OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ANALYSIS/MODEL COVER SHEET
Complete Only Applicable Items

1. QA: QA

Page: 1 of: 82

2. ☒ Analysis — Check all that apply

Type of Analysis
☐ Engineering
☒ Performance Assessment
☐ Scientific

Intended Use of Analysis
☐ Input to Calculation
☐ Input to another Analysis or Model
☒ Input to Technical Document
☐ Input to other Technical Products

Describe use:

Describes inclusions and exclusions

of FEPs in TSPA-SR

3. ☐ Model — Check all that apply

Type of Model
☐ Conceptual Model
☐ Abstraction Model
☐ Mathematical Model
☐ System Model
☐ Process Model

Intended Use of Model
☐ Input to Calculation
☐ Input to another Model or Analysis
☐ Input to Technical Document
☐ Input to other Technical Products

Describe use:

4. Title:

Clad Degradation – FEPs Screening Arguments

5. Document Identifier (including Rev. No. and Change No., if applicable):

ANL-WIS-MD-000008 REV 00 ICN 01

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Enclosure 2

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ANALYSIS/MODEL REVISION RECORD
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1. Page: 2 of: 82

2. Analysis or Model Title:
Clad Degradation – FEPs Screening Arguments

3. Document Identifier (including Rev. No. and Change No., if applicable):
ANL-WIS-MD-000008 REV 00 ICN 01

4. Revision/Change No.	5. Description of Revision/Change
REV 00	Initial Issue
REV 00 ICN 01	Changed corrosion FEPs to be consistent with the Waste Form PMR. Ensured consistence of FEPs summary in Waste Form PMR. Made some editorial corrections and updated procedure references. Addressed concerns cited in the self assessment documented in SA-PA-2000-005 (MOL.20000719.0414 Added full FEP discussion for Volume Increase of Corrosion Products - YMP No. 2.1.09.03.00 from ANL-WIS-MD-000009 Gap and Grain release of Cs, I - YMP No. 2.1.02.07.00 from ANL-WIS-MD-000009 Rockfall (Large Block)- YMP No. 2.1.07.01.00 from ANL-WIS-MD-000009 This ICN addresses the design change eliminating backfill. Changes are indicated by vertical change bars in the right margin.

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ACRONYMS

AMR	Analysis and Model Report
CFR	Code of Federal Regulations
CLST	Container Life and Source Term
CRWMS	Civilian Radioactive Waste Management System
CSNF	Commercial Spent Nuclear Fuel
DCCG	Diffusion Controlled Cavity Growth
DOE	U.S. Department of Energy
DRKBA	Discrete Region Key Block Analysis
EBS	Engineered Barrier System
EPA	U.S. Environmental Protection Agency
FEPs	Features, Events, and Processes
FR	Federal Register
GE	General Electric
HLW	High Level Waste
IAEA	International Atomic Energy Agency
INEEL	Idaho National Engineering and Environmental Laboratory
IRSR	Issue Resolution Status Report
KTI	Key Technical Issue
LWR	Light Water Reactor
M&O	Management & Operating (Contractor)
MIC	Microbiologically Influenced Corrosion
MGR	Monitored Geologic Repository
NEA	Nuclear Energy Agency of the Organization for Economic Cooperation and Development
NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
OCRWM	Office of Civilian Radioactive Waste Management
PA	Performance Assessment
PMR	Process Model Report
QAP	Quality Administrative Procedure
QARD	Quality Assurance Requirements and Description
SCC	Stress Corrosion Cracking
SNF	Spent Nuclear Fuel
SRB	Sulfate Reducing Bacteria
TSPA	Total System Performance Assessment
TSPAI	Total System Performance Assessment and Integration
TSPA-SR	Total System Performance Assessment - Site Recommendation
UK	United Kingdom
USIC	Unsaturated and Saturated Flow under Isothermal Conditions
YMP	Yucca Mountain Project
WF	Waste Form
WIPP	Waste Isolation Pilot Plant
WP	Waste Package

1. PURPOSE

The purpose of this report is to summarize the work of others into the Features, Events, and Processes (FEPs) for cladding degradation screening that is consistent with and used in the Total System Performance Assessment - Site Recommendation (TSPA-SR). This review is to describe what FEPs are to be included in the TSPA-SR and also document the reason for excluding others. In accordance with AP-2.13Q (since superceded by ICN 04), *Technical Product Development Planning*, a work plan, *Cladding FEPs Screening Arguments* (CRWMS M&O 1999a), was developed, issued, and utilized in the preparation of this document.

Under the provisions of the U.S. Department of Energy's (DOE's) *Revised Interim Guidance Pending Issuance of New U. S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain Nevada* (Dyer 1999; and herein referred to as DOE's interim guidance), the DOE must provide a reasonable assurance that the regulatory-specified performance objectives for the Yucca Mountain Project can be achieved for a 10,000-year post-closure period. This assurance must be demonstrated in the form of a performance assessment that: (1) identifies the features, events, and processes (FEPs) that potentially affect the performance of the geologic repository; (2) examines the effects of such FEPs on the performance of the geologic repository; (3) estimates the expected annual dose to a specified nearby population group. The performance assessment must also provide the technical basis for inclusion or exclusion of specific FEPs.

Although the NRC has not defined nor used the term "scenario" in the pertinent regulations, the Yucca Mountain Total System Performance Assessment - Site Recommendation(TSPA-SR) has chosen to satisfy the above-stated performance assessment requirements by adopting a scenario development process. This decision was made based on the Yucca Mountain TSPA-SR adopting a definition of "scenario" as not being limited to a single, deterministic future of the system, but rather as a set of similar futures that share common FEPs. The DOE has chosen to adopt a scenario development process based on the methodology developed by Cranwell et al. (1990) for the NRC. The first step of this process is the identification of FEPs potentially relevant to the performance of the Yucca Mountain repository; the second step includes the screening of each FEP.

The primary purpose of this Analysis/Model Report (AMR) is to identify and document the analysis, screening decision, and Total System Performance Assessment (TSPA-SR) disposition or screening argument for FEPs related to clad degradation. Both primary and secondary FEPs are considered. The screening decisions, and associated TSPA-SR disposition or screening argument, for the subject FEPs will be cataloged separately in a project-specific FEPs database (see Section 1.4). This AMR and the database are being used to document information related to the FEPs screening decisions and associated screening argument and to assist reviewers during the license review process.

Most commercial nuclear fuel is encased in Zircaloy cladding. This AMR does not address potential damage to assemblies that might occur at the YMP surface facilities. There are constraints, caveats and limitations to this report. This FEPs screening is based on commercial Pressurized Water Reactor (PWR) fuel with Zircaloy cladding but is applicable to Boiling Water Reactor (BWR) fuel. This analysis referenced in the other AMRs is also limited to fuel exposed to normal operation and anticipated operational occurrences (i.e., events which are anticipated to occur within a reactor lifetime), and is not applicable to fuel that has been exposed to severe accidents. Fuel burnup projections have been limited to the current commercial reactor licensing environment with restrictions on fuel enrichment (5% for material shipment, fuel manufacturing, fuel shipment and fuel storage), oxide coating thickness, and rod plenum pressures. This assumption is consistent with the projections for advanced reloads. The fuel considered in the cladding has burnup up to 75 MWd/kgU and half of the fuel is above 44 MWd/kgU, today's typical PWR burnup range. Ranges and uncertainties have been defined in the other AMRs. The information provided in this FEPs screening will be used in evaluating what cladding degradation mechanisms are included in the post-closure performance of the Monitored Geologic Repository (MGR) in relation to waste form degradation.

1.1 SCOPE

This AMR has been prepared to satisfy the FEP screening documentation requirements in the Work Scope/Objectives/Tasks section of the development plan entitled *Cladding FEPs Screening Arguments* (CRWMS M&O 1999a).

The current (as of the date this work was planned) FEPs list consists of 1797 entries (as described in Section 1.2). The FEPs have been classified as primary and secondary FEPs (as described in Section 1.2) and have been assigned to various Process Model Reports (PMRs). The assignments were based on the nature of the FEPs so that the analysis and resolution for screening decisions reside with the subject-matter experts in the relevant disciplines. The resolution of other than EBS FEPs is documented in AMRs prepared by the responsible PMR groups. Several relevant FEPs do not fit into the existing PMR structure. Criticality is the largest example, and is treated in FEP assignments as if it were a separate PMR. Some FEPs were best assigned to the TSPA-SR itself (i.e., system-level FEPs), rather than to its component models.

This AMR addresses the clad degradation FEPs. These FEPs represent the key features that result in degradation of the cladding. The clad degradation primary and secondary EBS FEPs addressed in this AMR are provided in Table 1.

On March 30, 2000 a change to the MGR Project Description Document (CRWMS M&O 2000l) became effective to resolve certain thermal design issues. This design change will result in a greater ability of the waste packages to reject heat after closure of the repository, thereby maintaining the two thermal requirements. The first requirement is protective of the fuel cladding, and the second requires that a section of the rock pillar between drifts remain below the boiling temperature of water, providing a path for water drainage. This AMR originally considered a design with backfill, and has been updated to evaluate the without backfill.

The design of the repository continues to evolve in preparation for the future license application. In general, these design changes can directly influence the screening arguments for FEPs and their inclusion in or exclusion from the waste form degradation model. The elimination of backfill is an important design change that has been made after the development of the current Waste Form Degradation Model. The primary effect of the elimination of backfill is the decrease of peak temperatures inside the WP, which is beneficial for the following reasons (CRWMS M&O 2000c, Section 6.2). The temperature decrease (1) reduces the chance for creep rupture and stress corrosion cracking of CSNF cladding; (2) reduces the degradation rates of the CSNF and HLW matrices; (3) improves the applicability of the current data for in-package chemistry; and (4) decreases the solubility of uranium. However, design change to eliminate backfill does not affect the FEPs included in this AMR or the Waste Form Degradation Model because temperature is explicitly included as a model variable. That is, the model is an explicit function of the surface temperature of the waste, so any changes in the surface temperature of the waste package because of the design change are automatically included.

1.2 FEPs IDENTIFICATION AND ANALYSIS

The initial set of FEPs was created for the Yucca Mountain TSPA-SR by combining lists of FEPs previously identified as relevant to the Yucca Mountain Project (YMP) (e.g. *Total System Performance Assessment-1995: An Evaluation of the Potential Yucca Mountain Repository,* CRWMS M&O 1995a) with a draft FEP list compiled by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). The NEA list is maintained as an electronic FEP database and is the most comprehensive list available internationally. The list currently contains 1797 FEPs organized under 151 categories, based on NEA category headings. Consistent with the diverse backgrounds of the programs contributing FEPs lists, FEPs have been identified by a variety of methods, including expert judgment, informal elicitation, event tree analysis, stakeholder review, and regulatory stipulation. All potentially relevant FEPs have been included, regardless of origin. This approach has led to considerable redundancy in the FEP list, because the same FEPs are frequently identified by multiple sources, but it also ensures that a comprehensive review of narrowly defined FEPs will be performed. The FEPs list is considered open and will continue to grow as additional FEPs are identified.

There is no uniquely correct level of detail at which to define scenarios or FEPs. Decisions regarding the appropriate level of resolution for the analysis are made based on consideration of the importance of the scenario in its effect on overall performance and the resolution desired in the results. The number and breadth of scenarios depend on the resolution at which the FEPs have been defined: coarsely defined FEPs result in fewer, broad scenarios, whereas narrowly defined FEPs result in many narrow scenarios. For efficiency, both FEPs and scenarios should be aggregated at the coarsest level at which a technically sound argument can be made that is adequate for the purposes of the analysis.

Table 1. FEPs Related to the CSNF Clad Degradation Component

FEP Number	FEP Title
2.1.02.07.00	Gap and Grain Release of Cs, I
2.1.02.07.01	Gap and grain release
2.1.02.07.02	Pb-I reactions (in waste form)
2.1.02.07.03	I, Cs-migration to fuel surface
2.1.02.11.00	Waterlogged Rods
2.1.02.12.00	Cladding Degradation Before YMP Receives It
2.1.02.12.01	Pin Degradation During Reactor Operation
2.1.02.12.02	Pin Degradation During Spent Fuel Pool Storage
2.1.02.12.03	Pin Degradation During Dry Storage
2.1.02.12.04	Pin Degradation During Fuel Shipment and Handling
2.1.02.13.00	General Corrosion of Cladding
2.1.02.13.01	Cladding Degradation Mechanisms At YMP, Pre-Pin Failure
2.1.02.13.02	Corrosion (of Cladding)
2.1.02.14.00	Microbiologically Influenced Corrosion (MIC) of Cladding
2.1.02.15.00	Acid Corrosion of Cladding From Radiolysis
2.1.02.16.00	Localized (Pitting) Corrosion of Cladding
2.1.02.17.00	Localized Corrosion (Crevice Corrosion) of Cladding
2.1.02.18.00	High Dissolved Silica Content of Waters Enhances Corrosion of Cladding
2.1.02.19.00	Creep Rupture of Cladding
2.1.02.19.01	Thermal cracking (in waste and EBS)
2.1.02.20.00	Pressurization From He Production Causes Cladding Failure
2.1.02.21.00	Stress Corrosion Cracking (SCC) of Cladding
2.1.02.21.01	Inside Out From Fission Products (Iodine) (Failure of Cladding)
2.1.02.21.02	Outside In From Salts Or WP Chemicals (Failure of Cladding)
2.1.02.21.03	Stress-corrosion cracking of Zircaloy cladding
2.1.02.22.00	Hydride Embrittlement of Cladding
2.1.02.22.01	Hydride Embrittlement From Zirconium Corrosion (of Cladding)
2.1.02.22.02	Hydride Embrittlement From WP Corrosion & H_2 Absorption (of Cladding)
2.1.02.22.03	Hydride Embrittlement From Galvanic Corrosion of WP Contacting Cladding
2.1.02.22.04	Delayed Hydride Cracking (of Cladding)
2.1.02.22.05	Hydride Reorientation (of Cladding)
2.1.02.22.06	Hydrogen Axial Migration (of Cladding)
2.1.02.22.07	Hydride Embrittlement From Fuel Reaction (Causes Failure of Cladding)
2.1.02.23.00	Cladding Unzipping
2.1.02.23.01	Cladding Degradation after Initial Cladding Perforation
2.1.02.23.02	Dry Oxidation of Fuel (Causes Failure of Cladding)
2.1.02.23.03	Wet Oxidation of Fuel (Causes Failure of Cladding)
2.1.02.24.00	Mechanical Failure of Cladding
2.1.02.26.00	Diffusion-Controlled Cavity Growth
2.1.02.27.00	Localized Corrosion Perforation from Fluoride
2.1.07.01.00	Rockfall (Large Block)
2.1.07.01.01	Rockbursts in container holes
2.1.07.01.02	Cave ins
2.1.07.01.03	Cave in (in waste and EBS)
2.1.07.01.04	Roof falls
2.1.09.03.00	Volume Increase of Corrosion Products
2.1.09.03.01	Swelling of corrosion products (in waste and EBS)
2.1.11.07.00	Thermally-Induced Stress Changes in Waste and EBS
2.1.11.07.01	Changes in in-situ stress field (in waste and EBS)
2.1.11.07.02	Stress field changes, settling, subsidence or caving

Consequently, each FEP has been identified as either a primary or secondary FEP. Primary FEPs are those FEPs for which the project proposes to develop detailed screening arguments. The classification and description of primary FEPs strive to capture the essence of all the secondary FEPs that map to the primary. For example, the primary FEP "Cladding Degradation Before YMP Receives It" can be used appropriately to resolve multiple and redundant secondary FEPs that address degradation during reactor operation, during spent fuel pool storage, etc. By working to the primary FEP description, the subject matter experts assigned to the primary FEP address all relevant secondary FEPs, and arguments for secondary FEPs can be rolled into the primary FEP analysis. Secondary FEPs are FEPs either that are completely redundant or that can be aggregated into a single primary FEP.

To perform the screening and analysis, the FEPs have been assigned so that the analysis, screening decision, and TSPA-SR disposition reside with the subject matter experts in the relevant disciplines. The TSPA-SR recognizes that FEPs have the potential to affect multiple facets of the project, may be relevant to more than one PMR, or may not fit neatly within the PMR structure. For example, many FEPs affect waste form (WF), waste package (WP), and the EBS. Rather than create multiple separate FEPs, the FEPs have been assigned, as applicable, to one or more process modeling groups, which are responsible for the AMRs.

At least two approaches have been used to resolve overlap and interface problems of multiple numbers for the same FEP. FEP owners from different process modeling groups may decide that only one PMR will address all aspects of the FEP, including those relevant to other PMRs. Alternatively, FEP owners may each address only those aspects of the FEP relevant to their area. In either case, the FEP AMR produced by each process modeling group lists the FEP and summarizes the screening result, citing the appropriate work in related AMRs as needed.

1.3 FEPs SCREENING AND ANALYSIS PROCESS

As described in Section 1.2, the first step in the scenario development process was the identification and analysis of FEPs. The second step in the scenario development process includes the screening of each FEP. Each FEP is screened for inclusion or exclusion in the TSPA-SR against three criteria, which are stated as regulatory requirements in NRC's proposed rule 10 CFR Part 63 (64 FR 8640), and in the U.S. Environmental Protection Agency's (EPA) proposed rule 40 CFR Part 197 (64 FR 46976). The screening criteria are discussed in more detail in Section 4.2 and are summarized here. FEPs are excluded from the TSPA-SR only if:

- They are specifically ruled out by regulation, are contrary to the stated regulatory assumptions, or are in conflict with statements made in background information regarding intent or directions of the regulations.
- They can be shown to have a probability of occurrence of less than 10^{-4} in 10^{4} years.
- Their occurrence can be shown to have no significant effect on the overall expected annual dose calculated by the TSPA.

The regulatory screening criteria contained in DOE's interim guidance (Dyer 1999) and in the proposed 40 CFR Part 197 (64 FR 46976) are relevant to many of the FEPs. FEPs that are contrary to DOE's interim guidance or to specific proposed regulations, regulatory assumptions, or regulatory intent are excluded from further consideration. Examples include: the explicit exclusion of all but a specified scenario to address treatment of human intrusion (10 CFR §63.113(d)), assumptions about the critical group to be considered in the dose assessment (10 CFR §63.115), and the intent that the consideration of "the human intruders" be excluded from the human intrusion assessment (64 FR 8640, Section XI. Human Intrusion).

Probability estimates used in the FEPs screening process may be based on technical analysis of the past frequency of similar events (such as igneous and seismic events) or, in some cases, on expert elicitation. Probability arguments, in general, require including some information about the magnitude of the event in its definition. Probability arguments are also sensitive to the spatial and temporal scales at which FEPs are defined. For example, the definition of the probability of a seismic event depends on the magnitude of the event. Probability arguments are therefore made at reasonably coarse scales.

Consequence-based screening arguments can be established in a variety of ways. Various methods include TSPA-SR sensitivity analyses, modeling studies outside of the TSPA-SR, or reasoned arguments based on literature research. For example, consequences of many geomorphic processes such as erosion and sedimentation can be evaluated by considering bounding rates reported in geologic literature. More complicated processes, such as igneous activity, require detailed analyses conducted specifically for the Yucca Mountain Project. Low-consequence arguments are often made by demonstrating that a particular FEP has no effect on the distribution of an intermediate performance measure in the TSPA-SR. For example, by demonstrating that including a particular waste form has no effect on the concentrations of radionuclides transported from the repository in the aqueous phase, it is also demonstrated that including this waste form in the inventory would not compromise compliance with the performance objectives. Explicit modeling of the characteristics of this waste form could therefore be excluded from the TSPA-SR.

Using the type of arguments discussed above, each FEP identified as relevant to the clad degradation was reviewed against the three exclusion criteria. Those that were determined to meet one of the three criteria were designated as "excluded" from further consideration within the TSPA-SR. Those that did not meet any of these criteria must, by definition, be "included."

1.4 ORGANIZATION OF FEP DATABASE

Under a separate scope, the TSPA-SR team is constructing an electronic FEP database to assist project reviewers during the license review process (CRWMS M&O 2000o, Appendix D). Each FEP has been entered as a separate record in the database. Fields within each record provide a unique identification number, a description of the FEP, the origin of the FEP, identification as a primary or secondary FEP for the purposes of the TSPA-SR, and mapping to related FEPs and to the assigned PMRs. Fields also provide summaries of the screening arguments with references to supporting documentation and AMRs, and, for all included or retained FEPs, statements of the disposition of the FEP within the TSPA-SR modeling system. The AMRs, however, contain the detailed arguments and description of the disposition of the subject FEPs.

