



QA: QA

ANL-WIS-MD-000008 REV 02

October 2004

Clad Degradation - FEPs Screening Arguments

Prepared for:
U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Office of Repository Development
1551 Hillshire Drive
Las Vegas, Nevada 89134-6321

Prepared by:
Bechtel SAIC Company, LLC
1180 Town Center Drive
Las Vegas, Nevada 89144

Under Contract Number
DE-AC28-01RW12101

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party's use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

QA: QA

Clad Degradation - FEPs Screening Arguments

ANL-WIS-MD-000008 REV 02

October 2004

OCRWM	SCIENTIFIC ANALYSIS SIGNATURE PAGE/ CHANGE HISTORY	Page iii
		1. Total Pages: 104

2. Scientific Analysis Title Clad Degradation - FEPs Screening Arguments			
3. DI (including Revision Number) ANL-WIS-MD-000008 REV 02			
4. Total Appendices Two (2)		5. Number of Pages in Each Appendix A-4; B-6	
6. Originator	Printed Name Randolph Schreiner	Signature <i>Randolph Schreiner</i>	Date 10/19/04
7. Checker	Printed Name Susan LeStrange	Signature <i>Susan LeStrange</i>	Date 10/19/04
8. QER	Printed Name Judith Gebhardt <i>JGB</i>	Signature <i>Judith Gebhardt</i>	Date 10/21/04
9. Responsible Manager/Lead	Printed Name Dennis Thomas <i>10/21/04</i>	Signature <i>Dennis Thomas</i>	Date 10/21/04
10. Responsible Manager	Printed Name Neil Brown	Signature <i>Neil Brown</i>	Date 10/21/04
11. Remarks			

Change History

12. Revision No.	13. Description of Change
00	Initial Issue.
00/01	Changed corrosion FEPs to be consistent with the Waste Form PMR. Ensured consistence of FEPs summary in Waste Form PMR. 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
01	Modified report to be consistent with new procedures including AP-SIII.9Q. Revised FEP names, numbers, and description to be consistent with the FEPs selected for LA. Revised screening arguments.
02	Complete revision to address issues raised during regulatory integration. Change bars were not used because changes were too extensive.

INTENTIONALLY LEFT BLANK

CONTENTS

	Page
1. PURPOSE	1-1
1.1 PLANNING AND DOCUMENTATION	1-1
1.2 SCOPE	1-1
1.3 SCIENTIFIC ANALYSIS LIMITATIONS AND USE	1-3
2. QUALITY ASSURANCE	2-1
3. USE OF SOFTWARE	3-1
4. INPUTS.....	4-1
4.1 DIRECT INPUTS	4-1
4.2 CRITERIA	4-3
4.2.1 Criteria from the Projects Requirements Document and the Yucca Mountain Review Plan.....	4-3
4.2.2 FEPs Screening Criteria.....	4-4
4.3 CODES, STANDARDS, AND REGULATIONS	4-6
5. ASSUMPTIONS.....	5-1
5.1 ANNUAL-EXCEEDANCE PROBABILITY	5-1
5.2 NATURALLY OCCURRING EVENTS	5-1
5.3 APPLICABILITY OF REGULATORY REQUIREMENTS.....	5-2
6. SCIENTIFIC ANALYSIS DISCUSSION.....	6-1
6.1 METHODS AND APPROACH	6-1
6.1.1 Features, Events, and Processes Identification and Classification	6-1
6.1.2 Features, Events, and Processes Screening.....	6-6
6.1.3 Supporting Reports and Inputs.....	6-7
6.1.5 Assumptions and Simplifications	6-7
6.1.6 Intended Use and Limitations	6-7
6.2 CLADDING FEPS SCREENING AND ANALYSES	6-8
6.2.1 Degradation of Cladding from Waterlogged Rods (2.1.02.11.0A).....	6-8
6.2.2 Degradation of cladding prior to disposal (2.1.02.12.0A)	6-10
6.2.3 General Corrosion of Cladding (2.1.02.13.0A)	6-11
6.2.4 Microbially Influenced Corrosion (MIC) of Cladding (2.1.02.14.0A).....	6-14
6.2.5 Localized (Radiolysis Enhanced) Corrosion of Cladding (2.1.02.15.0A).....	6-16
6.2.6 Localized (Pitting) Corrosion of Cladding (2.1.02.16.0A)	6-20
6.2.7 Localized (Crevice) Corrosion of Cladding (2.1.02.17.0A)	6-23
6.2.8 Enhanced Corrosion of Cladding from Dissolved Silica (2.1.02.18.0A)	6-24
6.2.9 Creep Rupture of Cladding (2.1.02.19.0A).....	6-25
6.2.10 Internal Pressurization of Cladding (2.1.02.20.0A).....	6-27
6.2.11 Stress Corrosion Cracking (SCC) of Cladding (2.1.02.21.0A).....	6-28
6.2.12 Hydride Cracking of Cladding (2.1.02.22.0A)	6-30
6.2.13 Cladding Unzipping (2.1.02.23.0A)	6-33

