J-13 water was exposed to a tuff environment at 90°C and 150°C for 64 days, were conducted. Results indicate that changes in water chemistry were minor [Oversby 1984].

This study is identical to the report above, except that it was carried out at 120°C for a period of 72 days. As before, the results indicate that heating has little effect on water chemistry [Oversby 1984].

The process used in selecting the ferrous canister materials is described. Cost of material, corrosion resistance, fabrication costs, and weldability were factors taken into consideration. On this basis, AISI 304L, AISI 321, AISI 316L, and Incoloy 825 were chosen as candidate canister materials. AISI 1020 steel is to be considered as a borehole liner material [Russell 1983].

Using uncertainty analysis techniques, the corrosion performance of nuclear waste canisters has been investigated. Based on their assumptions, the results indicate that with 0.99 probability, corrosion depth at 300 years will be less than 1.92 mm and at 1000 years, corrosion depth will be less than 2.14 mm [Sutcliffe 1983].

This is a description of the second series of tests conducted to evaluate the effectiveness of breached cladding as a barrier to radionuclide release. The objective of the test is to determine whether any credit for containment can be accorded to the cladding. The differences between this second test series and the first series are described [Wilson 1984].


1. Oversby, V. M., "Reaction of the Topopah Spring Tuff with J-13 Well Water at 90°C and 150°C," UCRL-53552, May 1984.

2. Oversby, V. M., "Reaction of the Topopah Spring Tuff with J-13 Water at 120°C," UCRL-53574, July 1984.

3. Russell, E. W., McCright, R. D., O'Neal, W. C., "Containment Barrier Metals for High-Level Waste Packages in a Tuff Repository," UCRL-53449, October 1983.

4. Sutcliffe, W. G., "Uncertainty Analysis: An Illustration from Nuclear Waste Package Development," UCRL-90042, October 1983.

5. Wilson, C. N., "Test Plan for Series 2 Fuel Cladding Containment Credit Tests," HEDL-TC-2353-3, October 1984.


Yucca Mountain Project --

Category 1 -- Reports currently being reviewed


1. UCRL-21013, "Summary of Results from the Series 2 and Series 3 NNWSI Bare Fuel Dissolution Tests," November 1987.

2. ANL-88-14, "The Reaction of Glass During Gamma Irradiation in a Saturated Tuff Environment, Part 3: Long-Term Experiments at $1x10^4$ rad/hr," February 1988.

3. Ringas, C. and Robinson, F., "Corrosion of Stainless Steel by Sulfate-Reducing Bacteria - Total Immersion Test Results," NACE, Corrosion, Vol. 44(9), September 1988.

4. UCID-21472, "An Annotated History of Container Candidate Material Selection," July 1988.

5. WHC-EP-0096 (formerly HEDL-7665), "Initial Report on Stress-Corrosion-Cracking Experiments Using Zircaloy-4 Spent Fuel Cladding C-Rings," September 1988.

Category 1 (continued) - Status of Reviews not yet sent to NRC and WERB

| Document No. | Assigned to Reviewer | First Draft Completed | Lead Worker | Program Manager |
|---|---|---|---|---|
| UCRL-21013 | 2/17/89 | 2/28/89 | | |
| ANL-88-14 | 2/17/89 | | | |
| Ringas, 1988 | | 1/30/88 | 2/10/89 | |
| UCID-21472 | 2/21/89 | | | |
| WHC-EP-0096 | 2/21/89 | | | |

Category 2 -- Review as time permits (new entries for this reference data file)

1. UCRL-89830, "Nuclear Waste Package Design for the Vadose Zone in Tuff," February 1984.

2. UCRL-90857, "Parametric Testing of a DWFF Borosilicate Glass," January 1985.

3. UCRL-87621, "Leach Testing of Waste Forms Interrelationshp of ISO and MCC Type Tests," May 1982.

4. UCRL-53629, "The Reaction of Topopah Spring Tuff with J-13 Water at 150°C - Samples from Drill Cores USW G-1, USW Gu-3, USW G-4, and UE-25h#1," March 1985.

