UNITED STATES DEPARTMENT OF COMMERCE
National Institute of Standards and Technology
Gaithersburg, Maryland 20899

April 23, 1990

Mr. Richard Weller, Section Leader
Materials Engineering Section
Division of High-Level Waste Management
Office of Nuclear Materials Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Monthly Progress Report for February and March 1990 (FIN-A-4171-0)

Dear Mr. Weller:

Enclosed is the February and March 1990 progress report for the project "Evaluation and Compilation of DOE Waste Package Test Data" (FIN-A-4171-0). The financial information is reported separately.

Sincerely,
Anna C. Fraker

Anna C. Fraker
Metallurgist
Corrosion Group
Metallurgy Division

Enclosures

Distribution:
WM Docket Control Center (1-original)

1/1

WM-11
A-4171
NH14

Progress Report for February and March 1990

Published April 1990

(FIN-A-4171-0)

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 Quarter	Previous Month
Number of citations	1298	1181
Number of completed reviews	100	92

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

Category 1 --

1. UCID-21466, "Simulations of the Near-Field Transport of Radionuclides by Liquid Diffusion at Yucca Mountain -- Comparisons with and without Emplacement Backfill", July 1988.

2. UCID-21444, "Numerical Modeling of the Thermal and Hydrological Environment Around a Nuclear Waste Package Using the Equivalent Continuum Approximation: Horizontal Emplacement", May 1988.

Category 2 --

1. UCRL-101097, "Evaluation of Copper, Aluminum Bronze, and Copper-Nickel for YMP Container Material", May 1989.

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

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

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

3. UCID-21700, "Yucca Mountain Project Waste Package Design for MRS System Studies", April 1989.

<u>Status of Recently Listed Reviewable Documents</u>

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, and the Category 2 documents are flagged "to review when time permits."

YUCCA MOUNTAIN PROJECT

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

GLASS -- VITRIFIED WASTE FORM

0 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 February and March 1990 include Metadex, NTIS, Engineered Materials Abstract, Compendex Plus and DOE Energy. Examples of the search conducted for each of these databases are in this report (see p. 7).

STATUS OF REVIEWS OF YUCCA MOUNTAIN PROJECT REPORTS

Yucca Mountain Project -- Reports recently identified for review

Five reports have been identified for review during this month. Two of these reports are on the subject of spent fuel, one on glass dissolution, one on container performance modeling, and one on modeling of hydrides on zircaloy.

Container Performance

A review of models and mechanisms that can lead to container failure, by localized corrosion and stress corrosion cracking, are described in this report [Farmer, 1989]. Consideration is given to the applicability of the models to the leading candidate materials, austenitic stainless steels and copper based alloys. The review indicates a need for developing a better understanding of moisture refluxing at a container surface, pit initiation and propagation.

Glass

This is a semiannual report of progress in research directed at understanding the behavior of glass waste at the Yucca Mountain repository [Bates, 1986]. The experiments are designed to provide information on the release of radionuclides from specific container designs and information on possible synergistic interactions within the container that may accelerate release rates. After 39 weeks of exposure in a simulated environment, the data show that gamma irradiation has minimal effect on the dissolution behavior of glass.

Spent Fuel

The temperature history of the spent fuel and the condition of the fuel rod cladding, affects the release behavior of the radionuclides [Woodley, 1983]. Reactor fuel temperatures in excess of 1400 C promote the release of radioactive gas, but even at lower temperatures, fission products tend to concentrate around the circumference of the fuel pellets where they are readily available in the event of a cladding breach.

This report is a compilation of specifications and data requirements set by NNWSI (Yucca Mountain Project) for acceptance of high level waste, and is intended as a guide the waste form producer [Oversby, 1984]. Information and quality control testing requirements, as identified by NNWSI, are described.

Zircaloy

The generation of hydrogen in nuclear reactor fuel results in the formation of hydrides in the zirconium cladding. The orientation of these hydride platlets is affected by the internal pressure in the canister [Stout, 1989]. Because these platlets are brittle and may provide a propagation path for a crack, their orientation is important. This report models the probable strain fields induced by hydrides and dependance of their orientation on the state of stress in the cladding.

1. Bates, J.K., Gerding, T.J., T.A. Abrajano, Jr., Ebert, W.L., Mazer, J.J., "NNWSI Waste Form Testing at Argonne National Laboratory Semiannual Report July-December 1986", UCRL-15801-86-2, February 1988.

2. Farmer, J.C., McCright, R.D., "Review of Models Relevant to the Prediction of Performance of High-Level Radioactive-Waste Disposal Containers", UCRL-100172, November 1989.

3. Oversby, V.M., "The Nevada Nuclear Waste Storage Investigations Project Interim Acceptance Specifications for Defense Waste Processing Facility and West Valley Demonstration Project Waste Forms and Canisterized Waste", UCID-20165, August 1984.

4. Stout, R.B., "A Deformation and Thermodynamic Model for Hydride Precipitation Kinetics in Spent Fuel Cladding", UCRL-100860, October 1989.

5. Woodley, R.E., "The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories", HEDL-TME-83-28, UC-70, October 1983.

Yucca Mountain Project --

Category 1 -- Reports currently being reviewed

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

2. WHC-EP-0065 (formerly HEDL-7637), "Electrochemical Corrosion - Scoping Experiments -- An Evaluation of Results", September 1988.

3. UCRL-100395, "Waste Package Performance Assessment for the Yucca Mountain Project", February 1989.

4. UCRL-21076, "Electrochemical Corrosion Studies on Copper-Based Waste Package Container Materials in Unirradiated 0.1 N $NaNO_3$ at 95°C", May 1988.

5. UCRL-101615, "Prototype Engineered Barrier System Field Tests Progress Report", July 1989.

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
WHC-EP-0096	2/21/89			
WHC-EP-0065	12/18/89			
UCRL-100395	12/18/89			
UCRL-21076	2/2/90			
UCRL-101615	2/12/90			

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

None this quarter.