Alphanumeric identifiers (called the "NEA category") previously used have been retained in the database for traceability purposes. Each FEP has also been assigned a unique YMP FEP database number, based on the NEA categories. The database number is the primary method for identifying FEPs, and consists of an eight-digit number of the form x.y.zz.pp.qq. The general structure of the database is reflected in the first two digits (x.y) as shown below:

- 0.0. Assessment Basis
- 1.0. External Factors
  - 1.1 Repository Issues
  - 1.2 Geological Processes and Effects
  - 1.3 Climatic Processes and Effects
  - 1.4 Feature Human Actions (Active)
  - 1.5 Other
- 2.0. Disposal System - Environmental Factors
  - 2.1 Wastes and Engineered Features
  - 2.2 Geologic Environment
  - 2.3 Surface Environment
  - 2.4 Human Behavior
- 3.0. Disposal System - Radionuclide/Contaminant Factors
  - 3.1 Contaminant Characteristics
  - 3.2 Contaminant Release/Migration Factors
  - 3.3 Exposure Factors

The next six digits (zz.pp.qq) define a grouping structure for the FEPs, with zz designating the category, and pp designating the heading. The exact details of this grouping structure are not important to the evaluation, since each FEP will be evaluated regardless of the database organization. Finally, the last two digits (qq) signify whether the FEP is primary (00) or secondary (other than 00). Each heading has a primary FEP associated with it, and may or may not have any secondary FEPs. In those cases where secondary FEPs do exist, the primary FEP encompasses all the issues associated with the secondary FEPs. The secondary FEPs either provide additional detail concerning the primary, or are a restatement of the primary based on redundant input from a different source.

2. QUALITY ASSURANCE

The Quality Assurance program applies to the development of this analysis documentation. The Performance Assessment Operations responsible manager has evaluated the technical document development activity in accordance with QAP-2-0, *Conduct of Activities*. The QAP-2-0 activity evaluation, *Conduct of Performance Assessment* (CRWMS M&O 1999c), has determined that the preparation and review of this technical document is subject to *Quality Assurance Requirements and Description* DOE/RW-0333P (DOE 2000) requirements. Note that the activity evaluation (CRWMS M&O 1999c) remains in effect even though QAP-2-0 has been superseded by AP-2.21Q, *Quality Determinations and Planning for Scientific, Engineering, and Regulatory Compliance Activities*. Preparation of this analysis did not require the classification of items in accordance with QAP-2-3, *Classification of Permanent Items*. This activity is not a field activity. Therefore, an evaluation in accordance with NLP-2-0, *Determination of Importance Evaluations* was not required. The methods used to control the electronic management of data as required by AP-SV.1Q, *Control of the Electronic Management of Information*, were not specified in the Development Plan, *Cladding FEPs Screening Arguments* (CRWMS M&O 1999a). With regard to the development of this AMR, the control of electronic management of data was evaluated in accordance with YAP-SV.1Q, *Control of the Electronic Management of Data*. The evaluation (CRWMS M&O 2000n) determined that current work processes and procedures are adequate for the control of electronic management of data for this activity. Though YAP-SV.1Q has been replaced by AP-SV.1Q, this evaluation remains in effect.

This AMR has been developed in accordance with procedure AP-3.10Q, *Analyses and Models*. Preparation of this analysis did not require the classification of items in accordance with QAP-2-3, *Classification of Permanent Items*. This activity is not a field activity. Therefore, an evaluation in accordance with NLP-2-0, *Determination of Importance Evaluations* was not required.

3. COMPUTER SOFTWARE AND MODEL USAGE

This AMR uses no computational software nor model. The AMR was documented using only commercially available software (Microsoft Word 97-SR2) for word processing, which is exempt from qualification requirements in accordance with AP-SI.1Q, *Software Management*. There were no additional applications (routines or macros) developed using this commercial software. The analyses and arguments presented herein are based on regulatory requirements, results of analyses presented and documented in other AMRs, or technical literature.

4. INPUTS

There are no input data sources used in this analysis.

4.1 Data and Parameters

This section is not applicable to this analysis.

4.2 Criteria

Programmatic requirements for this document are listed in the development plan (CRWMS M&O 1999a), which covers eight tasks needed to provide the submodels for the performance assessment Physical and Chemical Environment Abstraction Model. The development plan specifies that this document and all analyses described herein must adhere to the requirements of AP-3.10Q, *Analyses and Models*, and must address applicable NRC issue resolution status report (IRSR) criteria for the TSPA (NRC 2000a).

The U.S. Nuclear Regulatory Commission's Total System Performance Assessment and Integration (TSPAI) Issue Resolution Status Report (NRC 2000b) establishes generic technical acceptance criteria considered by the NRC staff to be essential to a defensible, transparent, and comprehensive assessment methodology for the repository system. The NRC's IRSR *Key Technical Issue: Container Life and Source Term* (NRC 1999) establishes generic technical acceptance criteria considered by the NRC staff for the waste form, with the cladding degradation analysis being part of this KTI. Attachment IV of *Initial Cladding Condition* (CRWMS M&O 2000b) describes how IRSR issues and criteria are addressed.). Section 6.3 describes how this AMR addresses the IRSR issues and criteria.

Technical objectives are provided in DOE's interim guidance (Dyer 1999) and have also been identified by the NRC in the proposed 10 CFR Part 63 (64 FR 8640) and by the EPA in the proposed 40 CFR Part 197 (64 FR 46976). Both proposed regulations specifically allow the exclusion of FEPs from the TSPA-SR if they are of low probability (less than one chance in 10,000 of occurring in 10,000 years) or if occurrence of the FEP can be shown to have no significant effect on expected annual dose. There is no quantified definition of "significant effect" in the guidance nor proposed regulations.

4.2.1 Low Probability

The probability objective is explicitly stated by the NRC in the proposed 10 CFR §63.114 (d):

> Consider only events that have at least one chance in 10,000 of occurring over 10,000 years.

The EPA provides essentially the same objective in 40 CFR §197.40:

> The DOE's performance assessments should not include consideration of processes or events that are estimated to have less than one chance in 10,000 of occurring within 10,000 years of disposal.

4.2.2 Low Consequence

Objectives for low consequence screening arguments are provided in DOE's interim guidance (Dyer 1999, Section 114(e) and (f)), which indicates that performance assessment shall:

(e) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the performance assessment. Specific features, events, and processes of the geologic setting must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

(f) Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

The EPA provides essentially the same objective in 40 CFR §197.40:

> ...with the NRC's approval, the DOE's performance assessment need not evaluate, in detail, the impacts resulting from any processes and events or sequences of processes and events with a higher chance of occurrence if the results of the performance assessment would not be changed significantly.

The terms "[not] significantly changed" and "not .. changed significantly" are undefined terms in DOE's interim guidance and in the EPA's proposed regulations. These terms are inferred for FEPs screening purposes to be equivalent to having no or negligible effect. Because the relevant performance measures differ for different FEPs (e.g., effects on performance can be measured in terms of changes in concentrations, flow rates, travel times, and other measures, as well as overall expected annual dose), there is no single quantitative test of "significance."

4.2.3 Reference Biosphere

Both DOE's interim guidance (Dyer 1999) and EPA's proposed regulations specify objectives (which in effect serve as objectives) pertinent to screening many of the EBS FEPs. Particularly germane are explicit objectives regarding the reference biosphere (10 CFR §63.115), and less so are objectives regarding the location and use of groundwater by the critical group used for calculation of exposure doses.

The objectives pertaining to the characteristics of the reference biosphere are presented in DOE's interim guidance (Dyer 1999, Section 115(a)1). The specified characteristics pertinent to the EBS FEPs are that:

- Features, events, and processes ...shall be consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site.

The EPA has specified a similar objective in proposed 40 CFR §197.15. This objective is stated as:

> ... DOE must vary factors related to the geology, hydrology, and climate based on environmentally protective but reasonable scientific predictions of the changes that could affect the Yucca Mountain disposal system over the next 10,000 years.

4.3 Codes and Standards

American Society for Testing and Materials (ASTM) Standard C1174-97–*Standard Practice for Prediction of the Long-Term Behavior of Materials, Including Waste Forms, Used in Engineered Barrier Systems (EBS) for Geologic Disposal of High-Level Radioactive Waste* (ASTM 1998) is used to support the degradation analysis development methodology, categorize the analysis developed with respect to their usage for long-term TSPA, and relate the information/data used to develop the analysis to the requirements of the standard.

This AMR was prepared to comply with the above NRC TSPAI acceptance objectives, as well as the DOE interim guidance (Dyer 1999) which requires the use of specified Subparts/Sections of the proposed NRC high-level waste rule, 10 Code of Federal Register (CFR) Part 63 (64 FR 8640). Subparts of this proposed rule that are particularly applicable to data include Subpart B, Section 15 (Site Characterization) and Subpart E (Technical Criteria). Subparts applicable to analysis are outlined in Subpart E, Sections 114 (*Requirements for Performance Assessment*) and 115 (*Characteristics of the Reference Biosphere and Critical Group*).

Material standards are also cited. ASTM B 551 and ASTM B 811 are referenced in Section 6.2.12.

5. ASSUMPTIONS

This AMR makes no assumptions beyond those reported and documented by the references. The key assumptions used for the modeling of the cladding degradation are given in CRWMS M&O 2000b and CRWMS M&O 2000c. The key assumptions are listed below.

5.1 It is assumed that the spent fuel (CSNF) that is accepted for disposal in the repository will have assumed characteristics similar to those described in the technical publications by industrial leaders. The basis for this assumption is: 1) the information in in technical publications was collected for a broad range of fuels, 2) it is compared with other sources in the open literature, 3) the information was measured on actual spent nuclear fuel of the various types currently in use or in storage at nuclear facilities, 4) this technical information was generally produced under NRC-accepted or foreign-nuclear-agency-accepted nuclear quality assurance programs, and 5) much of this information was produced to support the licensing process for the fuel. These assumptions are used to develop statistical distributions (ranges) for the properties of the fuel to be received. (used through out the cladding degradation analysis)

5.2 The Westinghouse 17 by 17 Lopar design (called W1717WL) fuel assembly was selected to represent all fuel cladding for creep, Delayed Hydride Cracking, and Stress

Corrosion Cracking analysis. The basis for this assumption is that this design is the most commonly used assembly, constituting 21 percent of the discharged Pressurized Water Reactor (PWR) assemblies. The W1717WL is the largest fraction of the more general W1717 type design that constitutes 33 percent of the discharged PWR fuel. The W1717 design is the thinnest Zircaloy clad fuel (570 microns cladding thickness) (CRWMS M&O 2000b). (Sections 6.2.9, 6.2.11, 6.2.12.4)

5.3 The Waste Packages (WP) will be loaded with spent fuel in the order of discharge of the fuel from the various reactors as a function of calendar years. This generates some variability in the fraction of rods failed within a WP. This loading sequence tends to place fuel with higher cladding failure rates into the same WP or consecutively loaded WPs and produces larger variations in rod failure fractions than would be expected if thermal blending were employed. This is a credible and reasonable assumption based on the fuel that current owners would be expected to ship first. (Section 6.2.2.1)

5.4 All rods are exposed to the conditions of dry storage with the center rod in the cask operating at the design temperature of the Castor Mark V (CRWMS M&O 2000a) cask. (Sections 6.2.9, 6.2.11)

5.5 BWR cladding degrades in a similar manner as the base case PWR fuel. This is reasonably bounding since, in comparison to PWR fuel, the BWR cladding is thicker, the BWR fuel typically is discharged with lower burnups and stresses, and each BWR fuel assembly is enclosed in a flow channel for additional protection. (Section 6.1)

5.6 Any rod with a stress greater than 180 MPa is assumed to fail from Stress Corrosion Cracking (SCC). In this AMR, sufficient iodine is assumed to exist such that SCC occurs at the stress threshold of 180 MPa. (Section 6.2.11)

5.7 It is assumed that localized corrosion by aggressive species can be modeled by idealized corrosion of fluorides. Aggressive species could be $FeCl_3$, MIC produced acids, or radiolysis produced acids. It is assumed that corrosion of the cladding is limited by the supply of fluoride and acting on a small area of rod. The rationale for this assumption is that little information is available on the rate of consumption of fluoride in a waste package. In the absence of information, a bounding approach is needed. Since the assumption is an upper limit, no confirmation is necessary. (Section 6.2.4, 6.2.5, 6.2.6)

5.8 It is assumed that, during the unzipping phase, the fuel reacts with water to form metaschoepite. Oxidation and hydration of uranium dioxide can result in a variety of mineral species. Of these, metaschoepite is the one that entails the largest change in volume, and larger volume increases correspond to faster degradation. Other uranium minerals (such as sodium boltwoodite) can also be formed from uranium dioxide, but these require a supply of a solute (such as sodium), so the rate of formation will be limited by the supply of solute. Therefore, the rationale for this assumption is that conversion to metaschoepite provides the largest plausible volume increase. Since the assumption is reasonably bounding, no confirmation is necessary. (Section 6.2.13)

5.9 It is assumed that the rate of reaction of the uranium dioxide with water is controlled by the intrinsic dissolution rate of UO_2. The rationale for this assumption is that this is the fastest rate at which reaction can advance into a uranium dioxide pellet surface. If the products of reaction of UO_2 limit the transport of water to the uranium dioxide surface, the reaction will necessarily be slower. Since this assumption is reasonably bounding, no confirmation is necessary. (Section 6.2.13)

5.10 It is assumed that all cladding breaches occur at the center of the active fuel length. The rationale for this assumption is that this location provides the fastest unzipping of a fuel rod. If the breach is at the center of the active fuel length, propagation of the breach by a distance of half the active fuel length (toward each end) will result in complete unzipping of the active fuel length of the rod. If the breach is at some other location, the required propagation distance will be larger for one end. Since this assumption is reasonably bounding, no confirmation is necessary. (Section 6.2.13)

5.11 The stainless steel (SS) clad fuel is loaded into WPs as it is received at the YMP facilities. This is used to define the number of WPs containing SS cladding and the fraction of SS cladding in these WPs. The basis of this assumption is that it simplifies surface facility operations. This assumption is not critical since the product of WPs containing SS cladding and fraction of SS in each WP is constant (i.e. there is a fixed amount of SS cladding). (Section 6.2.13)

6. ANALYSIS/MODEL - FEPs RELATED TO CLADDING DEGRADATION

To demonstrate that the regulatory-specified performance objectives of proposed 10 CFR Part 63 and proposed 40 CFR Part 197 can be achieved for a 10,000-year post-closure period, the Yucca Mountain Project is implementing a stochastic scenario development methodology based on the work of Cranwell et al. (1990). The methodology provides a systematic approach for considering, as completely as practicable, the possible future states of a repository system. It seeks to span the set of all possible future states using a finite set of scenarios. Each scenario represents the ensemble of possible future states corresponding to parameter and model uncertainties present in the group of FEPs composing the scenario. The methodology begins with a comprehensive FEP identification step followed by a rigorous FEP screening step. With its focus on FEPs related to cladding degradation, this Analysis/Model Report (AMR) considers these first two steps (i.e., FEP identification and FEP screening) of scenario development. Screening criteria for principal factors or potentially disruptive events are discussed in *Managing Technical Product Inputs*, AP-3.15Q. This analysis is classified as "Level 2" since it does support "Commercial Spent Nuclear Fuel Waste Form Performance" which is not classified as a principal factor.

6.1 Identification of FEPs

The first step of the scenario development methodology is the identification of FEPs potentially relevant to the performance of the Yucca Mountain repository. This process is summarized in Section 1.2.

Each FEP has been identified as either a "primary" or a "secondary" FEP. Primary FEPs are those FEPs for which the project proposes to develop detailed screening arguments. The classification and description of primary FEPs strive to capture the essence of each secondary FEP that maps to the primary FEP. For example, the primary FEP *Meteorite Impact* can be used appropriately to resolve multiple and redundant secondary FEPs that address size and effects of meteorite impacts. By working to the primary FEP description, the subject matter experts assigned to the primary FEP will address each relevant secondary FEP, and arguments for secondary FEPs will be rolled into the primary FEP analysis.

Table 2 identifies both the primary and the secondary FEPs related to cladding degradation.

6.2 Screening of FEPs

The second step in the scenario development methodology includes the screening of each FEP. Each FEP is screened for inclusion in or exclusion from the TSPA-SR against three objectives, which are stated as regulatory requirements in the NRC proposed rule 10 CFR Part 63 and in EPA proposed rule 40 CFR Part 197. FEPs are excluded from the TSPA-SR for one of the following reasons:

(1) They are specifically ruled out by regulation, are contrary to the stated regulatory assumptions, or are in conflict with statements made in background information regarding intent or directions of the regulations.
(2) They can be shown to have a probability of occurrence of less than 10^{-4} in 10^{4} years.
(3) Their occurrence can be shown to have no significant effect on the overall expected annual dose by the TSPA.

Examples of reason (1) are the explicit exclusion from consideration of all but a specific scenario to address treatment of human intrusion (10 CFR §63.113(d)) and assumptions about the critical group to be considered in the dose assessment (10 CFR §63.115).

Probability estimates used in the FEP screening process in relation to reason (2) may be based on technical analysis of the past frequency of similar events (such as seismic events), or, in some cases, on expert elicitation. Probability arguments, in general, require including appropriate information about the magnitude of the event in its definition. Probability arguments are also sensitive to the spatial and temporal scales at which FEPs are defined. For example, definition of the probability of meteorite impact depends on the size of the meteorite of interest and must consider that meteorite impacts are less likely in shorter time intervals and at smaller locations. Probability arguments are therefore made at reasonably coarse scales based on the judgment of the originator.

Consequence based screening arguments in relation to reason (3) can be established in a variety of ways. Various methods include TSPA-SR sensitivity analyses, modeling studies outside of the TSPA-SR, or reasoned arguments based on literature research. For example, consequences of many geomorphic processes such as erosion and sedimentation can be evaluated by considering bounding rates reported in geologic literature. More complicated processes, such as criticality, require detailed analyses conducted specifically for the Yucca Mountain Project. Low consequence arguments are often made by demonstrating that a particular FEP has no effect on the distribution of an intermediate performance measure of the TSPA-SR. For example, to demonstrate that including a particular waste form does not compromise compliance with performance objectives, one may show that aqueous phase concentrations of radionuclides transported from the repository would be unaffected by the exclusion of the specific waste form. Explicit modeling of the characteristics of this waste form could then be excluded from the TSPA-SR.

Table 2 presents the "screening decision" as either "*include*" or "*exclude*" for FEPs related to cladding degradation. However, for those cases in which there are secondary FEPs, one should be aware of the rule for assigning a composite decision to the primary FEP. Specifically, a primary FEP is designated as *include* if at least one of its secondary FEPs is designated as *include*. Conversely, a primary FEP is designated as *exclude* only if all of its secondary FEPs are also designated as *exclude*. Section 6.2 provides detailed documentation of both the screening argument and the TSPA-SR disposition for each of the primary FEPs. Table 2 lists the specific subsection where each FEP is discussed. If the primary FEP has no secondary FEPs, then the decision to *include* or *exclude* the primary FEP is made for that FEP itself.