CONTENTS (Continued)

	Page
6.2.14 Mechanical Impact on Cladding (2.1.02.24.0A).....	6-33
6.2.15 Naval SNF Cladding (2.1.02.25.0B).....	6-35
6.2.16 Diffusion-Controlled Cavity Growth in Cladding (2.1.02.26.0A).....	6-35
6.2.17 Localized (Fluoride Enhanced) Corrosion of Cladding (2.1.02.27.0A)	6-37
6.2.18 Rockfall (2.1.07.01.0A)	6-39
6.2.19 Volume Increase of Corrosion Products Impacts Cladding (2.1.09.03.0A) ...	6-40
6.2.20 Electrochemical Effects in EBS (2.1.09.09.0A)	6-40
6.2.21 Chemical Effects of Waste-Rock Contact (2.1.09.11.0A)	6-43
6.2.22 Thermal Expansion/Stress of In-Package EBS Components (2.1.11.05.0A) .	6-44
6.2.23 Gas Generation (He) From Waste Form Decay (2.1.12.02.0A)	6-45
6.2.24 Gas Generation (H ₂) From Waste Package Corrosion (2.1.12.03.0A)	6-47
7. CONCLUSIONS.....	7-1
7.1 YMRP ACCEPTANCE CRITERIA.....	7-2
7.1.1 Scenario Analysis Acceptance Criteria.....	7-2
8. INPUTS AND REFERENCES.....	8-1
8.1 DOCUMENTS CITED	8-1
8.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES.....	8-8
8.3 SOURCE DATA, LISTED BY DATA TRACKING NUMBER	8-9
APPENDIX A CLADDING TEMPERATURE ESTIMATES FOR POSTCLOSURE.....	A-1
APPENDIX B QUALIFICATION OF TECHNICAL INFORMATION THAT SUPPORTS CLAD DEGRADATION - FEPs SCREENING ARGUMENTS (ANL-WIS-MD-000008)	B-1

FIGURES

	Page
6-1. The Iso-Corrosion Diagram for Zirconium in Sulfuric Acid (H ₂ SO ₄)	6-12
6-2. pH Profiles Showing Radiolysis Impacts on pH.....	6-18
6-3. Creep Failure Fraction as a Function of Peak Cladding Temperature.....	6-27
6-4. Fracture Toughness vs. Hydrogen Content of Zircaloy-4	6-32

TABLES

	Page
1-1. Cladding Degradation Features, Events, and Processes	1-2
4-1. Direct Inputs.....	4-2
4-2. Relationships of Regulations to the Project Requirements and the YMRP Acceptance Criteria.....	4-4
6-1. Indirect Inputs	6-2
6-2. Chloride, Ferric Iron, and Hydrogen Peroxide Molality.....	6-18
6-3. Half-Life of Hydrogen Peroxide at Various Temperatures	6-19
6-4. Corrosion and Repassivation Potentials for In-Package Chemistry Conditions.....	6-21
6-5. Compositions of J-13 Well Water.....	6-38
7-1. Summary of Cladding Degradation and Waste Form FEPs.....	7-1