Category 3 -- File and cross reference

1. Dailey, W., Ramirez, A., Ueng, T., and Latorre, R., "High Frequency Electromagnetic Tomography," UCRL-101865, September 1989.

2. "Progress Report on the Scientific Investigation Program for the Nuclear Waste Policy Act (Section 113), DOE/RW-0217P, February 1990.

OTHER REPORTS ON VITRIFIED WASTE FORM --

Category 1 -- Reports currently being reviewed

None this quarter.

Category 2 -- Review as time permits

None this quarter.

Category 3 -- File and cross reference

None this quarter.

TASK 3 -- LABORATORY TESTING

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

February to March 1990:

Modifications to the laboratory, in preparation to our move, have been made, and transfer of the laboratory is now complete.

The laboratory draft report entitled "Effect of Oxygen Transport and Resistivity of the Environment on the Corrosion of Steel", which was submitted to the NRC late last year, has gone through review and is scheduled to be published as an internal report, NISTIR 90-4266. Copies of the report will be sent to the NRC as soon as they become available.

Because of the interest in retrieval of the 20 year stainless steel specimens now located at six underground test sites, information on accessibility of the sites and costs of retrieval are being obtained. Contact with personnel at these sites, all on government owned land, reveals that all sites are intact.

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

February and March 1990:

The purpose of this study is to provide information and data on the corrosion behavior of Zircaloy that can be used to determine the long-term durability of nuclear fuel cladding made of this material. Zircaloy obtains its corrosion resistance by the formation of a protective oxide surface film. All of the work in this task relates to the formation or breakdown of this film. This experimental work involves electrochemical measurements made primarily using potentiostatic polarization techniques to study the corrosion behavior of bulk Zircaloy-2 and -4 as well as specimens of cladding tubes made from these two materials. Measurements are made on both the inner and outer walls of the cladding, and most measurements have been made at 95°C. Measurements have been made in simulated J-13 water, a water that may be typical of that which could be present in the Yucca Mountain, Nevada site.

There is some concern regarding the effects of fluoride on the zirconium oxide film. Work of this reporting period has dealt with preparations for studying the effects of the fluoride ion on the formation and stability of the oxide film on zirconium alloys. Initially, this work will be conducted at 23°C for the purpose of acquiring reproducible data to determine effects of the fluoride ion on zirconium oxide. These data can also to be used in analyzing measurements taken at higher temperatures where experimental procedures are more difficult. Zircaloy-2, Zircaloy-4 cladding specimens (outside wall of the tubing exposed) have been prepared and immersed in the 0.1 M NaF solution. Some time/electrode potential, cyclic polarization and impedance tests have been conducted. These specimens will remain in the test solutions for an indefinite time. Another test that is in progress is Zircaloy-2 cladding (outside wall) that was exposed for seven months at 70°C after being exposed at 95°C. Work for the next reporting period will include data analysis, additional tests on the exposed specimens and preparation of specimens for additional tests.

SDI103, UD 9003, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-90/MAR(ISS03)

Sets selected:

Set	Items	Description
1	22	WASTE(W)PACKAGE?
2	37	CANISTER?
3	334	CORROSION (1974 DEC)
4	45	LEACHING (1974 DEC)
5	170	GLASS (1974 DEC)
6	17	VITRIFICATION (1974 DEC)
7	541	S3-S6/OR
8	30	HIGH(W)LEVEL(W)WASTE?
9	391	RADIOACTIVE(W)WASTE?
10	116	NUCLEAR(W)WASTE?
11	15	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

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

Date	Time	Description
16feb	13:50EST	PR 11/5/1-25 (items 1-15)

Total items to be printed: 15

SDI006, UD 9006, SER. DD016

File(s) searched:

File 6:NTIS - 64-90/ISSUE06
(COPR. 1990 NTIS)

Sets selected:

Set	Items	Description
1	0	WASTE(W)PACKAGE?
2	4	CANISTER?
3	25	CORROSION
4	13	LEACHING
5	63	GLASS
6	3	VITRIFICATION
7	100	S3-S6/OR
8	4	HIGH(W)LEVEL(W)WASTE?
9	46	RADIOACTIVE(W)WASTE?
10	11	NUCLEAR(W)WASTE?
11	3	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

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

Date	Time	Description
18feb	16:16EST	PR 11/5/1-25 (items 1-3)

Total items to be printed: 3

SDI293, UD 9003, SER. DD023

File(s) searched:

File 293:ENGINEERED MATERIALS ABS 86-90/MAR
(COPR. 1990 ASM INTERNATIONAL)

Sets selected:

Set	Items	Description
1	0	HIGH()LEVEL()WASTE? ? OR RADIOACTIVE()WASTE? OR NUCLEAR()WASTE?
2	312	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
23feb	20:27EST	PR 3/5/1-25 (no items to PRINT)

6 **Total items to be printed:** 0

SDI006, UD 9007, SER. DD016

File(s) searched:

File 6:NTIS - 64-90/ISSUE07
(COPR. 1990 NTIS)

Sets selected:

Set	Items	Description
1	1	WASTE(W)PACKAGE?
2	9	CANISTER?
3	30	CORROSION
4	12	LEACHING
5	71	GLASS
6	2	VITRIFICATION
7	111	S3-S6/OR
8	4	HIGH(W)LEVEL(W)WASTE?
9	59	RADIOACTIVE(W)WASTE?
10	8	NUCLEAR(W)WASTE?
11	5	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

10

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

Date	Time	Description
02mar	06:18EST	PR 11/5/1-25 (items 1-5)

Total items to be printed: 5

SDI103, UD 9004, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-90/MAR(ISS04)

Sets selected:

Set	Items	Description
1	18	WASTE(W)PACKAGE?
2	15	CANISTER?
3	134	CORROSION (1974 DEC)
4	32	LEACHING (1974 DEC)
5	115	GLASS (1974 DEC)
6	8	VITRIFICATION (1974 DEC)
7	280	S3-S6/OR
8	14	HIGH(W)LEVEL(W)WASTE?
9	313	RADIOACTIVE(W)WASTE?
10	45	NUCLEAR(W)WASTE?
11	8	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

11

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

Date	Time	Description
04mar	22:57EST	PR 11/5/1-25 (items 1-8)

Total items to be printed: 8

SDI008, UD 9003, SER. DA016

File(s) searched:

File 8:COMPENDEX PLUS - 70-90/MAR Copr. Engineering Info
Inc. 1990)

Sets selected:

Set	Items	Description
1	7	WASTE(W)PACKAGE?
2	4	CANISTER?
3	172	CORROSION
4	43	LEACHING
5	433	GLASS
6	11	VITRIFICATION
7	638	S3-S6/OR
8	6	HIGH(W)LEVEL(W)WASTE?
9	122	RADIOACTIVE(W)WASTE?
10	13	NUCLEAR(W)WASTE?
11	1	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

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

Date	Time	Description
10mar	20:08EST	PR 11/5/1-20 (items 1-1)

Total items to be printed: 1

SDI103, UD 9005, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-90/APR(ISS05)

Sets selected:

Set	Items	Description
1	17	WASTE(W)PACKAGE?
2	12	CANISTER?
3	208	CORROSION (1974 DEC)
4	86	LEACHING (1974 DEC)
5	131	GLASS (1974 DEC)
6	15	VITRIFICATION (1974 DEC)
7	391	S3-S6/OR
8	21	HIGH(W)LEVEL(W)WASTE?
9	301	RADIOACTIVE(W)WASTE?
10	154	NUCLEAR(W)WASTE?
11	14	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

13

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

Date	Time	Description
11mar	22:00EST	PR 11/5/1-25 (items 1-14)

Total items to be printed: 14

SDI032, UD 9004, SER. DD022

File(s) searched:

File 32:METADEX 66-90/APR
(COPR. 1990 ASM INTERNATIONAL)

Sets selected:

Set	Items	Description
1	3	HIGH()LEVEL()WASTE? ? OR RADIOACTIVE()WASTE? OR NUCLEAR()WASTE?
2	1644	STEEL? ? OR ZIRCALOY? ? OR TITANIUM? ? OR COPPER
3	2	1*2
4	0	ANNA FRAKER, 223, B-254, X6009

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

Date	Time	Description
21mar	06:47EST	PR 3/5/1-25 (items 1-2)

Total items to be printed: 2

14

SDI006, UD 9008, SER. DD016

File(s) searched:

File 6:NTIS - 64-90/ISSUE08
(COPR. 1990 NTIS)

Sets selected:

Set	Items	Description
1	0	WASTE(W)PACKAGE?
2	6	CANISTER?
3	64	CORROSION
4	21	LEACHING
5	137	GLASS
6	10	VITRIFICATION
7	213	S3-S6/OR
8	6	HIGH(W)LEVEL(W)WASTE?
9	125	RADIOACTIVE(W)WASTE?
10	17	NUCLEAR(W)WASTE?
11	1	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

15

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

Date	Time	Description
22mar	18:24EST	PR 11/5/1-25 (items 1-1)

Total items to be printed: 1

SDI103, UD 9006, SER. DD017

File(s) searched:

File 103:DOE ENERGY - 83-90/MAY(ISS06)

Sets selected:

Set	Items	Description
1	3	WASTE(W)PACKAGE?
2	8	CANISTER?
3	152	CORROSION (1974 DEC)
4	46	LEACHING (1974 DEC)
5	97	GLASS (1974 DEC)
6	13	VITRIFICATION (1974 DEC)
7	287	S3-S6/OR
8	26	HIGH(W)LEVEL(W)WASTE?
9	260	RADIOACTIVE(W)WASTE?
10	38	NUCLEAR(W)WASTE?
11	5	(S1 OR S2) AND S7 AND (S8 OR S9 OR S10)
12	0	ANNA FRAKER RM. B-106 BLDG. 223 X6009
13	0	JILL RUSPI

16

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

Date	Time	Description
30mar	20:45EST	PR 11/5/1-25 (items 1-5)

Total items to be printed: 5

NIST Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the
U.S. Department of Energy
Contract No. W-7405-Eng-48.

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

Nitao, J.
"Simulations of the Near-Field Transport of Radionuclides by Liquid Diffusion at Yucca Mountain -- Comparisons With and Without Emplacement Backfill"
UCID-21466, July 1988.

DATE REVIEWED: 2/14/90

PURPOSE

"We consider in this report simplified simulations of one-dimensional transport of radionuclides in the rock due to liquid molecular diffusion in order to determine the effects of an emplacement backfill."
"The radionuclide species that are considered are I-129, Sr-240, Pu-240 and U-234."

KEY WORDS

Theory, computer simulation of diffusion, one-dimensional molecular diffusion, simulated field, Yucca Mountain, tuff, backfill, ^{90}Sr, ^{240}Pu, ^{234}U, ^{129}I, ambient temperature, molecular diffusion, bentonite, ^{234}U, ^{90}Sr, groundwater

CONTENTS

24 pages, 13 figures, 1 table
The text includes 2 pages of introduction,
1 page which describes the mathematical model,
2 pages which describe how the model is solved numerically,
3 pages of applications and a 2 page conclusion.

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

1. "In this report we perform some preliminary calculations on the release rates for certain radionuclides based on a simplified model of diffusive transport in the rock. Convective transport is neglected..."

2. "Due to the lack of experimental data we have used a value of one for the tortuosity factor which is a conservative value from the standpoint of overestimating diffusive release rates. But in the future, simulations using lower values of the tortuosity factor should be made, in which case, convective transport will become important relative to diffusion."