5. UCRL-53442, "Reaction of Bullfrog Tuff with J-13 Well Water at 90°C and 150°C, September 1983.

6. Smith, H. D., "Spent Fuel Cladding Characteristics and Choice of Experimental Specimens for Cladding-Corrosion Evaluation Under Tuff Repository Conditions," HEDL-TC-2530, November 1984.

7. UCRL-98029, "Assessment of Engineered Barrier System and Design of Waste Packages," June 1988.

8. HEDL-TME 85-22, "Results from Cycles 1 and 2 of NNWSI Series 2 Spent Fuel Dissolution Tests," May 1987.

9. UCRL-21019, SAN-662,-027, "Recent Results from NNWSI Spent Fuel Leaching/Dissolution Tests," April 1987.

Category 3 -- File and cross reference

1. UCRL-97805, "An Approximate Calculation of Advective Gas Phase Transport of $^{14}C$ at Yucca Mountain, Nevada," December 1987.

OTHER REPORTS ON VITRIFIED WASTE FORM --

Category 1 -- Reports currently being reviewed

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

Status of Reviews not yet sent to NRC and WERB

| Document No. | Assigned to Reviewer | First Draft Completed | Lead Worker | Program Manager |
|---|---|---|---|---|
| PNL-5157 Chapter 4 | 6/20/88 | 1/28/89 | 2/3/89 | |

Category 2 -- Review as time permits

None this month.

Category 3 -- File and cross reference

None this month.

TASK 3 -- LABORATORY TESTING

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

March 1989 Report:

In preparation for the testing of specimen ST8, after the digital oscilloscope and the counter were set to the same threshold value, a method was devised for systematically recording the level of the signal amplification being used. A nylon ball of a known weight was dropped onto the test specimen from a fixed height through a nearly vertical glass tube. The nylon was found to give a response that is within the range of frequencies detected earlier for the acoustic signals associated with cracking of the test specimen. The nylon ball is held in place with house vacuum, which is shut off to release the ball onto the specimen. The acoustic signal is digitally recorded on a floppy disk. This ball-drop method worked well but no accurate method for establishing the height of the rebound was implemented, so that remains as a task to be done before the next specimen is tested. Using this method, the sensitivities of the available transducers were measured and the most sensitive transducer was chosen for use with the test of specimen ST8. The transducers used in this work are among the least expensive on the market and they have high variability in their output voltage for a given level of input signal.

Specimen ST8 was tested. The preliminary assessment of this test indicated some problems, but a large portion of the data appears to be suitable for analysis. Next month will be spent looking over the data to determine whether some aspects of the automated data acquisition system failed to perform adequately. For much of the code, this is the first set of data to be taken and analyzed, so it is expected that parts of the code will have to be rewritten (corrected) after inspection of the raw data and the results of the many calculations made from it.

**January 1989 Report -- Note -- A duplicate of the December report was mistakenly represented as the January report. The following is the true report for January.**

Six specimens were returned from the machine shop in the remachined condition. Specimens ST6v to ST11v were transformed into ST6 and ST11. This was done by using a comparatively blunt cutter, of about 0.025 inches (root radius), to cut away the metal that contained the sharp radius of about 0.007 inches. In order to assure that the sharp root radius was completely removed, a slightly deeper cut was required. The result was a notch that is about 0.048 inches deep on each side of the nominally 0.250-inch-thick specimens. In addition, the needed hydrogen sulfide gas has been received from the supplier.

The data tables collected during the test were reformatted to permit tabulations of pertinent parameters that will be used in the comparison of acoustic-energy detected and strain-energy released. These parameters include stress-intensity factor, strain-energy release, various measures of time, crack-length averages from electric-resistance measurements that represent behavior during various intervals of time, compliance of the test specimen, and acoustic-energy detected over various intervals of time.