INTENTIONALLY LEFT BLANK

ACRONYMS

ANL	Argonne National Laboratory
ASTM	American Society for Testing and Materials
BWR	boiling water reactor
CCDF	Complementary Cumulative Distribution Function
CSNF	commercial spent nuclear fuel
DCCG	diffusion controlled cavity growth
DHC	delayed hydride cracking
DOE	U.S. Department of Energy
DHLW	defense high-level waste
DSNF	DOE-owned spent nuclear fuel
EBS	engineered barrier system
FEPs	features, events, and processes
HLW	high-level waste
IAEA	International Atomic Energy Agency
INEEL	Idaho National Engineering and Environmental Laboratory
kgU	kilogram uranium
KTI	key technical issue
LA	License Application
LWR	light water reactor
MIC	microbially influenced corrosion
MWd	megawatt days
NRC	U.S. Nuclear Regulatory Commission
OCRWM	Office of Civilian Radioactive Waste Management
PWR	pressurized water reactor
RMEI	reasonably maximally exposed individual
SCC	stress corrosion cracking
SCE	saturated calomel electrode scale
SNF	spent nuclear fuel
SR	Site Recommendation
SRB	sulfate reducing bacteria
STP	standard temperature and pressure
TSPA	Total System Performance Assessment
TSPA/I	Total System Performance Assessment and Integration
TSPA-LA	Total System Performance Assessment – License Application
TSPA-VA	Total System Performance Assessment – Viability Assessment
YMP	Yucca Mountain Project

INTENTIONALLY LEFT BLANK

1. PURPOSE

The purpose of this report is to evaluate and document the screening of the clad degradation features, events, and processes (FEPs) with respect to modeling used to support the Total System Performance Assessment – License Application (TSPA-LA). This report also addresses the effect of certain FEPs on both the cladding and the commercial spent nuclear fuel (CSNF), DOE-owned spent nuclear fuel (DSNF), and defense high-level waste (DHLW) waste forms, as appropriate to address the effects on multiple materials and both components (FEPs 2.1.09.09.0A, 2.1.09.11.0A, 2.1.11.05.0A, 2.1.12.02.0A, and 2.1.12.03.0A). These FEPs are expected to affect the repository performance during the postclosure regulatory period of 10,000 years after permanent closure.

Table 1-1 provides the list of cladding FEPs, including their screening decisions (include or exclude). The primary purpose of this report is to identify and document the analysis, screening decision, and TSPA-LA disposition (for included FEPs) or screening argument (for excluded FEPs) for these FEPs related to clad degradation. In some cases, where a FEP covers multiple technical areas and is shared with other FEP reports, this report may provide only a partial technical basis for the screening of the FEP. The full technical basis for shared FEPs is addressed collectively by the sharing FEP reports. The screening decisions and associated TSPA-LA dispositions or screening arguments from all of the FEP reports are cataloged in a project-specific FEPs database.

1.1 PLANNING AND DOCUMENTATION

Documentation requirements for this report are described in *Technical Work Plan for: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package* (BSC 2004 [DIRS 171583]). Any changes in the assigned cladding degradation FEP list for TSPA-LA that resulted from the planned work scope are described in Section 6.1.

1.2 SCOPE

The scope of this report is to describe, evaluate, and document screening decisions and technical bases for the cladding degradation FEPs for TSPA-LA. For FEPs included in the TSPA-LA, this report provides a consolidated summary of how the FEP has been included and addressed in the TSPA-LA model based on the supporting analysis and model reports. It also provides a list of the reports that provide detailed discussions of the FEP. For FEPs excluded from the TSPA-LA, this report provides a screening argument, which identifies the basis for the screening decision (i.e., low probability, low consequence, or by regulation), and discusses the technical bases that supports that decision. It also provides appropriate references to project and non-project documentation that supports the exclusion. This information is provided in Section 6.2 and subsequent sections and subsections.

An overview of the YMP FEP analysis and scenario development process is available in *The Development of the TSPA-LA Features, Events, and Processes* (BSC 2004 [DIRS 168706], Sections 2.4, 3, and 4), describing the TSPA-LA FEP identification and screening process. As part of that process, the TSPA-LA FEP list (DTN: MO0407SEPFEPLA.000 [DIRS 170760]) was developed. This DTN was used as an initial input to the cladding degradation FEP analysis.

The list of cladding degradation TSPA-LA FEPs, presented in Table 1-1, was derived from DTN: MO0407SEPFPLA.000 [DIRS 170760] with subsequent modifications to the FEP list, names, or descriptions. These modifications are documented in the “FEP History File” in the FEP database (BSC 2004 [DIRS 168706], Table 6-2) and will be incorporated into a subsequent revision of the TSPA-LA FEP list (Section 7). Table 1-1 also includes the designation of shared FEPs.

Direct inputs supporting the screening decisions are listed in Table 4-1. Indirect inputs supporting the screening decisions are listed in Table 6-1. The individual FEP discussions providing identification (FEP number, name, and description) and screening (screening decision, screening argument or TSPA-LA disposition) information are in Section 6.2.