3. "The effect of spatial variations in tortuosity and sorption in the rock have not been considered."

4. "The simplified mathematical models used in our simulations are based on solving the one-dimensional diffusion-decay equation in spherical coordinates for molecular diffusion, sorption, and radioactive decay of a single aqueous specie... More sophisticated mathematical models may have to be developed in the future to take into account the complex hydrological and geochemical processes that may exist in the field."

5. "Our simulations of the no-backfill case are pertinent only to the, perhaps unlikely, scenario that a contiguous bridge of liquid water is sustained from the waste to the tuff. For the backfill case, it is assumed that such a bridge exists from the waste form to the backfill... our simulations and the resulting release rates do not take account the lower probability of a contiguous diffusion path occurring in the no-backfill case..."

CONCLUSIONS OF AUTHOR

1. "As expected, isotopes such as those of iodine, that have long half-lives and are not significantly sorbed by the tuff or Bentonite backfill, will show little difference in release rates between the backfill and no-backfill cases."

2. "Some isotopes such as Pu-240 have their release rates significantly reduced by a Bentonite backfill."

3. "Some isotopes such as U-234 will have an initial reduction in the release rate due to the backfill but will settle down to a steady state rate that is about the same as for the case without a backfill.

4. "For isotopes that are more sorbed by the backfill than by the tuff, the presence of a backfill against a container will significantly increase the rate of transport out of the container as compared to the case where the tuff is against the opening. However, subsequent diffusive transport from the backfill to the tuff is less or about the same in some cases. Steady-state concentration profiles in the tuff for species that escape the backfill barrier before decaying are in some cases significantly lower

with a backfill and in other cases are higher but not by more than 20 percent in the cases considered.

COMMENT OF REVIEWER

Because of all the simplifications made concerning the hydrology and geochemistry, the results of the model should be viewed as preliminary. The hydrological and geochemical environment for a real waste form at Yucca Mountain will be more complex than assumed by Nitao. Hence, it is not possible to determine how well the model describes realistic situations.

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

(a) Relationship to Waste Package Performance Issues Already Identified

2.3.5 How will packing, container materials (including overpacks, canisters, and any special corrosion-resistant alloys or spent fuel rod cladding, if applicable) and/or their alteration products interact with the waste form to cause its alteration and/or effect release of radionuclides?

(b) New Licensing Issues

(c) General Comments on Licensing

NIST Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the
U.S. Department of Energy
Contract No. W-7405-48 Eng.

(b) Author(s),

Nitao, J. J.
"Numerical Modeling of the Thermal and Hydrological Environment Around a Nuclear Waste Package Using the Equivalent Continuum Approximation: Horizontal Emplacement"
UCID-21444, May 1988

DATE REVIEWED: 1/31/90.

PURPOSE

The author has "performed computer simulations of the immediate thermal and hydrological environment around a nuclear waste package". "Fluid behavior in the rock was modeled using the equivalent continuum approximation." A computer code called TOUGH (Pruess, 1985) was modified to model the flow of water, water vapor and air through the rock.

KEY WORDS

Theory, modeling, computer modeling, TOUGH, simulated field, Yucca Mountain, air, water vapor, aerated water, high-level, heat capacity, thermal conductivity, hydrology.

CONTENTS

There are 72 pages of text, 59 figures and 3 tables. In the text, there are: 2 pages of introduction, 14 pages which describe the model assumptions, 2 pages concerning the computer code used, 7 pages which describe model results for different input parameters, 2 pages of concluding remarks and 3 pages of references.

UNCERTAINTIES IN DATA

The author's work is theoretical; there is no experimental data to validate the equivalent continuum approximation for flow. Further, there is no data to verify the values of the input parameters used in the model.

DEFICIENCIES/LIMITATIONS IN DATABASE

Because of so many simplifying assumptions about the rock matrix, fractures and flow, how well the model describes reality is an open question. The author states that, "considerable work needs to be done to develop better models of fluid behavior in partially fractured rock and more experimental data is needed to accurately model the waste package environment." The author remarks that "our results should be viewed as preliminary at this time because many of the physical input parameters to the model as well as the exact conditions in Yucca Mountain are either not known or have a high degree of uncertainty and many idealizations and approximations of complicated physical and chemical behavior have been made. "The author recommended research on the following issues."

1. The accuracy and applicability of the equivalent continuum model...

2. ...comparisons need to be made with heat conduction models that use variable specific heats... and the effect of using periodic boundary conditions needs to be considered.

3. The effect on container spacings and the placement of containers with different heat outputs...

4. Experimental data for the matrix and fracture characteristic curves at different temperatures, and under both drainage and imbibition conditions.

5. Discrete fracture modelling is needed to study the effect of gravity flow in fractures, fracture orientation, fracture cross flow, and highly permeable rubbilized shear zones.

6. The effect of thermally stressed fracture closings on the near field hydrology is not known.

7. The possible effect of film flow ... should be investigated...

8. Depending on suction potential curves, there may be significant amounts of surface absorbed water in matrix rock... the diffusive transport properties of this water may have to be determined experimentally."

Further, the author states that, "chemical effects, such as possible increased concentration of silica by dissolution in hot water moving toward waste package via "heat pipes" and deposition where water evaporates, are not taken into account."

CONCLUSIONS OF THE AUTHOR

The author notes that "the predictions reported in this paper are highly preliminary due to the current uncertainty in the model input parameters and the sub-models used in our simulations." Given this caveat, the conclusions are:

1. At the surface recharge rates studied and during the 2600 year time span of our simulations the liquid saturation at the borehole wall next to the container did not return to the initial native saturation after drying from the waste package waste heat began. Liquid pore

pressures remain below the borehole pressure so that no liquid flow into the borehole is predicted.