Before, this phase of this study will be considered complete, a fully successful test should be conducted. Perhaps this test should use a transducer that has high sensitivity to acoustic events. In addition, the electric-resistance measurements of crack length must be fully functional throughout this test. After a fully successful test has been conducted, and using that same transducer and test specimen, it would then be desirable to conduct a calibration of the acoustic system, using either or both of two methods. One method involves bouncing a ball onto the surface of the test specimen and measurement of the coefficient of restitution and the energy released by the ball to the test specimen. The other method involves use of a laser pulse. A third method (not recommended unless the others prove too difficult to execute) involves calibration using the elastic energy stored in a piano wire. The wire is snapped onto the surface of the test specimen to release known level of energy to the test specimen.

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

The report is for February and March 1989 is as follows: The surface of the specimens, after exposure, has been characterized using a profilometer, a device that measures surface irregularity. Both sides of each specimen were examined, and the maximum depth of attack was determined. An optical microscope, at 50x, was used to determine the number of pits on the metal surface. In general, specimens in a high-resistivity environment show localized attack while specimens removed from low-resistivity environments have undergone a more uniform form of attack.

**January 1989 Report -- Note -- A duplicate of the December report was mistakenly represented as the January report. The following is the true report for January.**

The second series of experiments has been terminated, and the specimens are in the process of being examined. The surface roughness measurements indicate that some specimens have undergone localized corrosion attack. The microscopic examination, which is yet to be done, will reveal the form of the attack. Visual examination reveals pitting in some of the specimens. The data are in the process of being evaluated.

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

Studies of literature and additional data and specimen analysis in preparation for writing a paper continue. The report that was submitted to NRC earlier will be put in the form of an NIST Internal Report.

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

Preparations were made for testing in unconcentrated J-13 water. Most previous tests have been conducted in ten-times concentrated J-13 water, and this was diluted to unconcentrated J-13 water. The concentrated mixture was diluted. There could be some difference in preparing the water in this manner and preparing a normal strength concentration originally. These differenes could arise due to the differences in the solubilities of the constituents added to make the J-13 water. This question will be addressed at a later time. The pH of the unconcentrated J-13 water prepared from the ten times concentrated J-13 is 8.3. A Zr-4 specimen was placed in this solution and the open-circuit electrode potential was monitored for twenty five minutes. The initial potential versus a saturated calomel electrode was -0.787 V and the potential after twenty five minutes was -0.515 V indicating that some passivation was occurring.

Cyclic polarization measurements of Zr-4 in unconcentrated J-13 water at 95°C. The resulting curve showed that passivation occurs, and there is no indication of pitting. The corrosion potential shifts from approximately -0.400 V to +0.300 V after the anodic portion of the cyclic polarization curve indicating further passivation and the presence of a surface film. Future plans are to complete a series of tests on the Zircaloy materials and on the Zircaloy unused cladding and to bring all of the data together in a report.

TASK 4 -- GENERAL TECHNICAL ASSISTANCE

## SITE CHARACTERIZATION PLAN

The NIST reviewed the Consultation Draft Site Characterization Plan (CDSCP) for Yucca Mountain, and developed sixteen critical comments for consideration by the DOE. These were submitted to the NRC in March 1989.

The Site Characterization Plan (SCP), became available for review in January 1989. NIST found that of the sixteen critical comments submitted in 1987, that thirteen had been implemented. In addition, three were partly implemented or not implemented at all. The 1989 review of the SCP resulted in development of eight new comments. In addition, the three from 1987 were reworded and resubmitted. The following is a brief review of these eleven comments.

### EIGHT NEW SCP COMMENTS SUBMITTED BY NIST

7.4.2.6.4 - Activities to Determine Transgranular Stress Corrosion Cracking Susceptibility
- Modeling assumes homogeneity of moisture on the container surface.

7.4.3.2 - Glass Waste Form Performance Research
- The SCP ignores local pH changes, from building materials, that effect dissolution and migration of radionuclides.

7.4.5.4.5 - Waste Package Environment Model
- Effects of air flow on the corrosion of container should be considered.