Also provided in each FEP section is a cross reference to key technical issues identified by the NRC as being important for the Yucca Mountain repository. These are identified as Issue Resolution Status Report (IRSR) issues. The key technical issues and subissues are listed below. The relevance of these subissues to the EBS FEPs is identified in Sections 6.2.1 through 6.2.15. Whenever the key technical issue (CLST, for example) is identified rather than a specific subissue, all subissues apply. All the FEPs discussed in this AMR fall into the Container Life and Source Term or the Total System Performance Assessment and Integration (TSPAI, NRC 2000b) subissues which are listed below:

Container Life and Source Term (CLST)

CLST1	The effects of corrosion processes on the lifetime of the containers
CLST2	The effects of phase stability of materials and initial defects on the mechanical failure and lifetime of the containers
CLST3	The rate at which radionuclides in spent nuclear fuel (SNF) are released from the EBS through the oxidation and dissolution of spent fuel
CLST4	The rate at which radionuclides in high-level waste (HLW) glass are leached and released from the EBS
CLST5	The effect of in-package criticality on WP and EBS performance

CLST6	The effect of alternate EBS design features on container lifetime and radionuclide release from the EBS

Total System Performance Assessment and Integration (TSPAI)

TSPAI1	Demonstration of the overall performance objectives
TSPAI2	Demonstration of multiple barriers
TSPAI3	Model abstraction
TSPAI4	Scenario analysis
TSPAI5	Transparency and traceability of the analysis

Table 2. Screening Decisions for FEPS Related to Cladding Degradation

YMP FEP Database Number	NEA Category	FEP Name	Primary (P) or Secondary (S) FEP	Screening Decision	Section
2.1.02.07.00	2.1.02e	Gap and Grain Release of Cs, I	P	Include	6.2.13.4
2.1.02.11.00	WF-3	Waterlogged Rods	P	Exclude	6.2.1
2.1.02.12.00	2.1.03ca	Cladding Degradation Before YMP Receives It	P	Include	6.2.2
2.1.02.12.01	2.1.03cb	Pin Degradation During Reactor Operation	S	Include	6.2.2.1
2.1.02.12.02	2.1.03cc	Pin Degradation During Spent Fuel Pool Storage	S	Exclude	6.2.2.2
2.1.02.12.03	2.1.03cd	Pin Degradation During Dry Storage	S	Include	6.2.2.3
2.1.02.12.04	2.1.03ce	Pin Degradation During Fuel Shipment and Handling	S	Include	6.2.2.4
2.1.02.13.00	2.1.03ch	General Corrosion of Cladding	P	Exclude	6.2.3
2.1.02.14.00	2.1.03ci	Microbiologically Influenced Corrosion (MIC) of Cladding	P	Include	6.2.4
2.1.02.15.00	2.1.03cj	Acid Corrosion of Cladding From Radiolysis	P	Include	6.2.5
2.1.02.16.00	2.1.03ck	Localized Corrosion (Pitting) of Cladding	P	Include	6.2.6
2.1.02.17.00	2.1.03cl	Localized Corrosion (Crevice Corrosion) of Cladding	P	Exclude	6.2.7
2.1.02.18.00	2.1.09f	High Dissolved Silica Content of Waters Enhances Corrosion of Cladding	P	Exclude	6.2.8
2.1.02.19.00	2.1.03cm	Creep Rupture of Cladding	P	Include	6.2.9
2.1.02.20.00	2.1.03cn	Pressurization From He Production Causes Cladding Failure	P	Include	6.2.10
2.1.02.21.00	2.1.03cp	Stress Corrosion Cracking (SCC) of Cladding	P	Include	6.2.11
2.1.02.21.01	2.1.03cq	Inside Out From Fission Products (Iodine) (Failure of Cladding)	S	Include	6.2.11
2.1.02.21.02	2.1.03cr	Outside In From Salts Or WP Chemicals (Failure of Cladding)	S	Include	6.2.11
2.1.02.22.00	2.1.03cs	Hydride Embrittlement of Cladding	P	Exclude	6.2.12
2.1.02.22.01	2.1.03ct	Hydride Embrittlement From Zirconium Corrosion (of Cladding)	S	Exclude	6.2.12.1
2.1.02.22.02	2.1.03cu	Hydride Embrittlement From WP Corrosion & H_2 Absorption (of Cladding)	S	Exclude	6.2.12.2

YMP FEP Database Number	NEA Category	FEP Name	Primary (P) or Secondary (S) FEP	Screening Decision	Section
2.1.02.22.03	2.1.03cv	Hydride Embrittlement From Galvanic Corrosion of WP Contacting Cladding	S	Exclude	6.2.12.3
2.1.02.22.04	2.1.03cw	Delayed Hydride Cracking (of Cladding)	S	Exclude	6.2.12.4
2.1.02.22.05	2.1.03cx	Hydride Reorientation (of Cladding)	S	Exclude	6.2.12.5
2.1.02.22.06	2.1.03cy	Hydrogen Axial Migration (of Cladding)	S	Exclude	6.2.12.6
2.1.02.22.07	2.1.03dd	Hydride Embrittlement From Fuel Reaction (Causes Failure of Cladding)	S	Exclude	6.2.12.7
2.1.02.23.00	2.1.03da	Cladding Unzipping	P	Include	6.2.13
2.1.02.23.02	2.1.03db	Dry Oxidation of Fuel (Causes Failure of Cladding)	S	Exclude	6.2.13.1
2.1.02.23.03	2.1.03dc	Wet Oxidation of Fuel (Causes Failure of Cladding)	S	Include	6.2.13.2
2.1.02.24.00	2.1.03co	Mechanical Failure of Cladding	P	Include	6.2.14
2.1.02.27.00	WFCld AMR-1	Localized Corrosion Perforation from Fluoride	P	Include	6.2.16
2.1.02.26.00	NEW	Diffusion-Controlled Cavity Growth	P	Exclude	6.2.17
2.1.07.01.00	2.1.07a	Rockfall (Large Block)	P	Exclude	6.2.15
2.1.09.03.00	2.1.09av	Volume Increase of Corrosion Products	P	Include	6.2.13.3
2.1.11.07.00	2.1.11ab	Thermally-Induced Stress Changes in Waste and EBS	P	Include	6.2.9

6.2.1 Waterlogged Rods - YMP No. 2.1.02.11.00

Related Primary FEPs: No closely related FEPs

FEP Description Failed fuel rods that occur in up to 0.1% of the fuel rods are currently being stored in commercial reactor spent fuel pools. This is attributed to breaches caused by manufacturing defects and reactor operations. Failed fuel contains water in the fuel rod void space. Such fuel is referred to as "waterlogged". The moisture remaining in a "dried" fuel rod is used to determine the extent of degradation of spent fuel cladding.

Screening Decision: Excluded based on low probability (credibility).

Screening Argument: The omission of waterlogged rods is justified on the basis that failure of fuel rods from waterlogging is not credible because the supply of water that remains in the fuel after cask drying will not be sufficient to oxidize the fuel to an unacceptable level. (NRC 1997, Section 8.V.1). Few rods are breached at the time they are received at the repository. Breached rods are dried effectively using standard cask drying techniques.

The fraction of fuel rods with breached cladding is small (less than 0.2%). Moisture can be removed from defective rods during the cask drying operations. The residual moisture of the cask atmosphere can be estimated based on the drying conditions to 5 mbars (= 500 Pa). The impact on the waste form, waste package, and cladding of a small amount of remaining moisture is negligible.

TSPA Disposition Waterlogged rods are excluded from the TSPA as described under the Screening Argument.

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 1995b, EPRI 1997, Kohli, R. and Pasupathi, V. 1986, Peehs, M. and Fleisch, J. 1986, NRC 1997, CRWMS M&O 2000b, Knoll and Gilbert 1987.

Basis for Screening Decision

The fraction of fuel rods with breached cladding is currently estimated to be 0.164% (CRWMS M&O 2000b, Table 13), or about 9.1 rods in an "average" waste package. EPRI 1997 (p. 4-1 and 4-2) gives the BWR and PWR fuel rod reliability as a function of calendar year based and their estimates are lower than that estimated in CRWMS M&O 2000b.

Since cask drying operations will remove most of the water from the fuel rods, it is reasonable to expect that the supply of water that remains in the fuel after cask drying will not be sufficient to oxidize the fuel to U_3O_8. NRC (1997, Section 8.V.1) describes the cask drying criteria with reference to Knoll and Gilbert 1987. Less than 0.43 mol of H_2O are expected to be present in a 7-m^3 cask after drying. This volume of water produces an insignificant potential for corrosion of the cladding during dry storage or during disposal (CRWMS M&O 1995b, p. 16).

Kohli and Pasupathi (1986, p. iii) discusses removal of water from waterlogged fuel rods. Two reactor breached fuel rods were tested, along with two fuel rods that were intentionally defected after irradiation. Since the initial amount of water in the reactor operations breached rods is unknown before drying, a predetermined amount of moisture was added to the intentionally defected rods to enable the extent of the moisture released during the drying to be determined. The rods were dried in flowing argon at atmospheric pressure while being heated in a furnace. The center 1.8m of the furnace was heated to 673 K; the remainder was heated to 473 K. The reactor operations breached rods were dried in the as-received condition, then a hole was drilled in the cladding, water was injected, and the experiment was repeated. In the reactor breached rods, the bulk of the uncombined water was removed in 3600 to 4800 sec and all measurable releases ended after 13200 sec (3.7 hours). This set of experiments demonstrated that standard cask drying procedures would remove the water from the failed rods.

Peehs and Fleisch (1986, pp. 199-202) described the behavior of waterlogged PWR fuel rods on heating at 400°C in a hot cell. The bulk of the water was released during the cask drying operation. Results of the testing were that the moisture can be removed from defective rods during the cask drying operations and the residual moisture of the cask atmosphere can be minimized.

If there were problems with the drying process and rods were filled with water, the extent of the potential corrosion can be estimated. CRWMS M&O (2000b Section 6.3.5) discusses the free volume in a rod and concludes that the average irradiated rod has 17.7 cm^3 of free volume although the rod started out with a free volume of 23.3 cm^3. Using the as-manufactured void volume, it is found that there can be no more than 23.3 g of water in one fuel rod. By comparison, 88.2 g of water is required to oxidize the fuel in one fuel rod from UO_2 to U_3O_8. That is as much water as could be supplied by about 3.8 fully waterlogged fuel rods.

In conclusion, waste form degradation from waterlogged fuel rods is excluded from TSPA-SR. There are few failed rods in any WP that could be waterlogged. Since the volume of water inside a rod is quite limited, the water could only affect a small amount of fuel. Fuel rod failure due to rod waterlogging has a low probability of occurrence (credibility), and is excluded from further consideration.

6.2.2 Cladding Degradation Before YMP Receives It - YMP No. 2.1.02.12.00

Related Primary FEPs: No closely related FEPs

FEP Description Certain aspects of cladding degradation occur before the spent fuel arrives at Yucca Mountain. Possible mechanisms include rod cladding degradation during reactor operation, degradation during wet spent fuel pool storage, degradation during dry storage, and rod degradation during shipping (from creep and from vibration and impact) and fuel handling.

Screening Decision: Include (Spent fuel pool damage – Excluded, low probability (credibility)).

Screening Argument: The performance of commercial Zircaloy clad fuel has improved with time. The overall rod failure rate is 0.01 – 0.05 percent. For the TSPA-SR model, CRWMS M&O (2000b, Section 7) gives the CCDF for the fraction of failed rods expected to be received at YMP. In terms of exposing UO_2 fuel pellets to the environment after the waste package (WP) fails, most of the fuel that has failed during reactor operation will still have some protection provided by the remaining cladding.

After discharge from the reactor, the fuel assemblies are stored in open spent fuel storage pools. An International Atomic Energy Agency (IAEA) survey (IAEA 1988, Table XXVI) reported no evidence of fuel degradation in spent fuel pools and no evidence of significant additional degradation of fuel damaged during reactor operation during residence in the spent fuel pool. The oldest fuel in the survey was Shippingport PWR fuel that has been in wet storage since 1959. Other fuel reported to have no significant additional degradation during residence in the spent fuel pool has been in wet storage since 1962, 1966, 1968, and the 1970s. The importance of the spent fuel pool storage experience is that fuel failure or significant further degradation is not expected during pool storage, and the fuel failure rates observed from reactor operation are appropriate for the cladding degradation analysis. This failure mechanism is excluded because of low probability (credibility) because it has not been observed in spent fuel pools.

The DOE has sponsored a Spent Nuclear Fuel Integrity During Dry Storage-Performance Tests Program at the INEEL since 1984. Approximately 26,500 rods have been studied in various commercial dry storage casks. This program also demonstrates what can be expected to happen to fuel in repository WPs during the early thermal period when temperatures are elevated. The cover gas has been monitored and very few rod failures have been observed during dry storage. The importance of the spent fuel dry storage experience (domestic and foreign) is that fuel failure or degradation is expected to be very small during dry storage, and the fuel failure rates are approximated in the cladding degradation analysis by CRWMS M&O (2000b and 2000c). This failure mechanism is included.

During normal shipping of fuel, no failures have been reported in the literature. Failure probabilities have been calculated for a specific drop test (9-m drop) of a cask with impact limiters, a 0.3-m drop, and normal transport (i.e., vibration). Results show that no yielding of the rods should occur below 63g acceleration. Normal transport would result in accelerations much below these values. IAEA surveyed shipping worldwide and reported in 1988 that there had been no major incidents during 30 years experience connected with irradiated fuel transport. The importance of the fuel transportation and handling experience (domestic and foreign) is in demonstrating that fuel failure or degradation during transportation from vibration or impact is expected to be very small, and that the fuel failure rates are appropriately estimated for the cladding degradation analysis.

TSPA Disposition For the TSPA-SR analysis, initial cladding failures from reactor operation, dry storage and transportation has a median value of 0.0948 percent with a range of 0.0155 percent to 1.28 percent for any group (bin) of WPs. In addition, cladding creep failures during dry storage and transportation are integrated into the creep failure analysis for the YMP post-closure clad degradation abstraction. Fuel degradation during spent fuel pool storage is excluded from TSPA-SR.

IRSR Issues: CLST1, CLST3, TSPAI4

References: Einziger, R.E.; Atkin, S.D.; Stellrecht, D.E.; and Pasupathi, V. 1982, Einziger, R.E. and Kohli, R. 1984, EPRI 1997, IAEA 1988, Johnson, A.B., Jr. 1977, Johnson, A.B., Jr.; Bailey, W.J.; Schreiber, R.E.; and Kustas, F.M. 1980, Johnson, A.B., Jr.; Dobbins, J.C.; Zaloudek, F.R.; Gilbert, E.R.; and Levy, I.R. 1987, Manaktala, H.K. 1993, McDonald, S.G. and Kaiser, R.S. 1985, McKinnon, M.A. and Doherty, A.L. 1997, Ravier, G.; Masuy, G.; and Willse, J.T. 1997, Sanders, T.L.; Seager, K.D.; Rashid, Y.R.; Barrett, P.R.; Malinauskas, A.P.; Einziger, R.E.; Jordan, H.; Duffey, T.A.; Sutherland, S.H.; and Reardon, P.C. 1992, Sasaki, S. and Kuwabara, S. 1997, Schneider, K.J. and Mitchell, S.J. 1992, Witte, M.C.; Chun, R.C.; and Schwartz, M.W. 1989, Yang, R.L. 1997, CRWMS M&O 2000b, CRWMS M&O 2000c.

Basis for Screening Decision

The bases for the screening decision are presented below in Sections 6.2.2.1 through Sections 6.2.2.4.

6.2.2.1 Pin Degradation During Reactor Operation - YMP No. 2.1.02.12.01

The performance of commercial Zircaloy-clad fuel has improved with time. Early cores (up to 1985) had a rod failure rate averaging 0.02 percent to 0.07 percent (EPRI 1997, p. 4-1). After 1985, the rod failure rate decreased to 0.006 percent to 0.05 percent. The overall rod failure rate for both time periods is 0.01 percent to 0.05 percent. Manaktala (1993, p. 3-4) shows fuel reliability as a function of calendar year for both PWR and BWR fuel from 1969 through 1976. He shows early PWR cores had failure rates over 1 percent but the rates dropped below 0.1 percent by 1973. The BWR failure rates fell below 0.1 percent after 1975. Yang (1997, Table 2, p. 10), summarized the frequency and type of assembly failures from 1989 through 1995. Yang's failure data are for assemblies, and can be converted to rod failure rates using 2.2 failed rods per failed assembly (EPRI 1997, p. 4-1), and an average of 221 rods per assembly.

Occasionally a specific core will have a higher failure rate. These data are included in the EPRI fuel failure rate, and affect the averages slightly. After a steam generator replacement in 1981, one reactor operator reported 0.26 percent of the rods (32 percent of the assemblies) were damaged by fretting from foreign particles, such as debris from the steam generator replacement (McDonald and Kaiser 1985, pp. 2-4, 2-5). This case of severe core damage shows that a relatively small fraction of the rods was actually damaged. The French (Ravier et al. 1997, Figure 4, p. 34) report annual rod failure rates from 5.5×10^{-4} to 0.0 from 1986 through 1996. Their reactors are PWRs based on the Westinghouse design, and have similar fuel designs as the United States reactors. The Japanese (Sasaki and Kuwabara 1997, pp. 13-14) report failure rates of 0.01 percent for BWRs and 0.002 percent for PWRs. Their BWR design is based on the General Electric (GE) design, and their fuel designs are similar to the United States reactors. Both rates are similar to the current United States observed rates.

Rod damage characteristics in reactor operation (EPRI 1997, pp. 4-2, 4-3) are:

Pinhole and through wall hairline cracks	80 to 90 percent
Intermediate condition	10 to 20 percent

Severe damage	0.04 to 0.9 percent

In terms of exposing UO_2 fuel pellets to the environment after the WP fails, most of the fuel that has failed during reactor operation will still have some protection provided by the remaining cladding.

Reactor operation failed rods are included in TSPA-SR. The details of the analysis are presented in CRWMS M&O 2000b, which also develops the stress distribution used in the creep analysis.

6.2.2.2 Pin Degradation During Spent Fuel Pool Storage -YMP No. 2.1.02.12.02

Rod failure during spent fuel pool storage is excluded because of low probability (credibility) from TSPA-SR. The importance of the spent fuel pool storage experience is that fuel failure or further degradation of rods already failed during reactor operation is not expected during pool storage, and the fuel failure rates observed from reactor operation, dry storage, and shipping are appropriate for the cladding degradation analysis.

After discharge from the reactor, the fuel assemblies are typically stored in spent fuel storage pools. An International Atomic Energy Agency (IAEA) survey (IAEA 1988, Table XXVI) reported no evidence of fuel degradation in spent fuel pools and no evidence of significant additional degradation of fuel damaged during reactor operation. The oldest fuel in the survey was Shippingport PWR fuel that has been in wet storage since 1959. Other fuel reported to have no further degradation has been in wet storage since 1962, 1966, 1968, and the 1970s.

Under the DOE Spent Fuel and Fuel Pool Component Inventory Program, the effect of storing both fuel with intact cladding and fuel with failed cladding has been studied. An international survey of in-water storage (Johnson et al. 1980, p. iii) reports no cases of fuel cladding degradation during pool storage.

Johnson (1977, p. 20), reports: "Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircaloy-clad fuel which developed defects during reactor exposures, e.g., Ginna, Oyster Creek, Nine Mile Point, and Dresden units I and II. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the GE-Morris pool without need for isolation in special cases. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity."

6.2.2.3 Pin Degradation During Dry Storage -YMP No. 2.1.02.12.03

Creep strain damage and pin damage (failure) during dry storage are included in the creep degradation analysis described in CRWMS M&O (2000c, Section 6.2). It is assumed that all fuel undergoes 20 years of dry storage starting at 350°C. The amount of creep strain during dry storage is calculated and added to the creep from shipping and the YMP thermal transient (CRWMS M&O 2000c, Figure 5). In addition, 0.045 percent of the rods are assumed to have failed in dry storage from handling and other causes (CRWMS M&O 2000b, Table 13).

The DOE has sponsored a Spent Nuclear Fuel Integrity During Dry Storage-Performance Tests Program at the INEEL since 1984 (McKinnon and Doherty 1997, pp. 2.1, 5.16). Approximately 26,500 rods have been studied in various commercial dry storage casks. This program also demonstrates what can be expected to happen to fuel in repository WPs during early storage times when temperatures are elevated. The dry storage cask fuel cover gas has been monitored and very few fuel rod failures have been observed during dry storage. During fuel consolidation, approximately 10 rods are believed to have developed small leaks. Some rods leaked, but the release was over a period of approximately two months, a very slow gas release. Only 0.5 percent of an individual rod's ^{85}Kr content was released from the leaking rods. After shipment to the INEEL, the observed rod failure was similar to that after reactor operation. For rods that were not consolidated, 2 rods leaked out of 16,700 rods producing a failure probability of 1.2×10^{-4} per rod, below the 0.045 percent failure rate used in TSPA-SR for failure in dry storage.

Some utilities experimented with fuel consolidation before the introduction of dry storage facilities. After consolidation at INEEL (McKinnon and Doherty 1997, pp. 2.1, 5.16), 12 rods failed out of 9800 (failure probability $=1.22 \times 10^{-3}$), an increase by 1 order of magnitude from the unconsolidated failure rate. Although no utilities are currently using consolidation, the rod failure rate on 0.045% used in TSPA-SR for dry storage contains a component of the failure rate from consolidation.

Dry storage tests were performed at the Nevada Test Site (Johnson et al. 1987, p. iv) with 17 spent fuel assemblies, each in an individual test. These tests contained 3468 rods, and cladding temperatures varied from 168 to 278°C. One of the fuel rods failed during these tests. This assembly was exposed to air at 275°C and had nine thermal cycles. The estimated hole size was 1 μm. No further degradation was observed in this one failed rod after the initial failure. No visible damage was observed in the other tests.

Accelerated high temperature tests were performed on 15 rods (Einziger et al. 1982, pp. 65, 69). Post irradiation studies of failure mechanisms of well-characterized pressurized water reactor rods were conducted for up to a year at 482°C, 510°C, and 571°C in limited air and inert gas atmospheres. No cladding breaches occurred. The cladding had crept away from the pellets and showed a smoother profile. Strains from 1.7 percent to 7 percent were measured (their Table IV). One rod had a local creep strain as high as 12 percent (their Figure 3, p.69). The extended lifetime is attributed to significant creep strain of the Zircaloy cladding, which decreases the internal rod pressure. The cladding creep also contributes to radial cracks through the external oxide layer and internal fuel-cladding chemical interaction layers (layer of a few micrometers where some zirconium/UO_2 interaction occurs), which propagated into and arrested in an oxygen stabilized alpha-Zircaloy layer. Since cracks extended only for a few micrometers and were arrested, their significance was small. There were no signs of either additional cladding hydriding, stress corrosion cracking (SCC), or fuel pellet degradation.

A second series of tests (Einziger and Kohli 1984, pp. 107 and 114, Table III) was performed on five PWR spent fuel rods. They were pressurized to a hoop stress in the range of approximately 145 to 155 MPa, for times up to 2101 hours at 323°C. The conditions were chosen for limited annealing of in-reactor irradiation hardening. With the stresses in the range of 145 MPa, creep of 0.004% to 0.16% was observed. No cladding breaches occurred, although significant hydride agglomeration and reorientation took place in one rod that cooled under stress. Einziger and Kohli (1984, pp. 107 and 114, Table III) state that these high-temperature tests based on creep rupture as the limiting mechanism indicate that storage at temperatures between 400 and 440°C may be feasible for annealed rods.