2. Small amounts of water due to capillary condensation first reach the borehole wall at about 200 years. This time is very sensitive to the capillary curves used at the low saturation range. Based upon the conservative recharge rate estimate of 0.5 mm/year [Montazer and Wilson, 1984], bulk water first reaches the borehole wall at about 1000 years from emplacement due primarily to capillary imbibition.

3. Predictions of temperatures near the waste package are significantly lower when flow is included as compared to the case when only thermal conduction and constant specific heat are considered.

4. Significant amounts of fracture water are predicted to occur due to the movement and condensation of water vapor in the fractures during the heating period. However, there is no water in the fractures in the vicinity of the waste package due the drying effects of the waste package heat."

COMMENTS OF REVIEWER

The author openly admits that his results depend on a model which greatly simplifies the physics and chemistry that govern the thermal and hydrological environment around the waste canister. The greatest weakness of the work is the lack of experimental data. For instance, the author assumes an equivalent continuum medium approximation for modeling flow through both the fractures and rock matrix. However, no data on fractures is presented. Without experimental data, it is not possible to verify the equivalent medium approximation. Rather than an equivalent medium continuum approximation, it may be necessary to solve for flow through discrete fractures. Thus, as the author recognizes, his conclusions depend on many simplifying assumptions that may not be valid.

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

(a) Relationship to Waste Package Performance Issues Already Identified

2.1 When, how, and at what rate will groundwater penetrate the packing around waste package containers.

(b) New Licensing Issues

(c) General Comments

NIST Review of Technical Reports on the High Level Waste Package for
for Nuclear Waste Storae

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the
U.S. Department of Energy

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

Kass, J. N.
"Evaluation of Copper, Aluminum Bronze, and Copper-Nickel for YMP Container Material"
UCRL-101097, May 1989

PURPOSES

"I will discuss our evaluation of the materials copper, 7% aluminum bronze, and 70/30 copper-nickel. These are three of the six materials currently under consideration as potential waste-package materials." "We are also considering alternatives to these six materials. This work is part of the Yucca Mountain Project (YMP), formerly known as the Nevada Nuclear Waste Storage Investigations (NNWSI) Project."

KEY WORDS

Experimental data, general corrosion, laboratory, Yucca Mountain, air, tuff composition, Cl, gamma radiation, ambient pressure, ambient temperature, lithostatic pressure, neutral solution, static (no flow), copper base, groundwater, stress or strain, corrosion (general), corrosion,(intergranular), corrosion (local), cracking, fracture, hydrogen embrittlement.

CONTENTS

This 33-page report consists of 8 tables and 9 figures and is written narratively without subtopics. Eight references were quoted.

TEST CONDITIONS

Materials: copper (CDA102), aluminum bronze (CDA613), and 70/30 copper-nickel (CDA715)

Specimen Preparation:

- waste package size: ~2 feet in diameter and ~14 feet high
- package: PWR bundles, BWR bundles or mixtures
- thickness: ~1 cm (aluminum bronze, copper-nickel)
  ~3 to 4 cm (pure copper)

Environment:

- repository: 700 feet above the water table and 300 to 1200 feet below the surface of the mountain
- temperature: Nominal - 250 to 97°C (some packages 250°C to <97°C)
- water composition and infiltration rate: 5 liters/year per container, neutral pH, ~10 ppm Cl-, NO_3-, O_2, ~20 ppm SO_4^{2-}, ~120 ppm HCO_3^-
- gamma flux: 10^4 to $<10^2$ rad/h, radiolysis products - Hydrogen peroxide and nitrogen-bearing species
- lithostatic and hydrostatic pressure: normally none
- Infiltration water rate: 5 liters/year per container

UNCERTAINTIES IN DATA

- approximated water chemistry
- approximated waste package size

DEFICIENCIES/LIMITATIONS IN DATABASE

- "A miscibility gap has been postulated (CDA715), but no phase separation has actually been reported." "Our research will have to justify that phase separation, indeed, does not occur."
- "The gamma fluxes used in these tests were about 10 times as high as expected in an actual repository."
- "These results (SCC for Cu/Al and Cu/Ni) are from constant-load tests."
- "The use of U-bend specimens makes the test very mild since the stresses will relax with time."
- We remain concerned about the possibility of alloy segregation with the Al bronzes."
- "Copper-based materials are not used in irradiated environments."
- "The performance of copper-based materials has not been established under repository-relevant conditions of temperature and oxidizing environment and gamma irradiation. This is particularly important to the early containment period."
- "High purity copper is very soft. Hence, we would need increased wall thicknesses for this material."
- "We need to determine the cause of ductility minimum around 250°C to 300°C."
- "We have to show that the fuel-cladding temperature will not exceed 350°C."
- "We need to guarantee that we can make the closure weld."
- "There are many inspection techniques available for the austenitic alloys, but not for the copper alloys."
- "We need to investigate the embrittlement of the weld metal over long periods of time."
- "We need to prepare weld materials samples and heat-treat them in various ways to simulate these long exposure periods."
- "We need to evaluate the possibility of alloy separation."

- "We need to know whether these threshold levels (ammonia for SCC) are lower in the presence of other species."
- "It is difficult to be convinced that what doesn't happen in a short test will not happen over a few hundred years."
- "There will be cost limitations."
- "We have not performed the key experiments yet" (copper-based materials).

CONCLUSIONS OF AUTHOR

- "Since the exposure environment is relatively benign, both general and localized corrosion performance will almost certainly be acceptable."
- "I am similarly optimistic about the mechanical properties," but "we need to determine the cause of the ductility minimum and how it will be influenced over many years at our temperatures."
- "Thermal problems are also very unlikely."
- "We do have fabrication techniques developed," "but we still need to guarantee that we can make the closure weld."
- "We haven't done much in the area of inspection."
- "We must show that the metallurgical structures are stable for 300 years at temperatures in the range of 250°C to 100°C."
- "The ability of NH_3 to cause SCC is a further concern."