7.4.5.4.6 - Corrosion Model
7.5.4.6 - Metal Barriers
- Use of the word "uniform" corrosion is misleading, recommend that the term "general" corrosion be used and defined.

8.3.5.9.2.1.1 - Establishment of Selection Criteria and their Weighing Factors
- The DOE Peer Review Panel must consist of recognized experts in materials science.

8.3.5.9.3.2.7 - Transgranular Stress Corrosion Cracking
- The SCP considers only one mechanism for T-SCC for modeling.

8.3.5.9.3.2.7 - Transgranular Stress Corrosion Cracking
- Placement of the repository in the unsaturated zone does not reduce uncertainties in understanding of corrosion behavior as implied in SCP.

8.3.5.10.2.1.1 - Dissolution and Leaching of Spent Fuel
- The SCP does not take into consideration the effects of metal contamination of water on the dissolution and migration of radionuclides.

THREE CDSCP COMMENTS RESUBMITTED BY NIST

Comment No. 74 - Waste Package Design Features that Affect the Performance of the Container
- Test Methods lack peer review

Comment No. 77 - State of Stress of the Container
- The effect of general corrosion on stresses in the weld and weld heat-affected zones should not be neglected.

Comment No. 80 - Assessment of Degradation Modes in Copper-Based Materials
- Comments in this section of SCP indicate a lack of familiarity with published literature and raises concerns about test plans.

Activites on the ambiguities of the term substantially complete containment as used in the Code of Federal Regulations, in 10 CFR-60 continued and a set of elements of proof developed by C. Interrante and reviewed by staff members of the NIST HLW program were submitted this month to the NRC program manager. A copy of this work is attached to these minutes.

TRIP REPORT -- U. BERTOCCI, JANUARY 26-28, 1989

a brief report on the January 26-28, 1989 visit of Dr. Bertocci to the CNWRA is given in response to our request:

The purpose of my visit to the San Antonio Center was to establish contacts with the resident geologists (in particular Bill Murphy), and discuss my concerns about physical and chemical conditions inside the repository, and their influence on corrosion. Bill Murphy could give me a more precise feeling, based on his expertise in geology, about what is known with certainty, what is reasonable, but not sure, and what is unknown or controversial about the conditions inside the Yucca Mountain, than I could derive by reading DOE reports.

The main topics discussed were about the availability of water inside the repository, and its estimated movements. Bill Murphy explained to me that very little solid information is in our hands at the moment, so that even the direction of motion of water, up or down, is not established with certainty.

We also discussed oxygen availability and transport: Bill Murphy pointed out that, apart from radioactive heating around the buried containers, the temperature inside the mountain is not subject to seasonal fluctuations. A possible mechanism for gas transport within the porous tuff, however, is atmospheric pressure fluctuations.

I considered my visit to the center to have been very useful, since it clarified many points on which I was uncertain and it allowed me to establish contacts for future interaction.

# SDI293, UD 8904, SER. DD023

File(s) searched:

File 293:Engineered Materials Abs 86-89/Apr
(Copr. 1989 ASM INTERNATIONAL)

Sets selected:

| Set | Items | Description |
|---|---|---|
| 1 | 2 | HIGH()LEVEL()WASTE? ? OR RADIOACTIVE()WASTE? OR NUCLEAR()WASTE? |
| 2 | 292 | STEEL? ? OR ZIRCALOY? ? OR TITANIUM? ? OR COPPER |
| 3 | 0 | S1*S2 |
| 4 | 0 | ANNA FRAKER, 223, B-254, X6009 |

Prints requested ('*' indicates user print cancellation) :

| Date | Time | Description |
|---|---|---|
| 23mar | 22:54EST | PR 3/5/1-25 (no items to PRINT) |

Total items to be printed: 0

11

# SDI032, UD 8904, SER. DD022

File(s) searched:

File 32:METADEX 66-89/APR
(Copr. 1989 ASM International)