Schneider and Mitchell (1992, p. 2.7) summarized experience in the foreign dry storage programs. At that time, seven countries had some fuel in dry storage. They conclude that Light Water Reactor (LWR) fuel can be stored for up to 100 years at temperatures of 320 to 400°C in an inert atmosphere and, if exposed to air, will last comparable times if the temperature is limited to 135 to 160°C. The Canadians have large quantities of irradiated fuel with Zircaloy cladding that has been exposed to air with favorable results. For over eight years, they tested the effects of exposing fuel with defective cladding to moist and dry air with favorable results (no observed strain from UO_2 oxidation nor cladding failure propagation).

The importance of the spent fuel dry storage experience (domestic and foreign) is that fuel failure or degradation is expected to be very small during dry storage for the current fuels. It is also important to note that under repository conditions, most of the cladding creep failures would be expected to occur in the first 100 years after closure, this is the time period when fuel temperatures are highest and conditions are closest to dry storage conditions. Dry storage is considered to be a good representation of this period.

6.2.2.4 Pin Degradation During Fuel Shipment and Handling - YMP No. 2.1.02.12.04

During normal shipping of fuel, no failures have been reported in the literature. Sanders et al. (1992, Table III-10, p. III-137), analyzed transportation accidents. Failure probabilities were calculated for a specific drop test of nine meters (9-m), a 0.3-m drop, and normal transport (i.e., normal vibration). The 9-m drop was by far the most severe, with failure probabilities on the order of 2×10^{-4}. The normal transport failure is 2×10^{-7} per rod. This reference generated acceleration versus frequency curves for trucking and rail shipping. It then looked at structural damage from a sampling of the hazard curves and the structural analysis and concluded that no additional damage is done in shipping. This value is conservatively bounded by a failure fraction of 1×10^{-4} to account for other transport accident conditions and is included in the TSPA-SR.

Witte et al. (1989) performed an analytical evaluation of the potential impacts of all transport movements. They concluded (Witte et al. 1989, p. 194, Table 3) that no yielding of the rods would occur below 63g acceleration. Normal transport would result in acceleration far below these values.

IAEA (1988, p. 114) surveyed shipping worldwide and reported that "to date, there have been no major incidents during 30 years experience connected with irradiated fuel transport."

The importance of the fuel transportation handling experience (domestic and foreign) is in demonstrating that very little fuel failure or degradation is expected during transportation from vibration or impact, and that the fuel failure rates observed from reactor operation are appropriate for the cladding degradation analysis.

6.2.3 General Corrosion of Cladding - YMP No. 2.1.02.13.00

Related Primary FEP: 2.1.02.16.00

FEP Description General corrosion of cladding could expose large areas of fuel and produce hydrides.

Screening Decision: Exclude, based on low probability (credibility).

Screening Argument: The omission of general corrosion of cladding will not significantly change the expected annual dose because general corrosion of Zircaloy cladding is very slow and will not be important in the first 10,000 years or even 100,000 years. On the scale of the repository model, no general corrosion of cladding is included in in-package chemistry. When general corrosion needs to be accounted for, it is included as a specific feature in the Localized corrosion (pitting) of Cladding, YMP FEP 2.1.02.16.00 where general corrosion might very well be the cause of localized corrosion and is included. The omission of general corrosion of cladding, as a system wide feature from the performance assessment models will not significantly change the expected annual dose.

General corrosion is synonymous with zirconium oxidation. The outer surface of the cladding becomes oxidized with ZrO_2, providing a protective film that adheres to the surface and slows down further oxidation. The oxidation could be from O_2 consumption (dry oxidation) or H_2O consumption (wet oxidation) and the dry and wet oxidation rates are very similar. For the fuel in the repository, oxidation does not occur until the WP is penetrated. Various analyses have concluded that cladding oxidation under repository conditions would not lead to rod failure. The amount of oxide film resulting from general corrosion as a function of time since WP failure has been calculated but is insignificant if the WP remains sealed for the first 100 years. There was no measurable degradation of the Zircaloy cladding after 21 years of in-water spent fuel pool storage times. The importance of the spent fuel dry storage experience (domestic and foreign) is that fuel degradation from oxidation is expected to be very small during dry storage since the dry storage canisters contain an inert atmosphere.

TSPA Disposition Exclude, low probability (credibility).

IRSR Issues: CLST1, CLST3, TSPAI3, TSPAI4

References: Bradley, E.R.; Bailey, W.J.; Johnson, A.B., Jr.; and Lowry, L.M. 1981, Einziger, R.E. 1994, Hillner, E.; Franklin, D.G.; and Smee, J.D. 1998, IAEA 1998, Rothman, A.J. 1984

Basis for Screening Decision

General corrosion is synonymous with zirconium oxidation. The outer surface of the cladding becomes oxidized with a ZrO_2 film, which adheres to the surface and slows down further oxidation. The oxidation could be from O_2 consumption (dry oxidation) or H_2O consumption (wet oxidation). For the fuel in the repository, this does not occur until the WP is penetrated.

The effect of surface oxidation is threefold. The oxidation could thin the cladding, contributing to cladding failure by creep rupture. Wet oxidation generates hydrogen, and some of the hydrogen is absorbed into the cladding to form hydrides. This could lead to delayed hydride cracking (DHC), or general hydride embrittlement. In the extreme, the oxidation could lead to cladding disintegration and expose the fuel pellets to the WP environment.

Wet oxidation has been studied for over forty years because of its importance in reactor operation. Rothman (1984, pp. 6 - 13) discusses cladding oxidation in repository conditions in great detail and his Table 3 compares the predicted cladding loss using six different oxidation correlations and predicts cladding thinning of 4 to 53 μm after 10,000 years at 180°C (a conservative temperature condition since the repository cools after a few hundred years) which would not lead to rod failure.

Recently, Hillner et al. (1998, p. 9) published a recommended Zircaloy corrosion correlation based on Bettis Atomic Power Laboratory experiments. Bettis developed Zircaloy for Navy reactors in the early 1950s and has an extensive data base on Zircaloy performance. They have continuous autoclave corrosion tests on some samples for 30 years. Some samples have developed oxide thickness as great as 110 μm, greater than those expected during repository corrosion. The experiments show that the corrosion rate is determined by diffusion of oxygen ions through the corrosion film. This corrosion film is generated in layers, with the lower layer staying very consistent. The consistency of the lower 2 μm of oxide film leads to a steady corrosion rate after a transition period. The recommended post-transition rate equation is:

$$\Delta Th = 1.72 \times 10^9 \times \exp(-11452/T) \qquad \text{(Eq. 1)}$$

where:

ΔTh = oxide growth rate, μm/yr

T = temperature, K

This correlation rate equation is similar to the equations developed by others, but predicts a slightly higher corrosion rate. It is taken from Equation 7 of Hillner at al. (1998), doubled for a conservative correction for irradiation (Hillner et al. 1998, pp. 6, 9) and converted to micrometers per year (1μm = 14.7 mg/dm^2) (IAEA 1998, p. 178). The pre-transition rate is slower than the post-transition rate. The effect of irradiation conditioning before beginning the corrosion is to accelerate the corrosion rate for a few micrometers. To be conservative, Hillner et al. doubled the corrosion rate for all time. The correlation shows a strong Arrhenius temperature relationship with the corrosion rate becoming very small below 200°C. This is consistent with the data from Hillner et al. Corrosion tests at 270°C for 8.2 years have produced approximately 4 μm of oxide, while corrosion tests at 360°C have produced films 88 μm thick in 7.8 years (Hillner et al. 1998, p. 25). This equation predicts an oxide thickness of only 0.22 μm at a long term repository temperature of 40°C for a million years. At 80°C, in one million years, the loss would be 14 μm, a small fraction of the cladding thickness (570 μm for a Westinghouse 17 * 17 design).

For general cladding corrosion, the WP must be breached. The amount of oxide film from general cladding corrosion as a function of time since WP failure has been calculated. These calculations were carried out from the WP failure time to one million years. The amount of cladding that is consumed is approximately 57 percent of the oxide thickness because of the volume increase associated with Zircaloy oxidation. The cladding oxidation calculation was done for the design bases (hot) rod in the hottest repository region. If the WP is not breached after 100 years from emplacement, little general corrosion is expected. Hillner et al. (1998, Figure 5) compare the weight gain of the samples in water which correlates to the corrosion rate in water (the correlation used here) with that of steam. The steam corrosion rate is about 30 to 40 percent slower. A steam environment is expected to last for 1000 to 5000 years in the near field. Einziger (1994, p. 556, Equation 14) states that dry oxidation of zirconium is slightly slower than the wet corrosion rate. The conclusion from CRWMS M&O (1998, Section 6.3.1.1.4.1), is that general cladding corrosion is not a problem for fuel cladding in the average WP but could be a problem for fuel cladding in the design WP if the design WP fails before 100 years. Premature WP failure has been eliminated by a combination of design and quality control in manufacturing.

Bradley et al. (1981, p. 38) performed metallurgical examinations of Zircaloy-clad fuel rods from two bundles (0551 and 0074) of the Shippingport PWR Core 1 "blanket" fuel after extended in-water spent fuel pool storage (21 years for 0551, and 16 years for 0074). The oxide film thickness on the Shippingport fuel rods after reactor operation was reported to be an average cladding oxide film thickness of 1.8 μm (0551) and 2.4 μm (0074). After extended in-water spent fuel pool storage, the average cladding oxide film thickness was found to be 1.7 μm (0551) and 2.3 μm (0074)(Bradley et al. 1981, p. 38). The slight disagreement in these values is attributed to differences in measurement technique and experimental error. These results led to the conclusion that no significant change in oxide thickness occurred even after 16 to 21 years of pool storage. This conclusion is further supported by the observation that Zircaloy tube sheets (that had been cut to remove bundle 0551 fuel rods in 1960) stored in water for over 20 years were unblemished and showed no evidence of reaction with water.

6.2.4 Microbiologically Influenced Corrosion (MIC) of Cladding - YMP No. 2.1.02.14.00

Related Primary FEP: 2.1.10.01.00

FEP Description Microbiologically Influenced Corrosion (MIC) of cladding potentially may be a local cladding corrosion mechanism where microbes produce a local acidic environment that could produce multiple penetrations through the fuel cladding.

Screening Decision: Include, for localized effects from microbial activity.

Screening Argument: MIC activity is included as part of the localized corrosion model where aggressive species including suppressed pH (possibly from MIC) fail the cladding. Microbiologically influenced corrosion of cladding by itself is not expected to occur (not probable or credible) because on the scale of the repository model, microbial activity is screened out as a significant bulk process. When microbial activity, as a localized process, needs to be accounted for, it is included as a specific feature in the YMP colloid FEP 2.1.10.01.00 where microbial activity might very well be the cause of localized corrosion and is included. There is no indication that MIC occurs on zirconium metal or alloys. The two major forms of MIC for materials being considered for waste packages are: (1) sulfide attack through the action of sulfate reducing bacteria (SRB) and (2) corrosion induced by organic acids secreted from certain bacteria. With respect to these forms of MIC: (1) SRB do not affect zirconium, and (2) corrosion induced by organic acids is unlikely because of zirconium's tolerance to organic acids and a wide range of pHs.

TSPA Disposition The possibility of local depression of pH by microbes is included in the conservative local corrosion model described in CRWMS M&O 2000a.

IRSR Issues: CLST1, CLST3, TSPAI3, TSPAI4

References: Little, B. and Wagner P. 1996, Yau, T.L. and Webster, R.T. 1987, Hillner, E.; Franklin, D.G.; and Smee, J.D. 1998, McNeil, M., and Odom, A. 1994, CRWMS M&O 2000a

Basis for Screening Decision

While microbial activity has been screened out as a significant bulk process at YMP (colloid FEP 2.1.10.01.00), it has not been excluded for localized effects. There is no experimental evidence of enhanced MIC of zirconium metal or alloys. However, the possibility of local depression of pH by microbes is included in the conservative local corrosion model.

The term microbiologically influenced corrosion (MIC) is commonly used to designate corrosion caused by the presence and activities of microorganisms at the surfaces of metals. Little and Wagner (1996, p. 367) published an overview of microbiologically influenced corrosion of metals and alloys used in the storage of nuclear wastes. They indicate that MIC is a form of localized corrosion that results in pitting, selective leaching, crevice corrosion, underdeposit corrosion, and enhanced erosion/corrosion. Little and Wagner (1996, pp. 367-368) describe several mechanisms for microbiologically influenced corrosion. In addition, various case studies are presented that document microbiologically influenced corrosion of alloys of iron, nickel, and copper. However, it should be noted that there is no indication in the literature that MIC occurs on zirconium metal or alloys. Yau and Webster (1987, p. 709) report that no corrosion of zirconium metal from marine organisms was found during sea water corrosion tests for 129 days.

Hillner et al. (1998, p. 11) indicate that there are two major forms of MIC for materials being considered for WPs. They are (1) sulfide attack through the action of sulfate reducing bacteria and (2) corrosion induced by organic acids secreted from certain bacteria. With respect to attack by SRB, Hillner et al. 1998 reference the work of McNeil and Odom (1994, p. 176). McNeil and Odom (1994, p. 176) indicate by thermodynamic calculations that SRB do not affect zirconium. With respect to corrosion induced by organic acids, Hillner et al. (1998, p. 11) noted that it is most unlikely because of zirconium's tolerance of a wide range of pHs and that it is unlikely that production of weak organic acids will have an adverse effect on the passivation of Zircaloy by a ZrO_2 film. Yau and Webster (1987, p. 717) also note that zirconium resists a wide range of organic compounds, including acetic acid, acetic anhydride, formic acid, urea, ethylene dichloride, formaldehyde, citric acid, lactic acid, oxalic acid, tannic acid, and trichloroethylene. CRWMS M&O 2000a summarize the corrosion potential of zirconium to many chemicals. MIC is included as a component of the localized corrosion model where MIC could cause a localized suppression of the water pH and permit other aggressive species to attack the cladding.

6.2.5 Acid Corrosion of Cladding From Radiolysis - YMP No. 2.1.02.15.00

Related Primary FEPs: No closely related FEPs

FEP Description Radiolysis in a nitrogen/oxygen gas mixture with the presence of water film results in the formation of nitric acid (HNO_3) and hydrogen peroxide (H_2O_2) ions that can enhance corrosion of the fuel cladding.

Screening Decision: Include for local suppression of pH resulting in localized corrosion.

Screening Argument:. Radiolysis activity is included as part of the localized corrosion model where aggressive species including suppressed pH (possibly from radiolysis) fail the cladding. Radiolysis by itself is not expected to damage the cladding (not probable or credible).

Acid corrosion of cladding from radiolysis of water by itself has a low probability (credibility) of occurrence, and is excluded from further consideration. Zirconium has excellent corrosion resistance to nitric acids and concentrated hydrogen peroxide. The current designed WPs do not fail for about 10,000 years. There is no radiolysis until the WP fails and by then the gamma dose is very low. Therefore little nitric acid and concentrated hydrogen peroxide will be produced and zirconium is resistant to these chemicals (CRWMS M&O 2000a, Section 4 and 6).

TSPA Disposition Radiolysis producing local suppression of pH resulting in localized corrosion of cladding is included in the localized corrosion model (CRWMS M&O 2000c) as a specific feature where acid corrosion might very well be the cause of localized corrosion.

IRSR Issues: CLST1, CLST3, TSPAI3, TSPAI4

References: CRWMS M&O 2000a, CRWMS M&O 2000f, CRWMS M&O 1999d, Hansson, C.M. 1984, Van Konynenburg, R.A.; Curtis, P.G.; and Summers, T.S.E. 1998, Yau, T.L. and Webster, R.T. 1987, IAEA 1993, Hillner et al. 1998.

Basis for Screening Decision

It is recognized that radiolysis may create various chemical species that may not otherwise be expected in the repository. Examples are the production of nitric acid as a result of the radiolytic reaction between nitrogen and oxygen and the formation of hydrogen peroxide. In either case, accelerated corrosion of Zircaloy would not be predicted in the absence of irradiation since zirconium alloys have been shown to be relatively inert in both media as discussed in Yau and Webster 1987 and CRWMS M&O (2000a, Sections 6.1.6 and II.4). As one example, zirconium equipment is used in the chemical processing industry where peroxide strengths of 90% are used. The service life has been increased by an order of magnitude compared to graphite components previously used, which were generally considered to be very inert. In nitric acid, zirconium and its alloys are inert up to acid concentrations of 65%. Provided radiolysis does not produce nitric acid at greater concentrations, there will be no impact on the uniform corrosion rate as a result of nitric acid production from radiolysis.

Radiolysis was reviewed in detail in the IAEA 1993 report (pp. 82-92). Specifically the topic of radiolysis of "thick oxide film effects" was discussed. The report noted (IAEA 1993, p. 91) that "the radiation enhancement of Zircaloy corrosion in the aqueous phase arises from the synergistic interaction of radiation and water chemistry. It has been recognized that in the early stages of Zircaloy corrosion the acceleration by reactor radiation, which is usually observed in oxygenated water, is suppressed in the presence of excess hydrogen in the aqueous phase, but that beyond the threshold oxide thickness Zircaloy specimens exposed to low-oxygen water corrode at accelerated rates under irradiation as if they were immersed in oxygenated water." This implies that under the worst conditions of high radiation levels and excess oxygen ions, the corrosion rate is only accelerated by a factor of three. However, an analysis of the gamma and neutron dose in the repository at 1000 years, thus well before the first anticipated breach of the waste package, predicts a total dose of 0.466 rem/hour at the waste package surface (CRWMS M&O 1999d, p. 17). This dose is significantly lower than that in an operating reactor and may in fact be too low to produce sufficient peroxide to produce an accelerated corrosion. In the unlikely event that premature failure of a WP should occur, the waste package temperature would preclude the presence of liquid water, and thus the presence of peroxide, for at least 1000 years. At this time the surface radiation dose will be very low level as noted above, so that the practical likelihood of producing peroxide is extremely low under any conditions.

Yau and Webster (1987, pp. 707-721) review the corrosion of zirconium under various chemical environments for commercial applications (corrosion rates and time scales of interest for industrial applications, not for repository time scales). Zirconium is resistant to corrosion from HCl to temperatures well above boiling (Yau and Webster 1987, p. 710). In basic solutions, Hansson (1984, summary page) measured corrosion of Zircaloy 2 in anaerobic cement pore solutions of pHs of 12.0 to 13.8. Hansson concludes: "Thus, it may be concluded that active corrosion of Zircaloy 2 in anaerobic concrete will not occur and by comparison with measurements on steel, it is likely that the passive corrosion rates will be even lower in concrete than those measured in the synthetic pore solution." Yau and Webster 1997 report no corrosion in seawater, brackish water, and polluted water. Zirconium is resistant to corrosion from sulfuric acids in concentrations less than 20 percent and corrodes slowly in terms of commercial applications (not repository time scales) in sulfuric acid solutions below 65 percent.

Van Konynenburg et al. (1998, p. 7-17) performed container material scoping tests using Zircadyne 702 (a Zirconium hafnium alloy) in 0.01 mol/L each of sodium formate ($NaCOOH$), nitric acid (HNO_3), NaCl, H_2O_2, and 0.02 mol/L sodium oxalate ($Na_2C_2O_4$), with an initial pH of approximately 4, temperature of 90±5 °C, and typical test duration of 96 hours. The test solution was designed to represent highly concentrated forms of acids formed by radiolysis. Formic acid, a potential pH buffer, was not present in the Zircadyne 702 test. They report a final pH of 4.26 and corrosion rate of 50 μm/yr for zirconium in the acid solution.

The duration of highly alkaline conditions in an emplacement drift depends on the amount of concrete used. If concrete (emplacement drift liner material) is not used, the duration of highly alkaline conditions in the emplacement drift is less than 5000 years, with the long-term pH at about 8. If concrete material is used in the liner, the expectation is that the pH in the emplacement drift will be quite basic for the first 10,000 years.

After emplacement of the waste packages, the radioactive decay of the waste will heat the drifts and disturb the normal percolation of water through the mountain. As the drifts cool, some of the ground water percolating through the mountain may drip into the drifts and some of the waste packages. Through time, the components of the engineered barrier are expected to degrade. The metallic materials of the WPs, pedestals, and drip shield are expected to undergo humid air and aqueous corrosion, and the concrete materials, if used, are expected to undergo hydrothermal degradation. As the materials deteriorate, the drift liner will fall onto and around the WP and the pedestals will collapse lowering the WPs toward the floor. The WPs and drip shield are expected to undergo humid corrosion, and if wetted by dripping or high relative humidity, to undergo aqueous corrosion. Corrosion degradation of Zircaloy fuel cladding does not begin until the WP is breached (typically 10,000 years or more after emplacement). A steam environment is expected to last for 1000 to 5000 years in the near field.

Once a WP is breached, water may enter the WP as water vapor or as drips. The WP and fuel will have cooled down by that time to an extent that will allow a water film to form on the WP surface and on the fuel cladding that is thick enough to support corrosion reactions. The radiation dose would also be a lowered by that time to a level that is considered insignificant to long-term engineered barrier performance.

In summary, the WPs are expected to be intact for thousands of years after emplacement. During the first 1000 to 5000 years, the environment around the WPs is expected to be steam. This environment will prevent nitric acid or hydrogen peroxide from pooling on or coating the WP (or fuel cladding) and would thus control acid corrosion. While generalized corrosion of the cladding by radiolysis is not credible, localized corrosion by aggressive species and possible pH suppression is included in TSPA-SR.