This report addresses cladding degradation FEPs, which represent the key phenomena that result in degradation of cladding and, in some cases, the CSNF pellets.

Table 1-1. Cladding Degradation Features, Events, and Processes

FEP Number	FEP Name	Addressed in Section	Sharing FEP With
2.1.02.11.0A	Degradation of cladding from waterlogged rods	6.2.1	
2.1.02.12.0A	Degradation of cladding prior to disposal	6.2.2	
2.1.02.13.0A	General corrosion of cladding	6.2.3	
2.1.02.14.0A	Microbially influenced corrosion (MIC) of cladding	6.2.4	
2.1.02.15.0A	Localized (radiolysis enhanced) corrosion of cladding	6.2.5	
2.1.02.16.0A	Localized (pitting) corrosion of cladding	6.2.6	
2.1.02.17.0A	Localized (crevice) corrosion of cladding	6.2.7	
2.1.02.18.0A	Enhanced corrosion of cladding from dissolved silica	6.2.8	
2.1.02.19.0A	Creep rupture of cladding	6.2.9	
2.1.02.20.0A	Internal pressurization of cladding	6.2.10	
2.1.02.21.0A	Stress corrosion cracking (SCC) of cladding	6.2.11	
2.1.02.22.0A	Hydride cracking of cladding	6.2.12	
2.1.02.23.0A	Cladding unzipping	6.2.13	
2.1.02.24.0A	Mechanical impact on cladding	6.2.14	
2.1.02.25.0B	Naval SNF Cladding	6.2.15	
2.1.02.26.0A	Diffusion-controlled cavity growth in cladding	6.2.16	
2.1.02.27.0A	Localized (fluoride enhanced) corrosion of cladding	6.2.17	
2.1.07.01.0A	Rockfall	6.2.18	WP, EBS
2.1.09.03.0A	Volume increase of corrosion products impacts cladding	6.2.19	
2.1.09.09.0A	Electrochemical effects in EBS	6.2.20	WP
2.1.09.11.0A	Chemical effects of waste-rock contact	6.2.21	
2.1.11.05.0A	Thermal expansion/stress of in-package EBS components	6.2.22	
2.1.12.02.0A	Gas generation (He) from waste form decay	6.2.23	EBS
2.1.12.03.0A	Gas generation (H ₂) from waste package corrosion	6.2.24	WP, EBS

NOTE: WP = waste package EBS = engineered barrier system

1.3 SCIENTIFIC ANALYSIS LIMITATIONS AND USE

The intended use of this report is to provide FEP screening information for a project-specific FEP database, and to promote traceability and transparency for included and excluded cladding degradation FEPs. It is intended for use as the documentation for inclusion or exclusion of cladding degradation FEPs within or from the TSPA-LA model. The following limitations apply to this report:

- Because this report cites other documents as direct input, the limitations of this report inherently include any limitations or constraints described in the cited AMRs or controlled documents.
- In cases where FEPs are shared, the scope of this report is limited to the degradation of the cladding of the spent nuclear fuel. The full technical basis for these shared FEPs is addressed, collectively, by all of the sharing FEP reports.
- For screening purposes, this report generally uses mean values of probabilities, mean amplitude of events, or mean value of consequences (e.g., mean time to waste package degradation) as a basis for reaching an include/exclude decision. Mean values are determined based on the range of possible values.

Changes in direct inputs listed in Section 4.1, in baseline conditions used for this evaluation, or in other subsurface conditions, are required to be evaluated to determine whether the changes are within the limits stated in the FEP evaluations. Engineering and design changes are subject to evaluation to determine whether there are any adverse impacts to safety, as required by 10 CFR 63.73 Subparts F and G [DIRS 156605] (see also the requirements in 10 CFR 63.44 [DIRS 156605]).

This report is limited to nuclear fuel exposed to normal operation and anticipated operational occurrences (i.e., events that are anticipated to occur within a reactor lifetime) prior to receipt at the repository, 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 percent for material shipment and fuel manufacturing, shipment, and storage), cladding oxide coating thickness, and rod plenum pressures. This is consistent with the projections for advanced reloads. The fuel considered has burnup up to 75 MWd/kgU and half of the fuel is assumed to have a burn-up of greater than above 44.7 MWd/kgU, which is consistent with the typical burn-up range of fuel in use in 2000 and beyond. Ranges and uncertainties have been defined in *Initial Cladding Condition* (CRWMS M&O 2000 [DIRS 151659], Section 6.2). The information presented in this report is used to determine which cladding degradation mechanisms are addressed by TSPA-LA.