Sets selected:

| Set | Items | Description |
|---|---|---|
| 1 | 4 | HIGH()LEVEL()WASTE? ? OR RADIOACTIVE()WASTE? OR NUCLEAR()WASTE? |
| 2 | 1824 | STEEL? ? OR ZIRCALOY? ? OR TITANIUM? ? OR COPPER |
| 3 | 1 | 1*2 |
| 4 | 0 | ANNA FRAKER, 223, B-254, X6009 |

Prints requested ('*' indicates user print cancellation) :

| Date | Time | Description |
|---|---|---|
| 11mar | 03:47EST | PR 3/5/1-25 (items 1-1) |

Total items to be printed: 1

12

Substantially Complete Containment

1. Problem -- The term substantially complete containment is ambiguous and creates a potential for undesirable legal delays at the time of licensing.

2. Solutions

(A) Use a restatement such as "the law intends that during a 1000-year containment period, no radionuclides shall escape the canister, within the limits of extant containment technology". However, the DoE must be required to demonstrate that it meets this requirement, and the term "extant containment technology" is still ambiguous.

(B) Establish the number of canisters that may fail without posing a safety or health or retrievability hazard, and require DoE to demonstrate that this requirement has been met. This approach may be interpreted as an assault upon the dual (redundant) legal requirements for both a containment period <u>and</u> a controlled release rate, and for this reason it may be ill advised.

(C) Define what is meant by no failures, as in 3 through 8 below. It is noted that there exist independent legal requirements for containment and release of radionuclides from a waste package. It is expected that the release rate requirements must be met even during the period of containment, i.e. if container failures that occur within the containment period satisfy the containment requirement but pose a release-rate violation, this is unacceptable.

3. Definition of substantially complete containment

Substantially complete containment is herein defined as no more than one container failure per _____ containers in the repository for the first 1000 years after repository closure, and less than one failure per _____ containers in the first 300 years. These specified maximum numbers of failures arise from the need, in a performance assessment, to define what is meant by no failures. To the writers of this definition, the above requirement is a practical interpretation of the requirement for complete containment. Perhaps only one of these requirements will be sufficient, either that for 300 or for 1000 y, and a choice to include both includes a consideration of conservatism in the design. It is also necessary to add to the above specified maxima, a statement about the uncertainty of the N failures in 1000 containers. The form of the statements of uncertainty (confidence intervals or any statements about the uncertainty of the N failures in 1000 containers) depends upon the models and the experimental data available. Thus, these statements can not be addressed in a simple way here.

Any argument or proof that the specified maxima will not be exceeded must be accompanied by a demonstration of sufficient understanding of pertinent factors, e.g. those listed below as 3.1 to 3.3 and it must include the use

of models and assessments, e.g. those outlined in 3.4. Together, the understandings and the assessments will constitute elements of proof that the requirement for substantially complete containment has been met.

3.1 The EBS (engineered barrier system) environment (chemistry, temperature, pressure) and uncertainties in the understanding of it over the service life

- 3.1.1 Engineering discrepancies in repository construction or calculations
- 3.1.2 Uniformity of waste-package sites within the repository
- 3.1.2.1 Thermal cycle and radiation field for each waste package site -- including determination of the variances of the thermal contents and the radiation fields of waste packages
- 3.1.2.2 Uncertainty due to geological (and hydrological) knowledge, repository heterogeneities, temporal effects, etc.
- 3.1.2.3 Scientific uncertainties, e.g. temporal effects, mechanism of ground water transport, amount of rainfall, effect of radiolysis on environment, etc.

3.2 Characteristics of waste package material/component used for containment credit

- 3.2.1 Variability and/or uncertainty in chemistry, stability, microstructure, stress state, etc. vis-a-vis behavior
- 3.2.1.1 Uncertainties related to laboratory testing
- 3.2.1.2 Uncertainties due to inhomogeneities
- 3.2.2 Scientific uncertainties related to critical elements of technical arguments, e.g. alteration mechanism and stability of container materials over total containment period, effects of long-term radioactivity on materials properties, etc.
- 3.2.3 Fabrication -- effects on service behavior

3.3 Mechanistic understanding of physical and chemical processes associated with material/environment alterations during service life of materials used for containment credit.