6.2.6 Localized Corrosion (Pitting) of Cladding - YMP No. 2.1.02.16.00

Related Primary FEPs: No closely related FEPs

FEP Description: Localized corrosion in pits could produce penetrations of cladding.

Screening Decision: Include

Screening Argument:

Localized corrosion is included as a perforation mechanism in the CSNF Cladding Degradation Component. Corrosion of zirconium has been observed in concentrated fluoride or chloride solutions at very low pHs or very high oxidation potential (CRWMS M&O 2000a, Section 4.1). These conditions are not predicted to occur in the bulk solution as discussed in *Summary of In-Package Chemistry for Waste Forms*, CRWMS M&O 2000f (entire report). Localized corrosion has not yet been ruled out for localized and/or non-equilibrium effects such as MIC, galvanic coupling, radiolysis in a humid environment, and extreme concentration by evaporation. Each of these mechanisms may locally depress the pH or increase the concentration of corrosive species such as fluoride or chloride, at least temporarily. However, until they can be ruled out, or shown to be too transitory to have negative consequences on cladding, a conservative model has been adopted. The model is described in *Clad Degradation – Summary and Abstraction*, (CRWMS M&O 2000c, Section 6.3).

TSPA Disposition Localized corrosion is included as a perforation mechanism in the CSNF Cladding Degradation Component as described in CRWMS M&O 2000c.

IRSR Issues: CLST1, CLST3, TSPAI3, TSPAI4

References: CRWMS M&O 2000a [ANL-EBS-MD-000012 REV 00], CRWMS M&O 2000f [ANL-EBS-MD-000050 REV 01], CRWMS M&O 2000c [ANL-EBS-MD-000007 REV 00]

Basis for Screening Decision

This FEP is included.

6.2.7 Localized Corrosion (Crevice Corrosion) of Cladding - YMP No. 2.1.02.17.00

Related Primary FEPs: No closely related FEPs

FEP Description Localized corrosion in crevices could produce penetrations of cladding.

Screening Decision: Exclude, based on low probability (credibility).

Screening Argument: Localized corrosion of the cladding through crevice corrosion has been excluded on the basis of low probability (credibility) of occurrence because zirconium and its alloys' do not corrode through crevice corrosion. The CSNF Cladding Degradation Component excludes a component that accounts for localized corrosion of the cladding through crevice corrosion because zirconium does not corrode in this manner. This FEP is the topic of AMR, Clad Degradation-Local Corrosion of Zirconium and its Alloys Under Repository Conditions CRWMS M&O 2000a. Zirconium is is not susceptible to crevice corrosion. CRWMS M&O 2000a discusses the crevice corrosion resistance of zirconium in various chemical solutions and in Section 4.1.3 summarizes seven (7) crevice corrosion tests and reports that crevice corrosion was not observed. The U-bend tests discussed in Section 4.1.4 are also designed to produce crevice corrosion testing under the U-bend test washers. In these tests, no crevice corrosion is reported. Section 6.1.10 discusses the theoretical reasons why zirconium is immune to this type of corrosion. Yau and Webster 1987 (p. 717) report that zirconium is among the structural metals that are most resistant to crevice corrosion. Greene et al. 2000 (p. 7) also reported no crevice corrosion. In summary crevice corrosion has not been observed in zirconium and is not expected at YMP conditions. Because of this low probability of occurrence, it has been excluded from further consideration.

TSPA Disposition This FEP is addressed in CRWMS M&O 2000a.

IRSR Issues: CLST1, CLST3, TSPAI3, TSPAI4

References: Yau and Webster 1987 ,CRWMS M&O 2000a ,Greene et al. 2000, CRWMS M&O 2000a

Basis for Screening Decision: Excluded based on screening argument presented above.

6.2.8 High Dissolved Silica Content of Waters Enhances Corrosion of Cladding - YMP No. 2.1.02.18.00

FEP Description It must be determined if the high dissolved silica content of waters enhances corrosion of cladding.

Screening Decision: Exclude, based on low probability (credibility).

Screening Argument: Enhanced corrosion of cladding due to high dissolved silica content of waters is not a credible occurrence because the potential for silica itself degrading the cladding is negligible; it is excluded from further consideration. Hansson (1984) reports corrosion tests with concrete pore fluids, which normally contain silica. Yau (1983) reports corrosion tests in seawater, which also contains silicon. Neither experimenters report significant corrosion. Both CRWMS M&O 2000a and Yau and Webster (1987, pp. Table 6) review the corrosion potentials for zirconium and show that it is resistant to most chemicals.

Silica is very stable. It is practically insoluble in water or acids, except hydrofluoric acid. It is not corrosive to most materials. However, there may be some fluoride contamination in silica, in which case it becomes corrosive to some materials, including zirconium, because fluorides in silica are soluble in water. For information on fluoride see YMP No. 2.1.02.16.00, Section 6.2.6.

TSPA Disposition Silica content is not an issue, but the fluoride content that is in the silica is an issue. The fluoride issue is discussed in YMP No. 2.1.02.27.00. Therefore this FEP is excluded from further evaluation based on the arguments described below.

IRSR Issues: CLST1, CLST3, TSPAI3, TSPAI4

References: CRWMS M&O 2000a, Yau and Webster 1987, Hansson 1984, Yau 1983.

Basis for Screening Decision

Silica content is not an issue, but the fluoride content that is in the silica is an issue. Although frequently used as a corrosion resistant coating on materials, silicon dioxide reacts with hydrofluoric acid to form fluosilicic acid. Since fluosilicic acid is highly corrosive to zirconium even at room temperature, silicon dioxide can not be used as a technique to prevent attack of zirconium by acidic fluorides. The fluoride corrosion itself is addressed in another FEP (YMP No. 2.1.02.27.00) and is included. The potential for silica itself degrading the cladding is negligible. Hansson (1984) reports corrosion tests with concrete pore fluids which normally contain silica. Yau (1983) reports corrosion tests in sea water which also contains silicon. Neither experimenters report significant corrosion. Both CRWMS M&O 2000a and Yau and Webster (1987, pp. Table 6) review the corrosion potentials for zirconium and show that it is resistant to most chemicals.

6.2.9 Creep Rupture of Cladding - YMP No. 2.1.02.19.00

Related Primary FEPs: 2.1.11.07.00

FEP Description At high temperatures (>350°C) for sufficiently long time intervals, creep rupture of Zircaloy cladding on spent fuel can occur and produce small perforations in the cladding to relieve stress. After the WP fails, the fuel can react with water and radioisotopes can thereby escape over time from the fuel rod.

Screening Decision: Include.

Screening Argument: Only moderately high temperatures (<350°C) are anticipated on the cladding surface and thus, creep rupture would not be a failure mechanism under storage conditions. Failure of Zircaloy cladding by creep rupture is included in the CSNF Cladding Degradation Component of the Waste Form Degradation Model as a perforation mechanism. Inclusion of the cladding creep process permits the TSPA-SR to evaluate, statistically, the potential for a small number of cladding ruptures because of a wide distribution of temperatures that may occur in the repository or during dry storage before receiving the fuel at the repository and a distribution of initial stresses that may be present in the spent fuel rods. Inclusion also permits an objective assessment of the cladding creep failures during storage and transportation prior to receipt at the repository.

TSPA Disposition The manner in which creep rupture could degrade cladding is discussed in detail in CRWMS M&O (2000c, Section 6.2). In summary, to estimate the percent of the rods that will perforate from creep rupture, the amount of creep is calculated for a temperature profile that includes a dry storage and repository temperature history. The WP surface is treated as an independent variable. The creep is then compared to a creep failure criteria and if it is exceeded, the rod is failed. This analysis is repeated for a statistical distribution of fuel rods and conditions.

The FEP Thermally-Induced Stress Changes in Waste and EBS (YMP No. 2.1.11.07.00) is included in the TSPA-SR in that the stress in the fuel rods is adjusted for the temperature at each time and rod location when analyzing both creep damage and the potential for SSC.

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 2000c

Basis for Screening Decision

No further elaboration in addition to the *Screening Argument* is necessary.

6.2.10 Pressurization From Helium Production Causes Cladding Failure - YMP No. 2.1.02.20.00

Related Primary FEPs: 2.1.12.02.00

FEP Description Increased pressure within the fuel rod due to the production of helium gas could contribute to cladding failure.

Screening Decision: Include.

Screening Argument: Both delayed hydride cracking (DHC) and strain failures are driven by the cladding stress which may be caused by the internal gas (including initial fill gas, fission product gases, and helium gas from alpha decay) pressure buildup. The gas pressure will slowly increase over time by the production of helium as a result of alpha decay.

TSPA Disposition This FEP is addressed in CRWMS M&O (2000b, Section 6.3.4). Both delayed hydride cracking (DHC) and strain failures are driven by the cladding stress which may be caused by the internal gas (including initial fill gas, fission product gases, and helium gas from alpha decay) pressure buildup. The gas pressure will slowly increase over time by the production of helium as a result of alpha decay. Manaktala (1993, Figure 3-4, p. 3-12) presents the helium pressure buildup for 100 °C as a function of time for a PWR fuel rod with 36 MWd/kgU burnup and an 100% helium release from the fuel into the fuel rod gap. This figure was used to develop an equation for helium buildup in a fuel rod. In TSPA-SR, the creep analysis was performed including the helium buildup at 100 yr.

IRSR Issues: CLST1, CLST3, TSPAI4

References: Manaktala, H.K. 1993, CRWMS M&O 2000b

Basis for Screening Decision

No further elaboration in addition to the *Screening Argument* is necessary.

6.2.11 Stress Corrosion Cracking (SCC) of Cladding - YMP No. 2.1.02.21.00

Related Primary FEPs: No closely related FEPs

FEP Description Stress corrosion cracking mechanisms can contribute to cladding failure. These mechanisms can operate both from the inside out from the action of fission products, or from the outside in from the actions of salts or other chemicals within the waste package.

Screening Decision: Include.

Screening Argument: SCC requires a susceptible material, an aggressive chemical environment, and high stress levels. Iodine-induced SCC requires an iodine concentration in the fuel-cladding gap greater than 5×10^{-6} g/cm^2 (Cunningham et al. 1987, pp. A.4, A.5). Below this threshold of free iodine concentration, Zircaloy cracking due to SCC has not been observed. In actual fuel rods, free iodine concentrations are expected to be negligible. However, over sufficiently long times at high stress and elevated temperatures the iodine may be present in sufficient quantities for SCC, such that, once cracking starts, there is sufficient time to propagate through the cladding (without considering crack velocities). As reported by Tasooji et al. (1984, p.600, their Figure 3), a rod with a stress above 180 MPa fails by SCC. Although few rods have such high stresses and the same rods that fail from SCC are also prone to fail from creep, the inclusion of SCC slightly increases the failure from creep alone for dry storage and transportation (CRWMS M&O 2000c).

TSPA Disposition The SCC model (CRWMS M&O 2000c) compares the existing stress with a SCC threshold stress and considers neither the crack size, chemical environment, nor the crack location. Therefore this model is applicable and its inclusion is discussed for Inside-Out SCC in FEP YMP No. 2.1.02.21.01 or Outside-In SCC in (FEP YMP No. 2.1.02.21.02.

IRSR Issues: CLST1, CLST3, TSPAI4

References: Cunningham, M.E.; Simonen, E.P.; Allemann, R.T.; Levy, I.S.; and Hazelton, R.F. 1987, Tasooji, A.; Einziger, R.E.; and Miller, A.K. 1984, CRWMS M&O 2000c

Basis for Screening Decision

No further elaboration in addition to the *Screening Argument* is necessary.

6.2.12 Hydride Embrittlement of Cladding - YMP No. 2.1.02.22.00

Related Primary FEPs: No closely related FEPs

FEP Description Cladding contains hydrogen after reactor operation. The cladding might also pick up more hydrogen from cladding general corrosion (wet oxidation) after the WP is breached. The hydrogen can exist both as zirconium hydride precipitates and as hydrogen in solid solution with zirconium. Hydrides might also form from UO_2 oxidation (after WP and cladding perforation). In addition, hydrides may dissolve in warmer areas of the cladding and migrate to cooler areas. Hydrogen can also move from places of low stress to places of high stresses, causing hydride reorientation or delayed hydride cracking (DHC). The buildup of hydrides can cause existing cracks to propagate by DHC or hydride embrittlement.

Screening Decision: Exclude based on low probability (credibility).

Screening Argument: Hydride embrittlement of cladding is omitted on the basis of low probability (credibility) of occurrence because experimental data indicate that the in-package environment and cladding stresses are not conducive to hydride cracking and embrittlement.

Hydrogen will be generated in the WP as the WP internals corrode. This hydrogen is not expected to be absorbed directly by the fuel cladding, because the H_2 molecules are not expected to migrate through the high-density ZrO_2 layer on the fuel cladding. Available data on zirconium hydriding indicate that corrosion of waste package internals will not result in hydriding of fuel cladding.

Although hydriding of Zircaloy cladding as a result of galvanic corrosion has been observed, Zircaloy cladding alloys resist galvanic corrosion when contacting carbon steel, and, if such a galvanic corrosion effect should occur, corrosion of the carbon steel would quickly break the electrical contact between the two materials.

Failure of the cladding by hydride reorientation is unlikely. The cladding material will maintain sufficient strength even if hydride reorientation did occur such that failure would not be expected.

Failure of the cladding by DHC is unlikely and has not been included in the abstraction for the TSPA-SR. Stresses (and stress intensity factors) are too low for crack propagation.

Failure of the cladding by hydrogen embrittlement is unlikely. Hydrogen absorption from fuel cladding surface oxidation and WP corrosion will be negligible. Hydrogen migration will be limited at the temperatures expected during disposal. Because of the negligible corrosion expected from hydride embrittlement, there is a low probability (credibility) of occurrence that the cladding will fail due to hydride embrittlement.

TSPA Disposition None

IRSR Issues: CLST1, CLST3, TSPAI4

References: Baker 1992, ASTM B811, ASTM B851, Clayton, J. C. 1984, CRWMS M&O 1997, Cunningham, M.E.; Simonen, E.P.; Allemann, R.T.; Levy, I.S.; and Hazelton, R.F. 1987, Dieter, G.E. 1961, Einziger, R.E. et. al., 1982, Einziger, R.E. and Kohli, R. 1984, Garzarolli et al. 1979, IAEA 1998, Kreyns, P.H.; Bourgeois, W.F.; White, C.J.; Charpentier, P.L.; Kammenzind, B.F.; and Franklin, D.G. 1996, Lanning, D.D.; Beyer, C.E.; and Painter, C.L. 1997, Mahmood, S.T.; Farkas, D.M.; Adamson, R.B.; and Etoh, Y. 2000, McEachern, R.J. and Taylor, P. 1997, McMinn, A.; Darby, E.D.; and Schofield, J.S. 2000, Peehs, M. and Fleisch, J. 1986, Pescatore, C.; Cowgill, M.G.; and Sullivan, T.M. 1990, Puls, M.P. 1988, Reed-Hill, R.E. 1973, Rothman, A.J. 1984, Shi, S.Q. and Puls, M.P. 1994, Wasywich, K.M. and Frost, C.R. 1992, Yau, T. L., 1984, Yau, T.L. 1983, YMP 1998, Clayton 1989, CRWMS M&O 2000b, CRWMS M&O 2000g, Smith 1966, Huang 1995, Peehs 1998, Paurbaix 1974, Mardon et al. 1997, ASM 1992.

Basis for Screening Decision

Hydrogen embrittlement results in a generally reduced resistance to fracture. In Zircaloy, hydrogen embrittlement is normally caused by formation of microscopic flakes of zirconium hydride. Since the hydride flakes are quite brittle, a crack can propagate more readily by preferentially following the hydrides. Resistance to fracture (fracture toughness K_{IC}) is a measure of resistance to crack propagation through the material. Fracture toughness is typically measured in terms of the critical stress intensity factor, that is, the value of the stress intensity factor that will cause growth of a crack. The stress intensity factor is proportional to the far-field stress times the square root of the crack length. Kreyns et al. (1996, p. 767, his Figure 5, reproduced here as Figure 1) show that for both irradiated and unirradiated material, such hydrides could decrease the fracture toughness (K_{IC}) from 42 MPa·m $^{0.5}$ to 8 MPa·m$^{0.5}$ as the hydrogen content increases from zero to 4000 ppm. As shown in CRWMS M&O 2000b (Section 6.10.2) the maximum stress intensity (K_I) for the statistical distribution of rods and crack sizes at 260°C is 2.7 MPa m$^{0.5}$ and therefore even with hydride concentrations of 4000 ppm, failure is not expected. In the limit, 100 percent hydride and no metal at all, the fracture toughness is about 1 MPa·m$^{0.5}$. The outer surface of the cladding could be fairly brittle (hydrogen content greater than 800 ppm) but much of the cladding thickness has a reasonable toughness.

6.2.12.1 Hydride Embrittlement from Zirconium Corrosion (of Cladding)- YMP No. 2.1.02.22.01

This FEP has been excluded from further consideration based on low probability (credibility) of occurrence because even the hottest rods will pick up only small quantities of hydrogen from cladding surface general corrosion if the WP remains sealed for the first 100 years. Additional hydrides could be formed by general corrosion of the cladding. WPs are failed at various times, permitting water or steam to enter the failed WP. Wet oxidation (or steam oxidation if the local temperature is above boiling) occurs, and approximately 17 percent (Lanning et al. 1997, Vol. 1, p. 8.4, Figure 8.2) of the hydrogen released from the water is absorbed by the fuel cladding in the failed WP. However, the fuel cladding picks up very little hydrogen because the corrosion rate is so slow. Even the hottest rods will pick up only small quantities of hydrogen from cladding surface general corrosion if the WP remains sealed for the first 100 years.

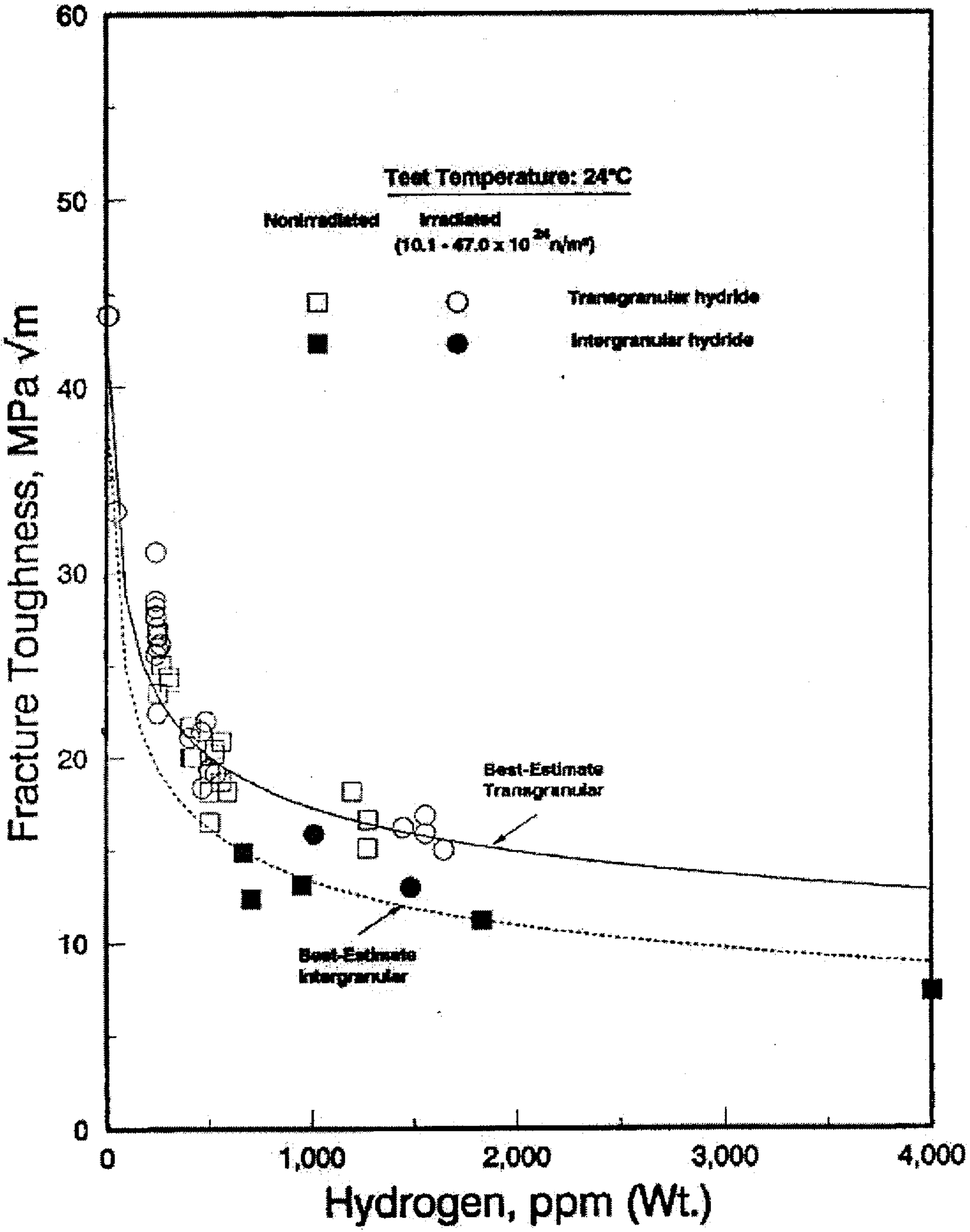

Figure 1. Fracture Toughness vs. Hydrogen Content of Zircaloy-4 (Kreyns et al. 1995)

6.2.12.2 Hydride Embrittlement from WP Corrosion & H_2 Absorption (of Cladding) - YMP No. 2.1.02.22.02

Cladding failure due to this FEP was excluded due to a low probability of occurrence (credibility) because hydrogen generated in the WP as the WP internals corrode is not directly absorbed by the Zircaloy cladding, and hydride embrittlement will not occur. Many investigators have considered the hydriding of zirconium alloys (for example, the papers cited in Clayton 1989, Tables 1 through 4). Many of these investigations have been straightforward measurements of the rate of hydriding under various conditions. However, at least one set of experiments directly determined the origin of hydrogen in the metal. IAEA 1998 (p. 92) discusses experiments in which zirconium-base alloys were oxidized in normal water (H_2O) with dissolved tritium gas (T_2). This experiment is extremely sensitive. If even one part in 10^6 of the hydrogen was from dissolved gas, the radioactivity of T_2 would result in thousands of decays per second for one square centimeter of surface. This level of activity would be readily detected. IAEA 1998 (p. 92) gives the following discussion of the experiment:

> Oxidation studies using T_2/H_2O mixtures ... have shown that, during normal oxidation, no T_2 enters the metal ... until the thermally-induced exchange reaction has progressed to the point where a measurable fraction of HTO has been formed. Thus, the hydrogen isotopes which enter the metal do so as an integral part of the reaction of the zirconium with water molecules, and not by reaction with any dissolved hydrogen in the water. Studies have shown that this situation persists ... until hydrogen over-pressures in the system of tens of MPa are present.