COMMENTS OF REVIEWER

1. The merit of this report is to evaluate the candidate container materials under newly identified Yucca Mountain Environments.

2. The temperatures are greater than 97°C to 250°C, which implies that aqueous environments are not likely to exist. Further, water infiltration rates, 5 liter/year (14 cm^3/day) per container will be ~1 cm thickness per day for 2x14' drum assuming no evaporation or no reaction. The reviewers assume infiltration rates are not conventional flow rates involving a velocity with water movement over the whole area whether flooded or not. Under these two circumstances, aqueous corrosion are unlikely to be a problem for the safe performance of containers. Then, the question is why the candidates (especially copper) are primary materials which were selected mainly due to good corrosion performance based on the Swedish work under anoxic conditions. This is particularly true for copper. In addition to the listed disadvantages in table 8, (1) the environments are oxic as stated in the content, and (2) there are also statements of author's concern of embrittlement or phase changes (especially for the weld), which were again supposed to be no problem. Therefore, the author's plan to re-evaluate alternatives will be very valuable.

3. The temperatures are likely to be above the boiling point of water. Therefore, the evaluation criteria of table 2 should be revised. For instance, oxidation or steam corrosion can be included as a primary criterion.

4. Porosity forms in the metal welds and can be compared with that of ceramics. The microstructure of voids should also be addressed. When the percentage of porosity is very low, it is very unlikely that the microstructure has continuous void channels.

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

(a) Relationship to Waste Package Performance Issues Already Identified

2.2.4 what are the potential corrosion failure modes for the waste package containers?

(b) New Licensing Issues

(c) Comments Related to Licensing

NIST Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the U.S. Department of Energy

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

O'Neal, W. C., Ballou, L. B., Gregg, D. W., and Russell, E. W.
"Nuclear Waste Package Design for the Vadose Zone in Tuff"
UCRL-89830, February 1984

DATE REVIEWED:

PURPOSE

"The objective is to develop and analyze waste package designs which incorporate qualified materials and which are fully compatible with the repository design." "The basic design philosophy for waste package conceptual designs is to meet NRC design criteria with flexibility in technical performance and cost."

KEY WORDS

Design, simulated field, nickel base, stainless steel, 304L stainless steel, 316L stainless steel.

CONTENTS

15 pages, 2 figures (container designs), and 2 tables.

AMOUNT OF DATA

Some data on costs of various package designs.

TEST CONDITIONS

None given.

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

CONCLUSIONS OF AUTHOR

"The conceptual design ensemble provides a group of designs from which a set can be chosen for eventual detailed design of waste packages. Within the variations in materials and configurations available, detailed designs should meet all requirements, barring unexpected results from long-term materials testing or packaging environment characterization activities."

COMMENTS OF REVIEWER

This report is a good, but very brief, overview of the process used in arriving at the present canister design for containment of high level nuclear waste, and includes a brief description of the various mechanical tests performed on the container, simulating possible scenarios in the repository. An economic analysis compares the cost of vertically emplaced canisters with and without packing.

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

(a) Relationship to Waste Package Performance Issues Already Identified

2.3.7 How will the design of the waste form accommodate all potential natural and waste package induced conditions?

(b) New Licensing Issues

(c) General Comments

AUTHOR'S ABSTRACT

This report presents an overview of the selection and analysis of conceptual waste package designs that will be used by the Nevada Nuclear Waste Storage Investigation (NNWSI) project for disposal of high level nuclear waste (HLW) at the proposed Yucca Mountain, Nevada Site.

The design requirements that the waste packages are required to meet are listed. Concept drawings for the reference designs and one alternative package design are shown. Four metal alloys; 304L SS, 321 SS, 316L SS and Incoloy 825 have been selected for candidate canister/overpack materials, and 1020 carbon steel has been selected as the reference metal for the borehole liners.

A summary of the results of technical and economic analysis supporting the selection of the conceptual waste package designs is included. Post-closure containment and release rates are not discussed in this paper.

NIST Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the
U.S. Department of Energy

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

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

DATE REVIEWED: 6/27/89

PURPOSE

"The purpose of this paper is to document the history of the part of the Metal Barrier Selection and Testing Task that concerns selection of candidate materials for the waste package container."

KEY WORDS

Design, literature review, Yucca Mountain, tuff composition.

CONTENTS

Text - 17 pgs., 24 references, 7 Attachments.

AMOUNT OF DATA

None given.

TEST CONDITIONS

None given.

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

CONCLUSIONS OF AUTHOR

None given.

COMMENTS OF REVIEWER

This is an excellent historical review of the events leading to selection of canister materials from 1981 to 1988 by an individual that has been intimately involved with these events at Lawrence Livermore National Laboratories since before 1981. The events are well documented by an extensive list of references and pertinent letters.

The author's sites were at Hanford, WA, Deaf Smith, TX, and Yucca Mountain, AZ. The review includes the changing container requirements as the site selection was narrowed to Yucca Mountain, and the horizon was changed from the saturated zone to the unsaturated zone. At the early stages of the selection process, only iron-based materials were chosen from consideration for the container, but in 1984, a high nickel alloy and three copper-based metals were included.

RELATED HLW REPORTS

1) Halsey, H.G. and McCright, R. D., "Metal Barrier Selection and Testing Task Scientific Investigation Plan," UCID-21262, November 1987.

2) Conceptual Waste Package Designs for Disposal of Nuclear Waste in Tuff," Westinghouse Electric Corporation Report AESD-Tme-3138, September 1982.

3) McCright, R. D., Van Konynenburg, R. A., and Ballou, L. B., "Corrosion Test Plan to Guide Canister Materials Selection and Design for a Tuff Repository, UCRL-89476, November 1983.

4) McCright, R. D., Weiss, H., Juhas, M. and Logan, R. W., "Selection of Candidate of Canister Materials for High-Level Nuclear Waste Containment in a Tuff Repository," UCRL-89988, April 1984.