- 3.3.1 Tests of materials durability, e.g. local/general corrosion, mechanical and other failures, etc. -- standardized test methods that have passed peer review shall be used to determine the durability of materials.

- 3.3.1.1 Arguments for the absence of a given mechanism of failure of a container will be supported by tests that statistically demonstrate a lack of interaction between the material and the range of environments. These tests will be either passed or failed for all expected environments, even for environments having a probability for their occurrence as low as one chance in a thousand.

3.3.1.2 Demonstrations that mechanistic arguments are adequate for the various parts of a container, e.g. weldments, base metal, lean-chemistry region, rich-chemistry regions, attachments, etc.

3.3.2 Scientific uncertainties shall be studied at the level of vigor needed to resolve all open questions over the period during which containers can be retrieved. These uncertainties might include long-term durability, the mechanisms for alterations that occur at times beyond those of empirical tests, the stability of alloy phases, etc. The uncertainties arise because the term of service behavior of the waste container system far exceeds the times for which empirical data can be made available.

3.3.3 Determinations of all reaction products are needed to describe the environment as a function of time.

3.4 Performance Assessment -- compute the times expected for container failures, using models and sensitivity analyses -- note that to do this acceptably, alteration mechanisms must meet some minimum levels of acceptability described above under 3.3.1.1, pass/fail criteria. In addition, due to any scientific uncertainties a fault tree shall be used and separate analyses shall be conducted for each branch, using "best judgement" where data can not be obtained to support estimates of performance. The performance will be such that no leaks from waste packages will be permitted during the containment period after repository closure, scientific uncertainties notwithstanding. An uncertainty calculation shall indicate that under the most adverse conditions foreseeable for reasonable scenarios, i.e. taking all scientific uncertainties (related to container materials and environment) into account, the effect on release of radionuclides from the waste package will be the same as if there were no more than one container failure per _____ containers in the repository.

3.4.1 Solid-state atomistic release of radionuclides from the exterior of a container, due to surface reactions, must be considered in the definition of failure. For the purposes of this document, any release due to this or any other cause shall be regarded as a failure if the release from the container is greater than ___Ci/y.

3.4.2 Clarity of exposition

3.4.2.1 Before submittal to the Nuclear Regulatory Commission for review, the DoE arguments, which are presented as elements of proof that the legal requirements are met for substantially complete containment, will be given peer review for clarity. Peer review can be done by various academia and industry personnel who are not directly involved in the development of the arguments.

3.4.2.2 Software -- any computer codes used to represent behavior predicted by a theory must be made available in two forms: (1) as descriptions that document how the code represents the theory and (2) as accessible code that can be both run to determine how it behaves using reviewer inputs and examined to determine how it is constructed.

3.4.3 Safety Factors -- in the performance assessment, express all engineering uncertainties with safety factors, and express all scientific uncertainties through conservative bounding calculations, as needed to satisfy the legal requirement for a conservative design.

3.4.4 Quality assurance

3.4.4.1 Assure complete containment (no leaks) at times of emplacement and of later inspections. Any leak that is discovered shall not be allowed to persist at the time of closure.

3.4.4.2 As required for credible performance assessments, conduct characterization and confirmation measurements

(A) of the repository (stability, water table, engineering aspect, etc.)

(B) of the waste package sites (uniformity, stability)

(C) of waste package components (retrievability, alteration mechanisms)

4. Exposition of requirements and specifications

The definition of the term substantially complete containment requires the DoE to furnish specific details on the proposed containment materials, various environmental factors and a mechanistic understanding of alterations that will occur over time. The level of assurance implied by the law shall be established by a set of requirements related to that information and to a performance assessment and to quality assurance. An abbreviated example of requirements and specifications that would be needed under this approach follows:

In the process used to obtain NRC approval for the design of any component of the EBS for which credit for containment will be taken, the DOE shall be required to furnish information on the following on items 4.1 to 4.5 below, taking the entire service life into account, including the repository operational period, and the pre- and postclosure periods:

4.1. The service environments (expected chemistry, temperatures, radiation and pressures as functions of time) both internal (where applicable) and external to the component, including the influence

of products of reactions between the environment and the various materials in the EBS.