Note that, according to the quotation given above, the hydrogen pressures required to cause hydriding are quite large. By comparison, the highest credible hydrogen pressure in a breached waste package is pure hydrogen gas at atmospheric pressure, or about 0.1 MPa.

It is instructive to examine why high hydrogen pressures can cause hydriding. Even at 200 °C, the solubility of oxygen in zirconium is greater than 1% by weight, and the solubility of oxygen increases with increasing temperature (Baker 1992, p. 2-326). As a result, the oxide film on the surface of a piece of zirconium is normally not thermodynamically stable. In its discussion of experimental procedure, Smith (1966 p. 325) notes "Zirconium samples were first annealed at 700° C under vacuum (~ 10^{-3} mm Hg) to remove any oxide film. The film dissolved into the sample, leaving them a bright metallic color." It is clear that the oxide film can be damaged or even destroyed by heat treatment in a suitable environment. However, the film can be maintained if there is a supply of oxygen. Water will serve as a source of oxygen, because the electrochemical domain of stability for zirconium metal lies well below that of water (Pourbaix 1974, p. 226).

The effect of the oxygen supply has been studied. Garzarolli et al. (1979, p. 64) state that:

> the effect of the composition of the gas atmosphere on the electrical properties of ZrO_2 corrosion films was measured ... . The results revealed a large decrease in the electric resistance when the atmosphere changed from oxidizing to non-oxidizing, indicating a drastic change of the morphology (passivity of the oxide film) ... the obvious implication of all available results is that massive hydriding can start when the availability of oxygen to continuously repair the protective oxide film falls below a critical value.

The conditions for hydriding have been quantified. Clayton (1989, p. 270) presents the equation

$$(p_{H_2O})\text{protective} \cong 0.2(p_{H_2})^{1/3} \qquad \text{(Eq. 2)}$$

where p_{H_2O} and p_{H_2} are the pressures of H_2O and H_2, respectively. For Equation 2 to be applicable, both pressures must be given in torr (millimeters of Hg head). (This fact is deduced as follows. First, Clayton (1989, Tables 1 through 4) indicates that pressures are measured in "mm". Second, pressures of "760 mm" (of Hg) (= atmospheric pressure) occur many times in these tables. Third, the values of "Critical p_{H_2O}/p_{H_2}" given in the tables are consistent with both values being in the same units.) For p_{H_2} = 101 kPa = 760 torr, it is found that p_{H_2O} (protective) = 0.24 kPa = 1.8 torr. Note that, by atmospheric standards, this p_{H_2O} corresponds to a dry gas. For comparison, the vapor pressure of water at 25 °C is about 3.2 kPa. For this small amount of humidity, corrosion of the waste package internals is not credible. However, if no corrosion occurs, no hydrogen is produced, so hydriding is impossible.

It should be noted that hydriding of zirconium by absorption of gas has been observed in the laboratory. For example, Smith 1966 (Table 3) gives data on hydrogen absorption. However, the environment for these experiments was hot, extremely pure hydrogen. Smith 1966 (Table 3) states that the temperatures for the hydrogen absorption experiments were 210 °C to 700 °C. Smith 1966 (p. 325) notes that "hydrogen was purified by passing it through a Deoxo unit, a bed of platinized asbestos (300° C), a tube of P_2O_5 and a liquid nitrogen trap." The evident intention is to react any oxygen impurities, and absorb or condense any water vapor that is formed. After this treatment, very little oxygen would have been available to maintain the oxide film. Such an environment is not relevant for a repository at Yucca Mountain because air, water vapor, or liquid water will be present and will maintain the protective oxide film.

It is understood that corrosion of waste package internals will occur at temperatures below the range for which the Equation 2 was developed. Data on hydriding at lower temperatures were not available in the literature.

In some respects, it can be argued that it is conservative to apply Equation 2 to a repository. Equation 2 was developed for coupons of Zircaloy-2. This alloy is more susceptible to hydriding than is Zircaloy-4 (Clayton 1989, Table 5), and coupons are more susceptible to hydriding than is tubing (Clayton 1989, Table 5). Therefore, Equation 2 should give conservative predictions of the susceptibility of spent fuel cladding to hydriding.

In summary, hydrogen will be generated in the WP as the WP internals corrode. This hydrogen is not directly absorbed, because the H_2 molecules do not migrate through the high-density ZrO_2 layer on the cladding. Hydrogen is introduced into the coolant in PWRs to reduce oxygen ions and reduce corrosion of components and yet it is not absorbed into the cladding. This degradation mode is excluded from TSPA-SR.

6.2.12.3 Hydride Embrittlement from Galvanic Corrosion of WP Contacting Cladding - YMP No. 2.1.02.22.03

This FEP has been excluded due to low probability (credibility) of occurrence because corrosion of waste package internals will not result in hydriding of fuel cladding. Although hydriding as a result of galvanic corrosion has been observed, cladding alloys resist such corrosion on contact with carbon steel, and, if such an effect should occur, corrosion of the carbon steel would quickly break the electrical contact between the materials. In current waste package designs, the fuel assemblies will be in contact with carbon steel tubes (CRWMS M&O 1997). Since the fuel cladding and WP basket tubes are of different materials, there may be galvanic corrosion with hydrogen charging of the fuel assembly. This would occur if there were electrical contact between the assembly and the fuel basket tube, oxidation of iron from the carbon steel fuel basket tubes, and reduction of hydrogen ions on the fuel assembly cladding. In addition, it would be necessary that at least the metal of the fuel assembly cladding absorb some of the reduced hydrogen.

It should be noted that there is only a limited period during which galvanic corrosion could occur in a specific WP. The fuel basket tubes will be completely corroded away within tens to a few hundreds of years after individual WP failures (YMP 1998, p. c-17). By the time the fuel basket tubes are completely corroded away in a specific failed WP, a thick layer of corrosion products will electrically insulate the fuel assembly cladding.

It is important to understand the conditions for galvanic hydrogen charging. These can be inferred from results given in Yau (1983, p. 26/10). Yau discusses a series of experiments in which zirconium alloy U-bend samples were exposed to boiling seawater for 365 days. Each sample was loaded by an uninsulated steel coupling. Three compositions were considered: Zr 702, Zr 704 with nickel, and Zr 704 without nickel. Each composition was tested in both unwelded and welded conditions. Of the six samples, only the welded sample of Zr 704 with nickel showed hydrogen pickup. ("Zr 702" and "Zr 704" are taken to be UNS R607024 and R6070, respectively. ASTM B551/B551M-97 (Table 1) gives composition limits for these materials; the maximum hydrogen content is 50 ppm.) Except for the welded sample of Zr 704 with nickel, the U-bend samples had hydrogen contents of 5 to 9 ppm at the end of the test. It is apparent that contact with a steel surface alone does not lead to galvanic hydrogen charging.

Although Yau (1983, p. 26/2) does not give the compositions of the samples of Zr 704 with nickel and Zr 704 without nickel, it is instructive to compare the amount of nickel in various zirconium alloys. According to the minimum concentrations for other elements, the specification for UNS R60704 admits up to 1.3% nickel, and that for UNS R60702 admits up to 0.8% nickel (ASTM B551, Table 1). UNS R60802 (Zircaloy-2) contains 0.03% to 0.08% nickel, and UNS R60804 (Zircaloy-4) contains no more than 0.007% nickel (ASTM B811-90, Table 2). It is clear that the Zircaloys contain very small amounts of nickel, so hydriding as a result of contact with the steel basket tubes is not expected. It might be argued that, because of its nickel content, Zircaloy-2 cladding could be hydrided. However, Zircaloy-2 fuel cladding is used only for boiling water reactors, and most of the fuel cladding will be separated from the basket tubes by the fuel channels.

Hydriding of zirconium alloys as a result of contact with dissimilar materials has been observed, so it is important to consider the conditions that promote hydriding. Clayton (1984, p. 578) lists a series of experiments in which Zircaloy-4 fasteners, clamped onto a sample of nickel-base alloy were hydrided by exposure to hot water with dissolved hydrogen. Clayton (1984, p. 573) discusses the mechanism:

> ... in mechanically attaching a Zircaloy fuel rod fastener to an Inconel support plate, relative motion occurred during assembly. The relative motion was sometimes sufficiently severe and the bearing stress sufficiently high to overcome the protective effects of the corrosion oxide film and graphite lubricant. Smearing and bonding of Inconel onto local regions of the mating Zircaloy contact surface occurred and provided the potential for accelerated hydriding. Inconel has a relatively high permeability for hydrogen ... and acts as a "window" for hydrogen entry into the Zircaloy.

A similar but very localized effect was noted for Zircaloy-4 smeared with Inconel alloy (4.6% nickel – 16.7% chromium – 3.5% copper – 75.2% iron).

Although hydriding was observed in the experiments discussed above, it does not follow that hydriding will occur under repository conditions. As is noted in Clayton (1984, p. 587), "Nickel alloy smearing and bonding to filmed Zircaloy, rather than just tight surface contact, is necessary for accelerated hydriding." Fuel assemblies will be lowered carefully into waste package baskets. The high contact pressures typical of fastener tightening are not expected. After emplacement, the fuel assembly will simply rest on the basket. Therefore, smearing and bonding should not occur.

If it is nevertheless supposed that there is some bonding between the fuel assembly and the basket, the significance of the effect must be examined. It should be noted that accelerated hydriding is a transient, not a persistent, effect. Clayton (1984, p. 572) notes that the initial high accelerated hydrogen ingress rate was effectively shut off during exposure at 271 °C in about 25 days. Because of the extremely small size of contact spots, corrosion of the carbon steel would quickly break the electrical contact between the two materials. Since the time of contact will be small, the amount of hydrogen absorbed as a result of contact will also be small. This conclusion is supported by the discussion of U-bend hydriding above.

6.2.12.4 Delayed Hydride Cracking (of Cladding) - YMP No. 2.1.02.22.04

Failure of the cladding by DHC in dry storage has been excluded from the TSPA-SR analysis due low probability (credibility) of occurrence because stresses (and stress intensity factors) are too low for crack propagation. Since the threshold for embrittlement failure is always greater than the threshold for DHC ($K_{IC} > K_{IH}$), embrittlement failures are also not expected. CRWMS M&O (2000g, Section 6.3) reviews DHC and CRWMS M&O (2000b, Section 6.10.2) analyzes DHC with the predicted crack size distribution and rod stress distribution expected at YMP. The general DHC process begins with precipitation of a hydride at the crack tip. The stress gradient at the crack tip provides a driving force for preferential hydride precipitation in the tensile stress field in front of the crack tip. If the crack tip stress is greater than the fracture stress of the brittle hydride, the hydride fractures and crack growth occurs.

There are three basic requirements for DHC: (1) incipient defects or cracks; (2) presence of hydride at or near the crack tip; and (3) sufficient stress to propagate the crack. Incipient cracks and defects are present in the cladding as a result of the manufacturing process and from irradiation. Hydrogen is present as a result of Zircaloy alloy impurities and in-reactor corrosion. Cooling of the fuel cladding after reactor operations (irradiation) promotes precipitation of hydrides. The high stresses required to crack a hydride exist only in front of a deep crack. The critical stress intensity for DHC is much greater than cladding stress intensities expected during dry storage. Additionally, at the high stress intensities required for DHC, SCC is the dominant crack growth mechanism.

During delayed hydride cracking, hydrides slowly form at a crack until the crack propagates through the hydride region at the crack tip and stops. This sequence repeats itself and the crack propagates slowly through the metal. The hydrides preferentially collect at the crack tip because tensile stress reduces solubility in that region. The critical stress intensity factor (K_{IH}) is the minimum stress intensity that will permit any DHC, regardless of velocity (velocity approaches 0). For this analysis (CRWMS M&O 2000a), the stress intensities (K_I) will be calculated and compared to the K_{IH}. If $K_I > K_{IH}$, then the crack will start to propagate and, because of long repository times, failure will occur. DHC failure occurred in some zirconium coolant tubes in a Candu reactor where high temperature gradients caused excess hydride buildup in a specific location.

The stress intensity factor, K_I, is a measure of the increased stress at the tip of a crack. The stress intensity factor is proportional to the far-field stress times the square root of the crack length. For a sharp crack, a limiting case, the stress intensity factor is (Reed-Hill 1973, p. 800):

$$K_I = \sigma\sqrt{\pi w} \qquad \text{(Eq. 3)}$$

where

K_I = Stress intensity factor, MPa·m$^{0.5}$
σ = Cladding stress, MPa
w = Crack depth, m

Equation 3 is slightly modified from the form given by Reed-Hill; the crack depth w is used in place of $c/2$, where c is the crack length (Dieter 1961, p. 194).

DHC is analyzed in CRWMS M&O (2000b, Section 6.10.2). The calculated crack size distribution is given in CRWMS M&O (2000b, Section 6.6, Figure 17). The median (P = 50 percent) value is 13 μm and the average crack is 18.6 μm. The largest size crack of the 2000 samples is 119 μm. The calculated stress distribution (CRWMS M&O 2000b, Section 6.7) is given in Figures 18 and 26 of that AMR. DHC is unlikely at temperatures above 260°C (Mahmood et al. 2000, p.139), because of the plasticity of the material. Rothman (1984, p. 37) reports that DHC is unlikely above 250°C because of the plasticity of the material. For this calculation, the temperature of 260°C is used and the pressure is adjusted accordingly. The crack size distribution and stress distribution can be combined to give the distribution of stress intensity factors, K_I, in Figure 27 of that AMR. Some of the properties of this distribution of the stress intensity factor are:

Mean:	0.47 MPa-$m^{0.5}$
Median:	0.40 MPa-$m^{0.5}$
95%:	0.097 MPa-$m^{0.5}$
5%:	1.078 MPa-$m^{0.5}$
maximum:	2.7 MPa-$m^{0.5}$
minimum:	1.61E-03 MPa-$m^{0.5}$

The work of Shi and Puls (1994, p. 239, Fig. 7), shows experimental K_{IH} in the range of 5 to 12 MPa-$m^{0.5}$ for zirconium alloy containing 2.5% Nb. Rothman (1984, p. 37), reports a K_{IH} of 6 MPa-$m^{0.5}$ for Zircaloy-2. Pescatore et al. (1990, Table 6, p.50) report values of 5 and 14. Huang (1995, p. 195) shows K_{IH} for irradiated Zircaloy-2 approaching 6 MPa-$m^{0.5}$. For this AMR, Huang's and Rothman's value for irradiated cladding of 6 MPa-$m^{0.5}$ was used. Because the observed values of K_I are well below these values, exactly which value of K_{IH} is selected is not important. No K_I values in this AMR's sampling of rods are near the threshold stress intensity value. The maximum observed K_I was 2.7 MPa-$m^{0.5}$ and the mean value was 0.47 MPa-$m^{0.5}$.

Rothman (1984, pp. 33 - 39) reviewed DHC in Zircaloy cladding in a repository. Rothman concludes that DHC is unlikely unless the fuel rods have large existing cracks (exceeding approximately 50 percent of wall thickness) and very high stresses (exceeding approximately 137 MPa). He also concludes that hydride reorientation is also unlikely because of the lack of large temperature gradients in the repository and the cladding stresses are lower than needed for reorientation. Peehs (1998, pp. 5, 6) concluded that neither DHC nor hydride reorientation would occur in dry storage.

6.2.12.5 Hydride Reorientation (of Cladding)- YMP No. 2.1.02.22.05

This FEP has been excluded on the basis of low probability (credibility) of occurrence because test rods at the low end of the experimental stress range, but significantly higher than the stress expected at the repository temperatures, showed no reorientation of hydrides. This lack of reorientation indicates that there should be little, if any, cladding degradation due to hydrides under normal repository temperatures.

CRWMS M&O (2000g, Section 6.4) reviews hydride reorientation. The total hydrogen content in the fuel cladding is not as important as the amount of hydrogen in hydrides aligned perpendicular to the largest principal tensile stress. Normally, the hydrides are oriented in the circumferential direction and do not weaken the cladding against a hoop stress. Reorientation usually occurs under tensile stresses ranging from 69 to 208 MPa. Test rods at the low end of this range, but significantly higher than the stress expected at the repository temperatures, showed no reorientation of hydrides. This lack of reorientation indicates that there should be little, if any, cladding degradation due to hydrides under normal repository temperatures (Einziger et al. 1982, p. 65).

In commercial reactor cladding during irradiation, hydrides form in a circumferential orientation (the normal to the platelet is in the radial direction). Such hydrides do not significantly weaken the cladding against hoop stress. In one dry storage test, reorientation to the radial direction was observed in one rod (Einziger and Kohli 1984, p. 119) although rod failure did not occur. This occurred in a fuel rod that had very high stresses (145 MPa at 323°C). Reorientation was not observed in fuel rods with lower stresses (13 to 26 MPa). The hydride reoriented so that the normals in the circumferential direction could possibly weaken the cladding. Reorientation was also observed in Candu reactor coolant pipes. Hydride reorientation under repository conditions was investigated as a potential cladding degradation mode.

Figure 2 summarizes the data that Pescatore et al. (1990, pp. 52-55) collected showing reorientation. The points from Marshall show different fractions of reorientation. Hardie's point was reported at minimum stress. The straight solid line suggests a region in temperature and stress where reorientation occurs. This figure shows that at the cladding maximum temperature of 350°C, the stress would have to be greater than 120 MPa before reorientation would start to occur. This corresponds to about 60 MPa at room temperature. From CRWMS M&O 2000b (Figure 26), it is estimated that 5% of the rods could have stresses this high and might undergo some reorientation. Rothman (1984) also studied cladding degradation in a repository and concluded that hydride reorientation would not occur. He was not considering fuel with burnups and stresses as high as considered in CRWMS M&O 2000b. The predicted cladding temperatures will be less than 350°C and therefore this comparison should be considered an upper bounding case.