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

(a) Relationship to Waste Package Performance Issues Already Identified

Does not apply.

(b) New Licensing Issues

(c) General Comments

AUTHOR'S ABSTRACT

This paper documents events in the Nevada Nuclear Storage Investigations (NNWSI) Project that have influenced the selection of metals and alloys proposed for fabrication of waste package containers for permanent disposal of high-level nuclear waste in a repository at Yucca Mountain, Nevada. The time period from 1981 to 1988 is covered in this annotated history. The history traces the candidate materials that have been

considered at different stages of site characterization planning activities. A present, six candidate materials are considered and described in the 1988 Consultation Draft of the NNWSI Site Characterization Plan (SCP). The six materials are grouped into two families, copper-base materials and iron to nickel-based materials with an austenitic structure. The three austenitic candidates resulted from a 1983 survey of a longer list of candidate materials; the other three candidates resulted from a special request from DOE in 1984 to evaluate copper and copper-base alloys.

NIST Review of Technical Reports on the High Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the
U.S. Department of Energy

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

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

DATE REVIEWED: 2/26/90

PURPOSE

To explain the general concept of "performance allocation" as it applies to the Yucca Mountain waste package.

KEY WORDS

Design, simulated field, Yucca Mountain, tuff, gamma radiation field, copper base, stainless steel, commercial high-level waste (CHLW), defense high-level waste (DHLW).

CONTENTS

13 pages of text, 1 table of Engineered Barrier System elements, 7 references.

AMOUNT OF DATA

None given.

TEST CONDITIONS

None given.

UNCERTAINTIES IN DATA

None given.

DEFICIENCIES/LIMITATIONS IN DATABASE

None given.

CONCLUSIONS OF AUTHOR

To meet the post-closure objectives for the waste disposal system established by the NRC in 10 CFR 60, computer modeling is required, and performance assessment is and will remain a key element of the waste package program through licensing.

COMMENTS OF REVIEWER

This report is a good, but very brief, description of the "performance allocation" process, whereby elements of the Engineered Barrier System are evaluated on their contribution to containment, credit, of the nuclear waste.

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

(a) Relationship to Waste Package Performance Issues Already Identified

2.2.8 How will the design of the waste package container accommodate all potential natural and waste package-induced conditions?

(b) New Licensing Issues

(c) General Comments

AUTHOR'S ABSTRACT

The U.S. Nuclear Regulatory Commission has established two post-closure performance objectives for the Engineered Barrier System (EBS) in a geologic repository. These require containment of the waste followed by controlled release. The EBS for a repository in unsaturated tuff at Yucca Mountain is designed to meet these performance objectives. The major components are the waste form, container, air gap, and borehole liner. Assessment of post-closure performance of the EBS is based on allocating performance for various components toward meeting overall design objectives. Because of the unprecedented time periods considered, 1000 to 10,000 years, computer modeling is essential and will be used in conjunction with testing to assess whether the performance allocations are met.

NIST Review of Technical Reports on the High-Level Waste Package for Nuclear Waste Storage

DATA SOURCE

(a) Organization Producing Data

Lawrence Livermore National Laboratory for the U.S. Department of Energy
Contract No. W-7048-Eng-48.

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

Nelson, T, Russell, E., Johnson, G. L., Morisette, R., Stahl, D., LaMonica, L., Hertel, G.
"Yucca Mountain Project Waste Design for MRS System Studies"
UCID-21700, April 1989

DATE REVIEWED: 2-29-90

PURPOSE

"This report, prepared by the Yucca Mountain Project, is the report for Task E. 'Waste Package Design' of the MRS (Monitored Retrievable Storage) system study." "Task E. Waste Package Design, defined a scope of work that involves developing information on current and alternative waste package designs, developing a canister usable at the MRS for either intact or consolidated fuel, and providing a discussion of the potential for developing a 'heat-tailored' repository."

The primary aim of the MRS system studies is to develop cost and schedule information relative to the application of current designs and technology."

"This section (5.0 subtask E-2) is intended to provide a summary description of waste package concepts being investigated under the Alternate Barriers Task. The objective of this task is twofold. First, the task is to meet the requirements of 10 CFR Part 60 for consideration and comparison of alternates. Second, the task is to select one or two alternates which can be carried through ACD (Advanced Conceptual Design). The latter is to provide a fall-back position in the event that results from site characterization render the reference metal container unacceptable."

"The objective (7.0 Subtask E-4) is to enhance the unsaturated nature of the emplacement horizon such that the mean time for liquid water reaching the waste-emplacement borehole can be maximized and the presence of liquid water on the waste container can be minimized."

KEY WORDS

Literature review, scoping test, field, Yucca Mountain, tuff composition, tuff, bronze, copper base, nickel base, stainless steel, titanium base, iron-base, aluminum, lead, zinc, 304L stainless steel, 316L stainless steel, Alloy 825, CDA 613, CDA 952, CDA 715, CDA 102, CDA 122, intact assemblies of fuel rods, glass (West Valley reference glass), glass (defense waste reference glass), commercial high-level waste (CHLW), defense high-level waste (DHLW), spent fuel, spent fuel (BWR), spent fuel (LWR), spent fuel (PWR),

CONTENTS

This 140-page report consists of an introduction and summary, 14 figures, 10 tables, and the following content:

TEST CONDITIONS

Materials:

- light water reactor (LWR) spent fuel: either pressurized water reactor (PWR) or boiling water reactor (BWR) spent fuel.
- defense high-level waste (DHLW).
- commercial high-level waste (CHLW).
- borosilicate glass inside a stainless steel canister for CHLW.
- six alloys as candidates for container materials: 304L stainless steel, 316L stainless steel, alloy 825, Aluminum Bronze (CDA 613 or CDA 952), Copper-Nickel Alloy (CDA 715), and High Purity Copper (CDA 102 and CDA 122).
- alternative container materials: single metals (titanium alloys, high nickel alloys), bi-metallics (nickel and iron-base alloys and copper alloys for the outer and inner liners respectively), coatings (aluminides, nickel-chrome-aluminum alloys, oxides, nitrides), fillers (magnetites, glass, aluminum, lead and zinc), alumina, and graphite.