4.2 State of the material before fabrication, after fabrication, and during and after the operational period.

4.3 The mechanisms by which the environment and the materials used in the waste package will alter one another and the effects of these alterations on the expected behavior of each pertinent component. Include understanding of variability of environment, materials, fabricational effects, and deformation due to handling and transportation.

4.4 The designed service life shall be computed for each failure mode of the component, using the best available information on mechanistic understandings and empirical data. A conservative design shall be assured by the use of appropriate engineering safety factors that account for any uncertainties in the materials, the environments, and this computation. For the unresolved scientific issues, bounding calculations will be needed to show the consequences of various possible alteration behaviors.

4.5 Using these bounding calculations, it will be shown the requirements for complete containment, as specified in 3.0, have been met. If the predicted number of failures exceeds the specified maximum number permitted for a given time period, the question will arise should that approach be prohibited, or with the expectation that the scientific question can be resolved before closure of the repository. Stated another way do you permit the DoE to use a questionable approach while proceeding to develop the understanding needed to resolve the scientific issues. Thus, if the latter approach is adopted, then resolution of scientific issues would become a part of the task of confirmation.

4.6 Information furnished to fulfill the requirements of 4.1 through 4.5 will not be regarded as complete unless it includes explanations of any rationale, methodology, or computations, as well as appropriate statistical treatments of data and extrapolations, including but not limited to uncertainty statements, sensitivity analyses, and appropriate bounding calculations. In addition, any computed times that are directly relatable to the release of radionuclides, such as the failure times for canisters, shall be expressed as a probability density function, but this shall be done only for factors that are independent from canister to canister, e.g. variations in canisters due to manufacturing procedures. On the other hand, scientific uncertainties that apply to the entire repository can not be expressed as distributions of time to failure. Rather, these uncertainties shall be expressed independently and shall be represented as being descriptive of a behavior of the entire repository, e.g. "There is one chance in 1000 that the repository shall be wet within a period of 50 years after repository closure."

5. Materials Specifications -- Materials used to fabricate each component used for containment credit shall be furnished to design specifications in accordance with the following good engineering practice:

5.1 Qualification and characterization tests will be used both to qualify materials for use (i.e. to assure that they meet design specifications) and to characterize physical and chemical properties as needed for mechanistic understandings of service behavior.

5.2 For both the materials and any component fabricated from them, permitted deviations from design specifications shall include only those that can be accommodated by engineering safety factors incorporated into the design.

5.3 Mechanistic understanding of expected behavior and alterations that result from service shall be supported by

(A) sufficient characterization testing as may be needed to predict behavior for the entire range of materials (composition, stress/strain states, heat treatments, etc.) that could be approved for use in a repository.

(B) adequate determinations of the breadth of environmental parameters (temperature, radiation, composition, etc.) posed by the waste package site.

(C) a definitive understanding of interactions between materials and the environments, at the level of detail needed to predict of materials behavior.

5.4 Materials specifications shall be written in a manner that assures a high degree of verifiable assurance that the material was strictly produced under a quality assurance program that sets requirements for verification of, for example, material identification, adherence to established processing procedures and compliance with specified composition, mechanical properties, inspection, rejection, etc. See Appendix B on Industry Practices for Eddy Current Testing of Carbon, Stainless and Alloy Steel Tubular Products -- AISI Steel Products Manual "Steel Specialty Tubular Products." See also 10CFR50 Section III of the ASME, Boiler and Pressure Vessel Code Committee. In addition, it is noted that fitness of a component for use in a waste package includes needed assessments of properties related to durability, and existing specifications do not address these types of assessments. Therefore, they must be added to fulfill requirements such as those indicated above, in 5.3(B) and 5.3(C), for alterations that may occur in service.