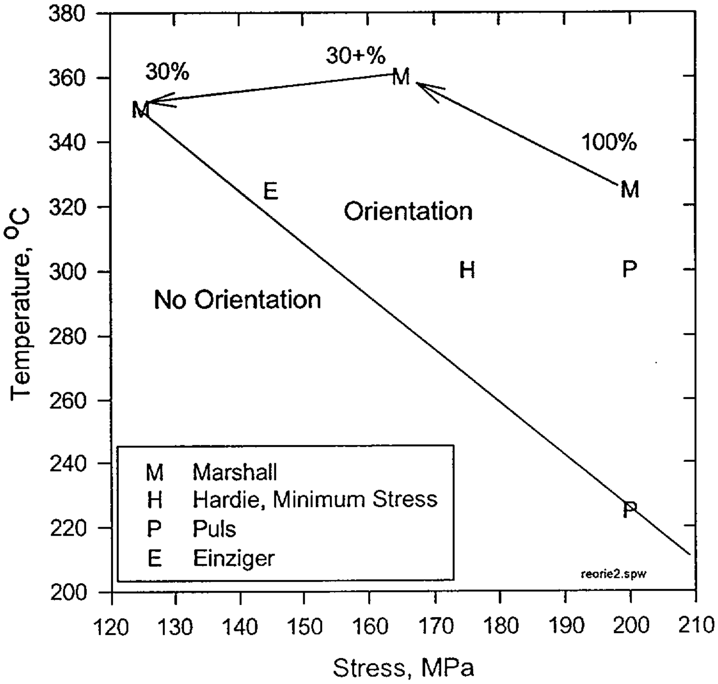

Figure 2. Temperature and Stress at which Hydride Reorientation was Observed for Cold Worked Zircaloy-2, 4 (Pescatore et al. 1990)

Pescatore et al. (1990, pp. 54, 69) state that even with hydride reorientation, stress levels will be insufficient to result in DHC and clad failure. For general DHC, they also said that crack propagation would tend to stop when the crack propagates into a hydride platelet. Puls (1988, p. 1507-1522) performed a series of strain tests on Zircaloy-2 with reoriented hydrides. His results are summarized in Table 3. He took samples of Candu coolant tubing and performed strain tests in the circumferential direction. The initial tubing is made in a similar fashion as cladding and develops hydride platelets with their normals in the radial direction. All tests were performed at room temperature. He used samples with both 20 ppm and 90 ppm hydrogen content and used two reorientation techniques to form hydrides of various lengths. One technique cooled the samples from 250°C with the stress near the yield point (designated y in Table 3). Other samples were cooled from 350°C at a stress of 200 MPa. This stress is more than twice than expected for most cladding in the repository (CRWMS M&O 2000b, Figure 26). Longer hydrides were produced by cooling the samples in a furnace, while shorter ones were produced by bench cooling the samples. Table 3 gives the range of hydride lengths. Two types of samples were used, smooth ones and notched ones (designated by n in the table). The table gives the stress (σ_y) for which 0.2 percent strain was measured. Also given is the ultimate stress at which necking and imminent failure was observed (or, for the arrested tests, was expected). For some experiments, the tests were stopped when the sample started to neck, but before failure (designated *a* for arrested in the table). In all of these tests, the reoriented hydrides did not significantly change the stress for 0.2 percent strain or the ultimate stress. Both stresses are much higher than those expected in repository cladding. The yield and ultimate strains reported are also higher than the strain failure criteria of 1 percent used in the cladding strain failure analysis developed by Peehs and Fleisch (1986).

Table 3. Puls' Zircaloy-2 Strain Tests on Zirconium with Reoriented Hydrides

Hydride Length (µm)	Type*	Yield Stress (0.2%) (MPa)	Tensile Stress (MPa)	Uniform/Total Strain (%)
Initial Material		627	650	-
7-20	-	632	678	4.7/15.8
7-20	y	627	675	4.7/15.8
7-20	-	612	659	4.7/15.8
7-20	n	783	885	-
7-20	n, y	774	882	-
7-20	n	933	1095	-
7-20	n	766	858	-
7-20	a	628	698	6 / 9
30-60	-	627	689	4.7/14.3
30-60	y	605	661	4.7/14.3
30-60	n	861	958	-
30-60	n, y	776	921	-
50-90	-	1079	1160	4.1/13.6
50-90	y	689	741	4.1/13.6
50-90	-	625	647	4.1/13.6
50-90	n	721	803	-
50-90	n, y	923	1032	-
50-90	n	811	936	-
50-90	a	633	701	- / 6
50-90	a	643	730	- / 6

*Type, y= hydride reoriented near yield stress, n - notched, a = arrested (test terminated before failure)
Source: M.P. Puls (1988, Tables 1,3,6)

Hydride reorientation might require all the hydride platelets to be dissolved before cool-down and reprecipitation starts. This could be necessary because the hydrides prefer to reprecipitate on existing platelets and these earlier, existing platelets are oriented in the circumferential direction. For all the hydrides to dissolve at a maximum cladding temperature of 350°C, the initial concentration must be less than 120 ppm (see Table 4). CRWMS (M&O 2000b, Figure 15) shows that only about 13 percent of the fuel has average concentrations this low. Knowing that the hydride concentration is directly proportional to the oxide thickness, CRWMS M&O (2000b, Figure 12) shows the fuel that could reorient under this hypothesis is the fuel with the lower burnups and therefore, lower stresses. Mardon et al. 1997 (their Figure 3, p. 408) shows hydride content as a function of burnup. To have less than 120 ppm, the figure shows that Zircaloy 4 fuel with burnups less than 22 MWd/kgU has low enough hydride concentrations to dissolve all the hydrides at 350°C. Again, these fuels would have the lowest stresses. The rod that Einziger observed reorientation was exposed to a 570°C transient which would support a solubility of 729 ppm (Table 4). It is possible that all the hydrides were dissolved in that experiment.

Further support for the above conclusions is provided by Garde 1989. A clear correlation between oxide thickness and hydrogen content is shown in his Figure 8. Of greater significance however is the conclusion that low ductility values were obtained on guide tube tensile samples (dogbone and ring tensile samples) at 300°C because of the high value of the hydride orientation factor and the fact that the hydrogen concentration was barely above the solubility limit for these samples (see comment column in Garde's Table 5). This observation implies that the hydrogen content of these samples was sufficiently low to go back into solution and then reorient on favorable precipitation sites. By contrast, cladding samples with higher hydrogen concentrations retained a number of hydride platelets and these sites were preferential locations for reprecipitation.

It should also be noted that the fabrication process can have an effect on these structures. In the case of CANDU cladding, the material is cold-drawn prior to stress-relieving or annealing (Cheadle et al. 1984). This produces a different structure to the pilgering process (with stress-relieving or annealing) as utilized in the US and which is considered to produce a more reorientation resistant structure. The Canadian tests showed that the fabrication process can affect the extent of hydride reorientation. Stresses of 175 MPa were required to increase the hydride reorientation angle by 10 to 20 degrees and a stress of 400 MPa was required to obtain complete reorientation close to the radial direction. In summary, failure of the cladding by hydride reorientation is unlikely and has not been included in the abstraction for TSPA-SR due to low probability (credibility) of occurrences. Stresses and temperatures are too low for hydride reorientation to occur in most of the fuel and the cladding material will maintain sufficient toughness even if hydride reorientation did occur such that failure would not be expected.

Table 4. Saturation Limits for Hydrogen in Zirconium as a Function of Temperature

Temperature (°C)	H Concentration (ppm)
40	0.1
50	0.2
60	0.3
70	0.4
80	0.6
90	0.9
100	1.2
110	1.6
120	2.1
130	2.8
140	3.6
150	4.6
160	5.8
170	7.3
180	9.0
190	11
200	13
210	16
220	19
230	23
240	27
250	32
260	38
270	44
280	50
290	58
300	66
300	66
310	75
320	85
330	96
340	108
350	120
360	134
370	149
380	165
390	182
400	201
410	221
420	242
430	264
440	288
450	313
460	339
470	367
480	396
490	427
500	459
510	493
520	529
530	565
540	604
550	644
560	686
570	729

Source: Pescatore et al. (1990, eq. 6, p.44)

6.2.12.6 Hydride Axial Migration (of Cladding)- YMP No. 2.1.02.22.06

This FEP should be excluded from further consideration based on low probability (credibility) of occurrence because it is unlikely that sufficient hydrogen can be moved because of a lack of large temperature gradients in the WPs.

Hydrides can form in cooler parts of the rod (end sections) because the hydrogen can dissolve into the fuel cladding metal matrix at a warmer area, diffuse toward the cooler area, and condense there. The effect was studied for dry storage (Cunningham et al. 1987, Appendix C) and for a 90-year period was determined not to be a problem. As the repository cools, the driving force for this redistribution (hydrogen solubility, temperature gradient, and diffusion rate) all decrease. The WP internals will act to minimize the temperature variation along the length of the fuel assembly. McMinn et al. (2000, Figure 15) show that a minimum of 42°C temperature difference between the hot location and cold location in the cladding is needed to move the hydrogen. At temperatures below 200°C, temperature differences over 75°C are needed to move the hydrogen. It is unlikely that sufficient hydrogen can be moved because of a lack of large temperature gradients in the WPs.

6.2.12.7 Hydride Embrittlement from Fuel Reaction (Causes Failure if Cladding)- YMP No. 2.1.02.22.07

This FEP should be excluded from further consideration based on low probability (credibility) of occurrence because cladding degradation from hydride embrittlement from fuel reaction is only observed in BWRs and a high temperature steam environment is required for failure propagation. Such conditions are unlikely at YMP after WP failure.

Hydrides can be formed in the cladding from the oxidation of UO_2 fuel (Wasywich and Frost 1992, p. 1171). The fact that the fuel is oxidizing means that the cladding has already been perforated. This type of cladding degradation is only observed in BWRs and a high temperature steam environment is required for failure propagation. Such conditions are unlikely at YMP after WP failure. If the embrittlement becomes severe and the cladding is mechanically loaded in excess of its fracture toughness, the cladding could fail exposing additional fuel. This type of failure has a small effect on the fuel oxidation rate since the fuel oxidizes after the first penetration. It does affect the amounts and rate of fission product migrating out of the secondary phases of the fuel dissolution. This type of secondary cladding failure has not been modeled.

6.2.13 Cladding Unzipping - YMP No. 2.1.02.23.00

Related Primary FEPs: No closely related FEPs

FEP Description In either dry or wet oxidizing conditions and with perforated fuel cladding, the UO_2 fuel can oxidize. The volume increase of the fuel as it oxidizes can create stresses in the cladding that may cause gross rupture of the fuel cladding (unzipping).

Screening Decision: Dry Oxidation – Exclude, based on low probability of occurrence (credibility).

Wet Oxidation – Include.

Screening Argument: In the TSPA-SR model, wet unzipping is included as the key element in exposing and dissolving fuel from damaged cladding. This mechanism conservatively bounds the slow diffusive release of radionuclides through pinholes of the fuel cladding.

As discussed in section 6.2.13.1 below, dry unzipping has been excluded due to a low probability of occurrence (credibility) because in the environment of the repository only a very small fraction of the fuel would undergo dry unzipping, and then only if the disposal container was breached in the first few hundred years, and also if the cladding was already perforated such that the UO_2 would be rapidly oxidized to U_3O_8.

TSPA Disposition Wet Oxidation - included as described in CRWMS M&O 2000e, and abstracted into TSPA-SR in CRWMS M&O 2000c.

IRSR Issuance: CLST1, CLST3, TSPAI4

References: Doubt, G. 1984, Einziger, R.E.; Atkin, S.D.; Stellrecht, D.E.; and Pasupathi, V. 1982, Gray, W.J. and Wilson, C.N. 1995, Hanson, B.D. 1998, IAEA 1988, Johnson, A.B., Jr. 1977, Johnson, L.H. and Taylor, P.; 1998, McEachern, R.J. and Taylor, P. 1997, CRWMS M&O 1998a, CRWMS M&O 1999c, CRWMS M&O 2000h, CRWMS M&O 2000e

Basis for Screening Decision

The bases for the screening decision are presented below in Sections 6.2.13.1 through Sections 6.2.13.4.

6.2.13.1 Dry Oxidation of Fuel (Causes Failure of Cladding) - YMP No. 2.1.02.23.02

This secondary FEP has been excluded due to a low probability of occurrence (credibility) because earlier analysis presented in the TSPA-VA (CRWMS M&O 1998a, Section 6.3.1.1.9) showed that the cladding could unzip if both the cladding is perforated and the WP also fails within the first 200 years. Such early failures of the WP are not expected.

The cladding that has failed during reactor operation, from creep rupture, or from DHC generally has small cladding perforations and very small areas of fuel exposed to the near-field environment. When the WP fails, the inert atmosphere inside the WP is replaced with the air and steam from the near-field environment. After WP failure, the near-field environment is almost completely steam and dry oxidation is not expected. If the dry oxidizing conditions existed after WP failure and with perforated cladding, the UO_2 phase in the spent fuel can oxidize in two stages, first to U_4O_9 and then to U_3O_8, causing an increase of the spent fuel matrix volume. These are nominal formulas only and do not necessarily describe the actual stoichiometry of these oxides. The stress from the volume increase can unzip the clad, causing a gross rupture of the clad and exposing the fuel inside. Oxidation also produces an increase in surface area and dissolution rate, the combined factor being 150 (Gray and Wilson 1995, p. vii). Dry fuel oxidation has been experimentally studied by Einziger et al. (1982, p. 65) and Hanson (1998, p. 2.3). McEachern and Taylor (1997, p. i) summarize the work of others. An improved model for dry unzipping is presented in (CRWMS M&O 2000h).

For defective WPs (failed at emplacement), it may be possible to unzip the cladding of the fuel rods that are perforated. This will probably not actually occur because the environment is mostly steam and the O_2 partial pressure is low at this early time. Therefore, dry oxidation is unlikely.

However if dry oxidation does occur at this early time, the degree of unzipping for defective WPs can be estimated. The combination of defective WPs, which is taken as 0.02 percent (Doubt 1984, p. 30) with premature rod failure, which is 1.25 percent including the stainless steel cladding, gives the fraction of rods expected to be converted to U_3O_8 to be 2.5×10^{-6} ($= 2 \times 10^{-4} \times 1.25 \times 10^{-2}$). Overall, when considering the presence of steam, timing of WP failures and decreasing temperatures, it is expected that only very small fractions of fuel will be converted to U_3O_8, and these small fractions will not affect the PA. Earlier analysis presented in the TSPA-VA (CRWMS M&O 1998a, Section 6.3.1.1.9) showed that the cladding could unzip if both the cladding is perforated and the WP also fails within the first 200 years. Such early failures of the WP are not expected.

6.2.13.2 Wet Oxidation of Fuel (Causes Failure of Cladding) - YMP No. 2.1.02.23.03

Cladding could unzip in a wet environment. As the UO_2 is converted to secondary phases such as metaschoepite and Na-boltwoodite, the fuel volume increases. These phases will only form where there is available liquid volume, or, alternatively, they might form at solid surfaces and tear the cladding. Cladding unzipping has not been observed in spent fuel pools where fuel has been stored for tens of years as noted by the IAEA (1988, Table XXVI) and Johnson (1977, p. 20). The Canadians tested a defective rod in a steam autoclave for ten years and did not measure any cladding strain (Johnson and Taylor 1998, p. 10). While wet unzipping has not been observed, the time periods are very short compared to the repository scale and a form of wet oxidation and secondary phase formation might tear the cladding.

The current wet unzipping model is developed in CRWMS M&O 2000e and the abstraction is described in CRWMS M&O 2000c (Section 6.6). All failed Zircaloy clad rods, all stainless steel clad rods (presumed to be failed) are available to unzip. The unzipping velocity is dependent on the fuel intrinsic dissolution rate. This rate is evaluated at each TSPA-SR time step because it depends on the temperature and chemistry inside the WP.

6.2.13.3 Volume Increase of Corrosion Products - YMP No. 2.1.09.03.00

Related Primary FEPs: 2.1.02.23.00

YMP Primary FEP Description: Corrosion products have a higher molar volume than the intact material. Increases in volume during corrosion could change the stress state in the material being corroded.

Screening Decision: Included: clad unzipping due to wet oxidation of CSNF

Excluded: clad unzipping due to dry oxidation of CSNF based on low probability.

Screening Argument: Dry unzipping due to volume increase of corrosion products has been excluded because dry oxidation occurs only for the very limited case of early waste package and clad failure and when relative humidity is low. This FEP discussion is limited to the wet and dry oxidation of CSNF spent fuel and its subsequent affect on clad behavior. Both wet and dry oxidation can lead to an increase in fuel-matrix volume and to unzipping of the cladding. The unzipping of the cladding results in a significant increase in surface area and degradation rate of the exposed fuel. Dry oxidation occurs only for the very limited case of early waste package and clad failure and when relative humidity is low.

Wet oxidation is possible but has not been observed. However, because unzipping in a wet environment could not be entirely ruled out and because complete exposure of the matrix would bound diffusive releases of radioisotopes through the perforation, the project included the possibility of the cladding unzipping in a wet environment due to volume increase of corrosion products. See also FEP YMP No. 2.1.02.23.00, "Cladding Unzipping."

TSPA Disposition: Fuel rods with perforated cladding are expected to remain intact until the WP fails and permit air and moisture to enter. Wet unzipping is modeled to start at WP failure for rods that are pre-breached or when rod perforation occurs, if after WP failure. In the model, the fuel matrix is dissolved at the intrinsic dissolution rate that is evaluated at the local temperature and in-package chemistry. The dissolved UO_2 forms a precipitate. This secondary phase isolates most of the fuel from the moisture and increases volume compared to UO_2. In time, the clad in the reaction region is torn as the reaction continues. This reaction region is cone shaped based on experimental observations of dry unzipping. The unzipping propagates along the rod at a rate approximately 40 times (range 1 to 240 times) faster than the intrinsic dissolution rate. It is accepted that the perforation is in the center of the rod. This maximizes the release rate. The time to unzip a rod as a function of temperature, local chemistry and pH. In TSPA, the unzipping velocity and fraction of fuel exposed is evaluated at each time step because of the evolution of in-package chemistry and temperature (CRWM M&O 2000e; CRWM M&O 2000c).

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 1998a, CRWMS M&O 1998b, CRWM M&O 2000e; CRWM M&O 2000c, CRWMS M&O 2000e, Gray and Wilson 1995, Einziger et al. (1982) and Hanson (1998). McEachern and Taylor (1997)

Basis for Screening Decision:

Wet Oxidation

CRWMS M&O 1998a discusses the mechanism of wet oxidation which is summarized as follows. Upon contact with humid air, the fuel is expected to form dehydrated schoepite and metaschoepite. Upon contact with groundwater (e.g., J-13 well water) the fuel is expected to also form sodium boltwoodite. These reaction products have greater molar volumes than that of the UO_2 from which they are formed. It is this increase in molar volume of the reaction products formed through oxidative dissolution of the fuel and precipitation of alteration phases that leads to the increased volume of corrosion products (CRWMS M&O 2000e).

Because wet unzipping has not been observed experimentally, its occurrence and, if it occurs, its rate were examined by decomposing the process into sub-process alternatives and examining the available technical evidence (CRWMS M&O 2000e). An ongoing testing program in the YMP provides some of the current information and will provide further information as the experiments continue.

Two situations that bound the range of possibilities for mass transport of the reactants to reaction locations within a fuel rod with breached cladding were considered (CRWMS M&O 2000e). The first was a "limited water" condition" where the reactions occur in a closed volume to which the water supply can not be replenished. This situation represents a condition that could develop if the reactions were "self sealing." The second situation is referred to as an "unlimited water" condition, in which the water supply is continuously replenished at the reaction sites within the rod. This situation represents a condition in which advective or diffusive transport could supply both the water and solutes at a sufficient rate to the reaction locations that mass transport limitations on the reaction progress are negligible. (Note: Silica mass transport limitations may be significant for limiting the formation of sodium boltwoodite).

The conclusion from CRWMS M&O 2000e was that the "limited water" situation will always lead to a volume decrease even though the solid alteration products have a greater specific volume than the starting UO_2 . The "unlimited water" condition, however, will lead to a volume increase. Also, the expansion caused by the reaction of the spent-fuel matrix and the precipitation of corresponding secondary products will not rupture the intact clad away from the breach. Rather, the larger volume of the secondary reaction products is likely to seal the fuel-clad gap and any cracks (CRWMS M&O 2000e).

These arguments outlined above are tantamount to saying that the fuel oxidative-dissolution reactions are likely to be self-sealing in a fuel rod with perforated clad. The statement that there is evidence that the reactions are self-limiting has been made (CRWMS M&O 1998b), but the experimental evidence was not presented. After the void volumes are filled, it is unlikely that the supersaturation ratios needed to generate significant crystallization pressures (i.e., crystallization pressures that could cause one or two percent hoop strain in the clad) can develop. However, only limited experimental evidence and no quantitative mass-transport calculations are available to support this point.

For these reasons, unzipping in a wet environment could not be entirely ruled out. Also, because complete exposure of the fuel matrix bounds the releases of radioisotopes, the project included the possibility of the clad unzipping in a wet environment. See FEP 2.1.02.23.00, "Cladding Unzipping."

6.2.13.4 Gap and Grain Release of Cs, I - YMP No. 2.1.02.07.00

Related Primary FEPs: No closely related FEPs

YMP Primary FEP Description: While in the reactor at the high temperatures, radionuclides such as I and Cs may migrate and preferentially accumulate in cracks in the fuel matrix, grain boundaries of the UO_2, and in the gap between the fuel and cladding. After the waste package fails and the cladding perforates, the release rate of this fraction of the radionuclides could be rapid. In addition, reactions between materials within the spent fuel assembly or waste package backfill such as I and Pb may affect this process.

Screening Decision: Include gap and grain-boundary inventory produced prior to emplacement in the repository.

Exclude I and Cs migration after emplacement in the repository based on low probability (credibility)

Screening Argument: This FEP and its associated secondary FEPs present three issues as discussed below. I and Cs Migration release after emplacement in the repository is excluded due to a low probability of occurrence (credibility) because the use of a reported Cs diffusion coefficient results in between 490 million and 860 million years for half of the Cs to diffuse to the boundary of a grain with a 5 μm radius. This is based on a diffusion coefficient for Cs published by Walker et al. (1996). The diffusion coefficient, which was extrapolated downward from 1200°C to 300°C is in the range of 2.8×10^{-25} to 4.9×10^{-25} cm^2/s. Use of this diffusion coefficient results in between 490 million and 860 million years for half of the Cs to diffuse to the boundary of a grain with a 5 μm radius (Darken and Gurry 1953, p. 447).