Specimen Preparation:

- Intact assemblies of fuel rods or consolidated fuel rods. The consolidated fuel rods may be contained in a canister. Ninety-five percent of the spent fuel would be consolidated. (Combinations are in Tables 2-3, 4-1, and 4-2.)

- A fixed spent fuel loading in each waste package type with a maximum thermal output at the time of emplacement of 3.5 KW/package. -Table 2-2. Waste Form Assumptions. Essentially a right cylinder of 310-480 cm height, 50-75 cm diameter, and 1.0-2.5 cm thickness, with a lid containing a handling pintler (Table 4-2).
- Fabrication and welding for metal containers: back extrusion, roll-and-welding, centrifugal casting, deep drawing, electron beam welding, plasma arc welding, and friction welding.
- Alumina containers: similar sizes to those of metal containers: an alumina shell and a thin stainless steel jacket (Table 5-1). Hot isostatic pressing and active metal brazing are used for processing.
- NDE: ultrasonic inspection, radiography and dye penetrating testing.
- Canister: 304L stainless steels (or alloy 825 if not closed well) 6~9 in. side and 175 in. long of square cross sections of 55 gallon.

Environment:

- Yucca Mountain Repository, with its unsaturated "dry" environment.
- Dry spent fuel.
- A maximum fuel cladding temperature limit of 350°C.
- The nominal thermal output at emplacement is 3300 W per package for the spent fuel and 470 W per package for the high-level waste. APD (The Areal Power Density) limit is 57 KW/acre.
- Model-calculated temperatures:
  Model 1 - borehole wall (56~113°C), interior of panel (>97°C), panel center (25-50°C).
  Model 2 - borehole wall (22°C), at below 97°C (15% by not heat-tailoring, 7.5% by heat-tailoring).

UNCERTAINTIES IN DATA

LLNL MRS canister design (tolerance limits in figures 10 and 14).

The metal containers will have the wall thickness of <u>about</u> 1 cm, except for high purity copper, where a <u>2.5-3 cm</u> thickness is required, and will be of a right cylinder of <u>310-480 cm</u> height, <u>50-75 cm</u> diameter, and a <u>1.0-2.5 cm</u> thickness.

DEFICIENCIES/LIMITATIONS IN DATABASE

"In order to perform a system study such as the MRS system study, it is necessary that a number of assumptions be made to fix some of the variables still under consideration. These assumptions may not represent an optimum set, however, and it should not be assumed that they represent a preferred set. They should be viewed simply for what they are: a set of reference conditions for the purpose of the MRS system study." "These assumptions do not represent the total variations anticipated in the variables listed."

"No one concept would work for all scenarios because of the different functions being performed by the MRS and the repository."

CONCLUSIONS OF AUTHOR

"A number of assumptions were necessary prior to initiation of this system study." "Existing concepts were utilized because of schedule constraints."

"With the exception of rod consolidation considerations, the system study should not be sensitive to the parameters assumed for the waste package."

"Although stainless steel is assumed for this study, a container material has not been selected for Advanced Conceptual Design from the six candidates currently under study."

"The alternate waste packages and materials" are "being considered in the event that the waste package emplacement environment is more severe than is currently anticipated."

"A concept is provided "for an MRS canister to contain consolidated fuel for storage at the MRS and eventual shipment to the repository."

"Heat-tailoring is not necessary to achieve the required performance of the waste package. Although no performance has been allocated to the potential benefits of 'heat tailoring' because of the many uncertainties involved, this benefit is being considered in the design of the repository."

COMMENTS OF REVIEWER

The content of this report is based heavily on many assumptions made for Monitored Retrievable Storage (MRS). However, the list of those assumptions does not seem to be complete. The cost of rod consolidation consideration in the sensitivity analysis is not mentioned. Such assumptions should be justified by tabulating existing data from many on-going projects such as vitrification data in the West Valley demonstration project, or from seismic behavior.

The site for the MRS is not clearly indicated and may be confused with that for the permanent storage. Where are the candidate sites for the MRS? Also, it is not clear whether retrievability procedures for the MRS can be used for the permanent storage as well.

The candidate container materials are based heavily on the corrosion performance of those materials. The selection of the candidate materials should include consideration of other factors such as fabricability, nondestructive evaluation (NDE), and cost. Especially when the site is heat-tailored, there may be no aqueous environments necessary for the corrosion of candidate materials. In this sense, alternative container materials appear to be very important. Therefore, the description of an alumina container is very useful and such detailed descriptions for other alternative materials also would be helpful.

The importance of the canister was described quite precisely. It is not clear whether the canister itself is an effective barrier for the NRC total containment criteria of radionuclides release.

It would be helpful to have the guidelines of heat-tailoring explained in a more illustrative manner. Also, it is not clear whether or not this heat-tailoring is applicable to both the MRS and the permanent storage.

Other minor comments are:

(1) The combinations in "Table 2-1. MRS System Packaging Scenarios" does not seem to have all possible combinations. Is this justified by the general assumptions made by the authors?
(2) On page 1-1, the meaning is not clear in the conclusion, "Heat tailoring restricts specific fuel shipment from reactors."
(3) The reason for using thick copper is stated to obtain a proper mechanical strength. Another reason for using copper is for the corrosion resistance.

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

(a) Relationship to Waste Package Performance Issues Already Identified

2.2.8 How will the design of the waste package container accommodate all potential natural and waste package induced conditions?

(b) New Licensing Issues

(c) General Comments on Licensing