The CSNF cladding performance is used to bound naval fuel cladding behavior. Cladding degradation for DOE-owned spent nuclear fuel (DSNF) is not addressed in this report, but is addressed in a general waste form screening report *Waste-Form Features, Events, and Processes* (BSC 2004 [DIRS 170020]).

This report is directly supported by input from several model and analysis reports: *Clad Degradation – Summary and Abstraction for LA*, *CSNF Waste Form Degradation: Summary Abstraction*, *Defense HLW Glass Degradation Model*, *In-Package Chemistry Abstraction*, *Multiscale Thermohydrologic Model*, and *Pitting Model for Zirconium-Alloyed Cladding*. This report does not directly feed any downstream reports.

2. QUALITY ASSURANCE

Development of this report and its supporting analyses are subject to the Office of Civilian Radioactive Waste Management (OCRWM) quality assurance (QA) program as identified in *Technical Work Plan For: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package* (BSC 2004 [DIRS 171583], Section 8.1, Work Package ARTM03).

Cladding has been determined to be important to waste isolation in accordance with AP-2.22Q, *Classification Analysis and Maintenance of the Q-List*, and, therefore, is classified as Safety Category (SC) in *Q-List* (BSC 2004 [DIRS 168361], Appendix A). The technical work plan contains the Process Control Evaluation used to evaluate the control of electronic management of data (BSC 2004 [DIRS 171583], Attachment I) during modeling and documentation activities. This evaluation determined that the methods identified in the implementing procedures are adequate. There were no deviations from these methods. This report was prepared using AP-SIII.9Q, *Scientific Analyses*.

INTENTIONALLY LEFT BLANK

3. USE OF SOFTWARE

This analysis report uses no computational software; therefore, this analysis is not subject to software controls. The analyses and arguments presented herein are based on guidance and regulatory requirements, results of analyses presented and documented in other reports or on other technical literature. Software and models used in the supporting documents are cited in this analysis report for traceability and transparency purposes, but were not used in its development.

This analysis report was developed using only commercial-off-the-shelf software (Microsoft Word 2000) for word processing, which is controlled under the software configuration management system, but is not required to be qualified or documented per LP-SI.11Q-BSC, *Software Management*. No additional applications (routines or macros) were developed using this commercial off-the-shelf software.

INTENTIONALLY LEFT BLANK

4. INPUTS

AP-3.15Q, *Managing Technical Product Inputs* categorizes the use of technical product input as either direct or indirect input. Direct input is used to develop the results or conclusions in a technical product. Indirect input is used to provide additional information that is not used in the development of results or conclusions. Direct inputs are addressed in this section. Indirect inputs are addressed in Section 6.1.3.

Section 4.1 identifies all direct inputs used in this FEP report. The direct inputs were obtained from controlled source documents and other appropriate sources in accordance with AP-3.15Q, *Managing Technical Product Inputs*. Section 4.2 identifies the FEP screening criteria described at 10 CFR Part 63 [DIRS 156605] along with the required screening criteria. Section 4.3 identifies applicable codes, standards, and regulations.

4.1 DIRECT INPUTS

The TSPA-LA FEP list (DTN: MO0407SEPFEPLA.000 [DIRS 170760]) was used as a direct input to provide the initial list of cladding degradation FEPs for screening in this report. The TSPA-LA FEP list identifies a FEP report or a set of sharing reports for each FEP. Subsequent additions to or changes from that list (numbers, names, or descriptions) are reflected in the information provided in Section 6.2 and can be traced through the “FEP History File” in the FEP database (BSC 2004 [DIRS 168706], Table 6-2).

The direct inputs used for the FEP screening in this report are listed in Table 4-1. The direct inputs listed in Table 4-1 have been evaluated and have been determined to be appropriate for their intended use of providing the necessary information for the disposition of the cladding FEPs. The data from sources requiring justification per the requirements of AP-SIII.9Q or AP-SIII.2Q are considered qualified for their intended use per the documentation in Appendix B.

This report does not utilize any previously developed and validated model to complete this report.