Gap-and-Grain-Boundary Release

The first issue, "gap-and-grain boundary release," applies mainly to UO_2-type fuels and possibly other non-metallic fuels, such as MOX fuel, where the fuel is in the form of relatively large pellets. It does not apply to metallic fuels or to oxide fuels where the fuel is dispersed in the form of small particles in a matrix such as aluminum.

In non-dispersed oxide fuels in the form of relatively large pellets, a portion of some elements (notably Cs and I) can migrate out of the fuel matrix into the grain boundaries and the fuel/pellet gaps under the influence of temperature gradients during reactor operation. In these regions, the Cs and I are readily available for dissolution by any water that penetrates the fuel cladding. In the CSNF Cladding Degradation Component of the Waste Form Degradation Model, a fraction of Cs and I radionuclides in the inventory are released immediately upon perforation of the cladding in the CSNF Cladding Degradation Component of the Waste Form Degradation Model.

I and Cs Migration

The second issue , "I and Cs migration," considers the possibility that, under repository temperatures, more radionuclides such as Cs and I could diffuse out of the fuel matrix and accumulate at the grain boundaries, cracks in fuel matrix, and gap between the fuel matrix and cladding (CRWMS M&O 2000c). Because of the relatively low temperatures in the repository at the expected time of clad failure (CRWMS M&O 2000j; 2000k), this effect is excluded from CSNF Cladding Degradation Component based on low probability (credibility). Specifically, based on a diffusion coefficient for Cs published by Walker et al. (1996) at 1200 °C, the diffusion coefficient extrapolated to 300°C is in the range of 2.8×10^{-25} to 4.9×10^{-25} cm^2/s. Use of this diffusion coefficient results in between 490 million and 860 million years for half of the Cs to diffuse to the boundary of a grain with a 5 μm radius (Darken and Gurry 1953, p. 447).

Pb-I Reactions

The third issue, "Pb-I reactions," considers the possibility that reactions, such as those involving Pb and I, would minimize the gap-and-grain-boundary inventory, thus mitigating any release from that inventory. No lead backfill ("filling") inside the WP is planned by the Yucca Mountain Project and so cannot be included. Other reactions, such as those between I and Cu or Cs and Mo, tend to mitigate radionuclide releases from the gap-and-grain-boundary inventory, and are conservatively excluded from TSPA analysis.

TSPA Disposition: Although past TSPAs have commonly accepted that the gap-and-grain-boundary inventory for commercial spent fuel is 2% of the total inventories of I and Cs, a more defensible bounding approach was taken for the CSNF Cladding Degradation Component of the Waste Form Degradation Model of TSPA-SR. As more fully described in the AMR, "Clad Degradation—Summary and Abstraction" (ANL-WIS-MD-000007) (CRWMS M&O 2000e), the total gap- and grain-boundary fraction was made up of two components: the gap inventory and the inventory of radionuclides that will be dissolved from the fuel matrix before cracks in the fuel plug and unzipping of the cladding begins. The gap inventory of I was released in the same proportion as an observed maximum fission gas release of 4.2%; the gap inventory of Cs was one third of this amount (i.e., 1.4%). The inventory of all other radioisotopes was sampled from a uniform distribution that ranged between 0 and 0.4%. The gap inventory of I and Cs was added to this sampled inventory.

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 2000c, CRWMS M&O 2000j; CRWMS M&O 2000k CRWMS M&O 2000e, Walker et al. (1996), Darken and Gurry 1953

Basis for Screening Decision:

Prior discussion provides adequate basis.

6.2.14 Mechanical Failure (of Cladding) - YMP No. 2.1.02.24.00

Related Primary FEPs: 1.2.03.02.00

FEP Description Mechanical failure of cladding may result from external stresses, such as ground motion and rockfall during earthquakes.

Screening Decision: Include.

Screening Argument: The analysis of seismic events is included in the TSPA-SR. Ground motion damage with a frequency of 1.1×10^{-6} events/year break all of the fuel cladding.

Such events are sampled, and, when such an event occurs, all cladding is failed and is considered to be available for unzipping.

The effect of a rubble bed consisting of rocks from a drift collapse on bare fuel rods (no WP or possible drip shield protection) is not included in TSPA-SR because this would not occur until well after the 10,000 years considered for the TSPA-SR. Since high accelerations are needed to fail the cladding, rock drops onto an intact WP will not cause rod failure and therefore was not included in the TSPA-SR.

TSPA Disposition Mechanical failure will be represented in the performance assessment for the TSPA-SR as described in CRWMS M&O 1999b, Section 6.1.

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 1999b, Witte et al. 1989

Basis for Screening Decision

Seismic analysis (CRWMS M&O 1999b, Section 6.1) shows that most of the rods in the WPs would fail from a very severe earthquake ground motion damage with an annual frequency of about 10^{-6}, but no rods would fail for less severe and moderate frequency seismic events. This is consistent with studies (Witte et al. 1989, p. 194) of rod damage during transportation accidents that concluded that 63 g accelerations are needed to fail the rods in the shipping container (or WP). Therefore, the seismic failures have been included in the TSPA-SR as a disruptive event. See also the discussion in FEP 1.2.03.02.00 (CRWMS M&O 2000c)

The analysis of seismic events is included in the TSPA-SR. Based on the analysis (CRWMS M&O 1999b, Section 6.1), seismic events with a frequency of 1.1×10^{-6} events/year would break most of the fuel. Such events are sampled, and, when such an event occurs, all cladding is failed and is considered to be available for unzipping.

6.2.15 Rockfall (Large Block)- YMP No. 2.1.07.01.00

Related Primary FEPs: *2.1.02.24.00*

YMP Primary FEP Description: Rockfalls occur with block large enough to mechanically tear or rupture waste packages.

Screening Decision: Exclude based on low probability (credibility).

Screening Argument: The omission of rockfall damage to cladding is due to a low probability of occurrence because analyses indicate rock drops onto an intact WP will not cause rod failure.

Cladding perforation from the collapse of the WP with a large block rockfall is not considered since cladding perforation from other modes occurs before the WP can collapse. CRWMS M&O

1999b (Section 6.2) also considered the effect of a rubble bed consisting of rocks from a drift collapse on bare fuel rods (no WP or possible drip shield protection). The analysis showed that the bare fuel assemblies would fail under the static loading of the rocks. This effect was not included in the cladding degradation abstraction because it does not occur until after the WP no longer affords any protection. The first patches (about 100 cm by 100 cm) penetrate the WP in 50,000 to 60,000 years, and a significant number of patches (about 100 patches) are open in about 350,000 years. With the potential for rubble bed damage not occurring until well after the 10,000 years considered for the TSPA-SR, rubble bed damage to the cladding was neglected. Witte et al. 1989 (p. 194, Table 3) analyzed rod damage during transportation accidents that concluded that 63 g accelerations are needed to fail the rods in the shipping container (or WP). This high acceleration indicates that rock drops onto an intact WP will not cause rod failure and therefore this FEP was not included in the TSPA-SR.

The perforation of cladding from shaking in a severe earthquake is considered in a disruptive event analysis(cladding FEP 2.1.02.24.00) and disruptive events FEP AMR (CRWMS M&O 2000m).

TSPA Disposition: None

Basis for Screening Decision:

A combination of drip shield and backfill has been considered as possible EBS design features. The use of backfill significantly reduces the drop distances of falling rock blocks, and thus reduces the kinetic energy and impact velocity of all falling rocks hitting backfill (instead of waste packages/waste forms). Also, a backfill consisting of granular materials with dimensions orders of magnitude smaller than the size of the falling blocks of rock will be a highly effective energy/impulse-load absorbers, reducing and preventing damage to HLW glass and other waste forms. Thus, the glass-waste-form surface-area increase in response to falling rock blocks which could theoretically cause form fracture and fragmentation, is expected to be nil.

Bounding static analyses performed in CRWMS M&O 1999b (Section 5.3) showed that the combined load from rockfall and backfill (27 psi or 0.18 MPa) was much less than the estimated collapse pressure of a drip shield idealized as a cylindrical shell (375 psi or 2.6 MPa). Further analyses of the drip shield, idealized as either a cylindrical shell or flat plate, bounded the maximum deflection of the crown of drip shield between 19 mm and 77 mm. These analyses indicate that it is very unlikely that rockfall will ever come in contact with waste. Elimination of the backfill in these static analyses would not change their conclusion. The elimination of backfill in EBS design reduces the static load on the drip shield, and therefore, would reduce crown deflection.

TSPA Disposition: ***None***

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 1999b, Witte et al. 1989, CRWMS M&O 2000m

6.2.16 Localized Corrosion Perforation from Fluoride - YMP No. 2.1.02.27.00

Related Primary FEPs: No closely related FEPs

FEP Description Fluoride is present in Yucca Mountain groundwater, and zirconium has been observed to corrode in environments containing fluoride. Therefore, fluoride corrosion of cladding may occur in WPs.

Screening Decision: Include.

Screening Argument: Numerous processes have been identified that might cause localized corrosion and eventually perforation of Zircaloy cladding on CSNF. Yet as summarized in FEP 2.1.02.16.00 and discussed in more detail in the AMR, *Clad Degradation—Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (ANL-EBS-MD-000012) (CRWMS M&O 2000a), the conditions necessary for these processes generally do not exist in the repository. Even pitting corrosion in the presence of fluorides does not appear likely since the concentration of fluoride in pore water at Yucca Mountain is low, this low initial amount of fluoride is consumed during the corrosion process, and the pH of the in-package chemistry does not drop low enough to promote corrosion. In the TSPA-SR, one important condition was made to make the process feasible under the modeled conditions of the repository: the fluoride was to interact with one small segment of the cladding until the rod perforated, and not diffuse or migrate to other regions of the waste package (CRWMS M&O 2000c).

TSPA Disposition Localized corrosion in the present of fluorides was included in the TSPA-SR to account for modeling uncertainty of the in-package chemistry since conditions for fluoride corrosion were considered more likely to occur relative to the other processes examined (CRWMS M&O 2000a).

IRSR Issues: CLST1, CLST3, TSPAI4

References: CRWMS M&O 2000a, CRWMS M&O 2000c

Basis for Screening Decision Corrosion of zirconium has been observed in fluoride-containing environments. Since fluoride is present in Yucca Mountain groundwaters, localized corrosion from fluoride is considered as a perforation mechanism in the CSNF Cladding Degradation Component. Two scenarios for fluoride corrosion have been considered. In the first (water-filled WP scenario), the WP is full of water, and fluoride ions are transported to the cladding by aqueous diffusion. This scenario is less severe than the second one and is not addressed further. In the second (flow-through scenario), water enters the WP through a breach on the top and drips out through a breach on the bottom. The fluoride is transported rapidly through the WP by advection, and fluoride attack is localized on a relatively small area of cladding (10 mm of rod length). All the fluoride that enters the WP is consumed on the surface of the first rod it encounters until that rod fails. After one rod fails, the fluoride attacks the next rod. After WP breach, the fraction of cladding that fails is proportional to the volume of water that enters the package, reaching one (100%) when 2,424 m^3 of water enters the WP. The uncertainty range placed on this estimate is plus or minus a factor of ten. In this approach, the fraction of cladding perforated is linearly dependent on the water inflow (percent failing equals

$0.0413 \times m^3$ water in WP). The water flow into the WP increases with time as additional patches on the WP open. Cladding perforation rate also depends on the location of the WP group in the repository because of different drip rates in different repository regions (CRWMS M&O 2000c).

6.2.17 Diffusion-Controlled Cavity Growth - YMP No. 2.1.02.26.00

Related Primary FEPs: 2.1.02.19.00

FEP Description Diffusion-Controlled Cavity Growth (DCCG) was once thought to be a possible creep rupture mechanism that could occur under the temperature and pressure conditions that prevailed during dry storage of spent fuel and might occur during disposal.

Screening Decision: Excluded based on low probability (credibility) since glide and Coble creep rupture process is included and would account for creep failed fuel—see FEP 2.1.02.19.00).

Screening Argument: Omission of this FEP is justified on the basis that DCCG is a low probability occurrence (credibility) because DCCG, as a mechanism to fail Zircaloy cladding, has not been observed experimentally. Glide and Coble creep rupture processes are included in the analysis and would account for creep failed fuel. Applicants for dry storage licenses for CSNF were once required by the U.S. Nuclear Regulatory Commission (NRC) to assume diffusion controlled cavity growth (DCCG) to evaluate dry storage designs. That is, the NUREG-1536 (NRC 1997, Section 4.V.4.b) endorsement of the DCCG method which was used to calculate a maximum cladding temperature limit for a dry storage design. However, this design limit is overly restrictive and relatively inflexible. Recent literature does not support the use of this model for zirconium-based materials (Pescatore and Cowgill, 1994, p. 83-85) since it has not been validated, and voids and cavities are rarely seen in irradiated Zircaloy. Pescatore and Cowgill (1994, p. 85) recommend a methodology similar to the approach used of calculating the amount of creep and comparing it to a creep failure criteria. The current NRC Interim Staff Guidance (ISG) Number 11 (NRC 2000a) recognizes the controversy with the DCCG conceptual model and permits license applicants to use other creep models in their license application. The use of creep model presented in CRWMS M&O (2000c, Section 6.2) is consistent with this ISG. The German dry storage creep analysis approach (Peehs 1998) is also consistent with the methodology used in CRWMS M&O (2000c, Section 6.2) and cited in Section 6.2.9 of this AMR. DCCG is excluded as a creep rupture mechanism. Creep rupture, as a cladding perforation process, however, is included in the CSNF Cladding Degradation Component of the Waste Form Degradation Model as described in FEP 2.1.02.19.00. The CSNF Cladding Degradation Component uses a creep strain method based on phenomena expected to be encountered under dry storage conditions as explained in the AMR *Clad Degradation—Summary and Abstraction* (CRMWS M&O 2000c).

TSPA Disposition None.

IRSR Issues: CLST1, CLST3, TSPAI4

References: Pescatore, C. and Cowgill, M. 1994, NRC 1997, NRC 2000a, CRWMS M&O 2000c, Peehs 1998

Basis for Screening Decision

No further elaboration in addition to the *Screening Argument* is necessary.

6.3 NRC Issue Resolution

Both the expectations of and the strategic planning activities by the NRC emphasize the early identification and resolution of licensing issues, prior to the NRC's receipt of the license application to construct a geologic repository. The NRC's objective is to reduce the number of issues and to better define in advance the issues that may be in dispute between licensee and regulator. NRC regulations and a 1993 agreement between NRC and DOE expand on this initiative by allowing NRC staff-level issue resolution to be achieved during the pre-licensing consultation period. Such resolution, however, would not preclude the issue's being raised and reconsidered during licensing proceedings.

6.3.1 Staff-Level Issue Resolution

To structure staff-level interactions, NRC has focused on the topics the NRC considers most critical to post-closure performance of the proposed geologic repository. At present, NRC staff has developed 10 Key Technical Issues, nine of which relate to post-closure performance assessment. The nine KTIs are identified in Table 5.

Table 5. Key Technical Issues Related to Post-Closure Performance Assessment

Number	Issue
1	Total System Performance Assessment and Integration
2	Container Lifetime and Source Term
3	Evolution of the Near Field
4	Radionuclide Transport
5	Unsaturated and Saturated Flow Under Isothermal Conditions
6	Thermal Effects on Flow
7	Repository Design and Thermal Mechanical (TM) Effects
8	Structural Deformation and Seismicity
9	Igneous Activity

Each KTI is configured with sub-issues, sub-issue components, and the NRC's acceptance criteria, thus facilitating the NRC's issuance of Issue Resolution Status Reports. As an important part of the staff-level interaction process, the IRSR provides the primary mechanism that NRC staff uses to provide feedback to DOE.

Full resolution at the staff level concerning a particular issue is achieved during pre-licensing whenever the NRC staff have no further questions or comments regarding how the DOE program is addressing the issue, and the NRC has documented this situation. Furthermore, there may be some cases in which staff-level resolution during pre-licensing may be limited to documenting a common understanding regarding differences in NRC and DOE technical positions.

6.3.2 Cross-Reference Between FEPs Related to Cladding Degradation and NRC Key Technical Issues

The following Tables 6 and 7 contain cross-references between NRC and Container Lifetime and Source Terms (CLST) and Total System Performance Assessment and Integration (TSPAI) Key Technical Issues and the cladding degradation FEPs.

Table 6. Cross-Reference Between FEPS Related to Cladding Degradation and Container Life and Source Term Key Technical Issues

CLST Sub-Issue	Acceptance Criteria	FEPs	
		Number	Title
3. The rate at which radionuclides in SNF are released from the EBS through the oxidation and dissolution of spent fuel	4. DOE has identified and considered likely processes for SNF degradation and the release of radionuclides from the EBS, as follows: dissolution of the irradiated UO2 matrix, with the consequent formation of secondary minerals and colloids; prompt release of radionuclides; degradation in the dry air environment; degradation and failure of fuel cladding; preferential dissolution of intermetallics in DOE SNF; and release of radionuclides from the WP emplacement drifts.	2.1.02.07.00	Gap and Grain Release of Cs, I
		2.1.02.11.00	Waterlogged rods
		2.1.02.12.00	Cladding degradation before YMP receives it
		2.1.02.13.00	General corrosion of cladding
		2.1.02.14.00	Microbiologically Influenced Corrosion (MIC) of cladding
		2.1.02.15.00	Acid corrosion of cladding from radiolysis
		2.1.02.16.00	Localized corrosion (pitting) of cladding
		2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding
		2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding
		2.1.02.19.00	Creep rupture of cladding
		2.1.02.20.00	Pressurization from He production causes cladding failure
		2.1.02.21.00	Stress corrosion cracking (SCC) of cladding
		2.1.02.22.00	Hydride embrittlement of cladding
		2.1.02.23.00	Cladding unzipping
		2.1.02.24.00	Mechanical failure of cladding
		2.1.02.27.00	Localized Corrosion Perforation from Fluoride
		2.1.02.28.00	Diffusion-Controlled Cavity Growth
		2.1.07.01.00	Rockfall (Large Block)
		2.1.09.03.00	Volume Increase of Corrosion Products
		2.1.11.07.00	Thermally-Induced Stress Changes in Waste and EBS

Table 7. Cross-Reference Between FEPS Related to Cladding Degradation and Total System Performance Assessment and Integration Issues

TSPAI Sub-Issue	Acceptance Criteria	FEPs	
		Number	Title
4) Scenario Analysis			
1) Identification of an Initial Set of Processes and Events	Criterion T1: DOE has identified a comprehensive list of processes and events that: (1) are present or might occur in the Yucca Mountain region and (2) includes those processes and events that have the potential to influence repository performance.	2.1.02.07.00	Gap and Grain Release of Cs, I
		2.1.02.11.00	Waterlogged rods
		2.1.02.12.00	Cladding degradation before YMP receives it
		2.1.02.13.00	General corrosion of cladding
		2.1.02.14.00	Microbiologically Influenced Corrosion (MIC) of cladding
		2.1.02.15.00	Acid corrosion of cladding from radiolysis
		2.1.02.16.00	Localized corrosion (pitting) of cladding
		2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding
		2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding
		2.1.02.19.00	Creep rupture of cladding
		2.1.02.20.00	Pressurization from He production causes cladding failure
		2.1.02.21.00	Stress corrosion cracking (SCC) of cladding
		2.1.02.22.00	Hydride embrittlement of cladding
		2.1.02.23.00	Cladding unzipping
		2.1.02.24.00	Mechanical failure of cladding
		2.1.02.27.00	Localized Corrosion Perforation from Fluoride
		2.1.02.28.00	Diffusion-Controlled Cavity Growth
		2.1.07.01.00	Rockfall (Large Block)
		2.1.09.03.00	Volume Increase of Corrosion Products
		2.1.11.-7.00	Thermally-Induced Stress Changes in Waste and EBS

7. CONCLUSIONS

The following conclusions may be drawn:

- This AMR contributes to the Yucca Mountain scenario development methodology by screening the FEPs related to cladding degradation.
- This AMR develops screening arguments and TSPA-SR disposition statements for these FEPs. This AMR provides information for the YMP FEP Database and guidance to TSPA-SR analyses, which are appropriate for both site recommendation and license application documents.
- Screening decisions reached in this AMR are documented in Table 2.
- By providing references to appropriate other AMRs, this AMR provides a valuable link between issues related to cladding degradation and the research directed at their resolution.
- Finally, this AMR correlates FEPs related to cladding degradation with the NRC Container Life and Source Term and Total System Performance Assessment and Integration Key Technical Issues.

This AMR uses no models and performs no analysis but only summarizes the models or analysis performed in the references. The references address cladding degradation before receiving it at YMP and after emplacement. Inherent is the presumption that the cladding is not damaged at the YMP surface facilities. This requires that the YMP will have both procedures and design features for the surface facilities to ensure that cladding degradation does not occur at the surface facilities. This AMR was developed using the methodology recommended in ASTM C 1174.

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8. INPUTS AND REFERENCES

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8.2 Codes, Standards, Regulations and Procedures

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