6. Fabrication specifications -- fabrication shall be done using accepted practices, "ASME Boiler and Pressure Code -- Section IX, Welding." Inhomogeneities that result from fabrication, such as variations in microstructure, and chemical potential, in and nearby weld joints, shall be characterized. Performance qualification tests of weldments will be required.

7. Confirmation tests -- during the period from emplacement until closure of the repository, in-situ and any necessary laboratory tests will be conducted to confirm the validity of conclusions that relate to alterations of the containment materials in the repository environments (extant and expected).

8. Inspections and Reports

8.1 Inspection reports shall be required for all transportation operations, of as-received materials, fabricated components, etc., so as to furnish full documentation of damages, if any, that ensue as a result of transportation.

8.2 Inspection of materials shall be conducted at each step in the life of a material used to fabricate any component that is used for radionuclide containment credit, i.e. after the material is received at the fabrication site, after fabrication of a component, after transportation, after any accident, after emplacement in a bore hole, and during the in-service preclosure period. The purpose of these inspections is to disclose any alteration that may affect the performance of the or retrievability of the component.

8.3 In service inspection shall be conducted during the period prior to repository closure to assure no leakage of radionuclides and to assure environmental conditions and expected materials alteration scenarios are accurate. Any leaking container shall be removed from the repository.

8.4 Recognized inspection procedures shall be used, such as those given in the National Board Inspection Code, a manual for boiler and pressure vessel inspectors published by the National Board of Boiler and Pressure Vessel Inspectors, Columbus, OH.

9. Other reference documents that may furnish applicable guidance -- these should be reviewed to determine the extent to which all or parts of them are applicable to the problem of assments of long-term performance of nuclear waste canisters

9.1 Precision and Accuracy -- See related ASTM practices listed below for applicability to this problem.

ASTM E 177-71 (1980) Recommended Practice for Use of Terms Precision and Accuracy as Applied to Measurement of a Property of a Material (E-11, Vol 14.02)

ASTM G16 - 71 (1977) Recommended Practice for Applying Statistics to Analysis of Corrosion Data (G-1, Vol 03.02)

ASTM C670-84 Practice for Preparing Precision Statements for Test Methods for Construction Materials (C-9, Vol 04.02)

9.2 ASTM Research Report Format Guide

9.3 Standard Methods for Mechanical Testing of Welds, ANSI/AWS B4.0-85, An American National Standard approved by ANSI Feb 20, 1985, American Welding Society, Miami, Florida.

9.4 Pipeline Safety Code, Model Code of Safe Practice, Part 6, Fourth Edition, December 1982, The Institute of Petroleum, London, John Wiley & Sons, New York.

9.5 Applicable NUREG citations

10. Specification writing -- The obvious options on the question "Who should author the specifications?" are

(A) the NRC autocratically.
(B) the NRC, the DoE and interested parties. It might take forever to reach agreement using this approach.
(C) the DoE, with NRC acting as an approving authority, which will modify various aspects as may be necessary, as for example, to promote a more conservative design, or to obtain a more definitive product description, etc.

Option (C) would appear to this worker to be the most expedient, provided that the DoE furnishes a timely response to an NRC request for the needed specifications. If it is not and if NRC is fighting a rulemaking clock, then the DoE would hold the power to hinder the rulemaking attempted by the NRC. Hence, it would be desirable to (1) keep the needed specifications themselves out of the rulemaking process and (2) include within the process a requirement for the DoE to furnish various specifications in a timely manner.

NOTE TO THE USERS OF THIS DRAFT -- The above information is furnished to provide guidelines or examples of the types of information that must be considered for inclusion in a definition of substantially complete containment. The actual development of such a definition and related rules will involve non-scientific judgement and, as such, is outside the purview of NIST.

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