Table 4-1. Direct Inputs

Technical Product Input	Used From	Used In	Input Description
10 CFR 63 [DIRS 156605]	Part 114 (d, e, f, g), part 342	1.3, 4., 4.2.1, 4.2.2, 4.3, 5.2, 5.3, 6.1.2, Table 4-1, Table 4-2, Appendix B	FEP Screening requirements
BSC 2004 [DIRS 170023]	Entire, 6.2.1, 6.2.4, 6.2.3, 6.0, 6.24	Table 4-1, 6., 6.2.1, 6.2.2, 6.2.13, 6.2.15, 6.2.19, 6.2.23, 7.	Inclusion arguments for cladding degradation and cladding degradation mechanisms, fuel rod characteristics, Fuel rods w/breached cladding
BSC 2004 [DIRS 169987]	6.2.1	Table 4-1, 6.2.22	Fuel pellet fragmentation
BSC 2004 [DIRS 166692]	Entire	Table 4-1, 6.2.12, Appendix A	Internal Temperature across WP
BSC 2004 [DIRS 169988]	Section 7.5.3	Table 4-1, 6.2.22	Melting temperature of glass
BSC 2004 [DIRS 169991]	Section 7.1	Table 4-1, 6.2.4	Due to the environmental constraints, the microbial activity in the YMP is expected to be low, and its impacts on drift chemistry can be insignificant
BSC 2004 [DIRS 169997]	6.2	Table 4-1, 6.2.14, 6.2.18	Information on WP FEPs
BSC 2004 [DIRS 167621]	Appendix B, 6.3	6.2.3, 6.2.5, 6.2.17, 6.2.20, 6.2.21, Table 4-1	In-package chemistry with radiolysis
BSC 2004 [DIRS 169565]	6.2, Figure 6.3-67	Table 4-1, 6.2.23, Appendix A	Temperature across WP surface
BSC 2004 [DIRS 170043]	Section 8.2, Entire	6.2.3, 6.2.4, 6.2.6, 6.2.20, 6.2.21, Table 4-1	Pitting potentials, chemical environment where pitting is expected
Greene et al. 2000 [DIRS 145073]	Entire, p.7	Table 4-1, 6.2.7, Appendix B	Resistance to crevice corrosion
Hansson 1984 [DIRS 101676]	p. 6	Table 4-1, 6.2.20, Appendix B	Rapid passivation
Hayes et al. 1999 [DIRS 164598]	Figures 2,5,6,8,11, Entire	Table 4-1, 6.2.16, Appendix B	Diffusion controlled cavity growth is unlikely if $T < 330^{\circ}\text{C}$ to 440°C
Hillner et al. 1998 [DIRS 100455]	Entire, p. 6, p. 9, p. 25, Fig. 5, Table 4, p. 11	Table 4-1, 6.2.3, 6.2.4 Appendix B	General corrosion of zirconium, MIC
IAEA 1998 [DIRS 150560]	p. 92	Table 4-1, 6.2.12, 6.2.24	H_2 not absorbed through passive layer
M00407SEPFEPLA.000 [DIRS 170760]	Entire	1.2, 4.1, 6.1.1, Table 4-1	FEP numbers, titles, and descriptions
NRC 2002 [DIRS 164593]	p. 2	Table 4-1, 6.2.9, Appendix B	Little creep damage below 400°C
NRC 1997 [DIRS 101903]	Sec. 8.V.1	Table 4-1, 6.2.1	Waterlogging not a problem
Piron and Pelletier 2001 [DIRS 165318]	5.3, Entire, p. 232	Table 4-1, 6.2.10, 6.2.23, Appendix B	Helium pressurization
Wolfram et al. 1996 [DIRS 165268]	pp. iii, iv	Table 4-1, 6.2.4, Appendix B	MIC and Zirconium
Yau and Webster 1987 [DIRS 100494]	Table 15, p. 718, Table 6, Entire	Table 4-1, 6.2.7, 6.2.8, 6.2.11, 6.2.20, Appendix B	Corrosion properties of zirconium, Galvanic corrosion

4.2 CRITERIA

The technical work plan for this activity (BSC 2004 [DIRS 171583], Table 3-1) has identified the acceptance criteria (AC) applicable to this report based on the requirements of *Project Requirements Document* (PRD) (Canori and Leitner 2003 [DIRS 166275]) and *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]). The technical work plan (BSC 2004 [DIRS 171583], Table 3-1) identifies two additional acceptance criteria (2.2.1.1 and 2.2.1.3.4). This report deviates from the technical work plan (BSC 2004 [DIRS 171583]) in that these acceptance criteria were deemed to be germane to the individual AMRs and not to this report.

4.2.1 Criteria from the Projects Requirements Document and the Yucca Mountain Review Plan

The licensing criteria for postclosure performance assessment are stated in 10 CFR 63.114 [DIRS 156605]. The requirements to be satisfied by TSPA-LA are identified in *Project Requirements Document* (PRD) (Canori and Leitner 2003 [DIRS 166275]). The acceptance criteria (AC) that will be used by the NRC to evaluate the adequacy of technical arguments are identified in *Yucca Mountain Review Plan, Final Report* (YMRP) (NRC 2003 [DIRS 163274]). Table 4-2 provides a crosswalk between the regulatory requirements, the PRD (Canori and Leitner 2003 [DIRS 166275]) and the acceptance criteria provided in the Yucca Mountain review plan (YMRP) (NRC 2003 [DIRS 163274], Sections 2.2.1.2.1.3 and 2.2.1.2.2.3).

Project Requirements Document (Canori and Leitner 2003 [DIRS 166275]) documents and categorizes regulatory and other project requirements. The regulatory requirements include criteria relevant to performance assessment activities, in general, and to FEP-related activities as they pertain to TSPA-LA, in particular. In Table 4-2, YMRP acceptance criteria are correlated to the corresponding regulations as they pertain to FEPs-related criteria.

The basis of the NRC review is described in the YMRP (NRC 2003 [DIRS 163274], Section 2.2.1.2), and the bases for acceptance are stated as acceptance criteria. The following acceptance criteria that apply to this AMR are based on meeting the requirements of 10 CFR 63.114 (a), (b), (d), (e), and (f) [DIRS 156605] from Section 2.2.1.2.1.3 of the YMRP (NRC 2003 [DIRS 163274])

- AC 1: The Identification of a List of Features, Events, and Processes Is Adequate
- AC 2: Screening of the List of Features, Events, and Processes Is Appropriate.

Section 7.1 describes how applicable acceptance criteria are addressed. Acceptance criteria for FEP screening reiterate the regulatory screening criteria of low probability and low consequence but also allow for exclusion of a FEP if the process is specifically excluded by the regulations (Section 4.2.2).

Acceptance criteria listed in Section 2.2.1.2.2.3 of the YMRP (NRC 2003 [DIRS 163274]) pertaining to identification of events with probability greater than 10^{-8} per year are not considered because this analysis does not develop probabilities for such events.

Table 4-2. Relationships of Regulations to the Project Requirements and the YMRP Acceptance Criteria

Description of the Applicable Regulatory Requirement or Acceptance Criterion	10 CFR Part 63 [DIRS 156605]	Canori and Leitner 2003 [DIRS 166275]	Associated Criteria in the YMRP [DIRS 163274]
	Regulatory Citation	Associated PRD	
General Requirements and Scope Pertinent to FEP Screening			
Include data related to geology, hydrology, geochemistry, and geophysics	63.114(a)	PRD-002/T-015	2.2.1.2.1.3 Acceptance Criterion 1
Include information of the design of the engineered barrier system used to define parameters and conceptual models	63.114(a)	PRD-002/T-015	2.2.1.2.1.3 Acceptance Criterion 1
Account for uncertainties and variabilities in parameter values and provide the technical basis for parameter ranges, probability distributions, or bounding values	63.114(b)	PRD-002/T-015	2.2.1.2.2.3 Acceptance Criteria 2 and 5
FEP Screening Criteria			
Provide the justification and technical basis for excluding FEPs specifically excluded by regulation.	Not Applicable	Not Applicable	2.2.1.2.1.3 Acceptance Criterion 2
Provide the technical basis for either inclusion or exclusion of FEPs. Provide the justification and technical basis for those excluded based on probability.	63.114(d) 63.342	PRD-002/T-015 PRD-002/T-034	2.2.1.2.1.3 Acceptance Criterion 2 2.2.1.2.2.3 Acceptance Criteria 1 and 2
Provide the technical basis for either inclusion or exclusion of FEPs. Provide the justification and the technical basis for those excluded based on lack of significant change in resulting radiological exposure or release to the accessible environment.	63.114 (e and f) 63.342	PRD-002/T-015 PRD-002/T-034	2.2.1.2.1.3 Acceptance Criterion 2 2.2.1.2.2.3 Acceptance Criteria 1 and 2

FEPs = features, events, and processes; PRD = Project Requirements Document

4.2.2 FEPs Screening Criteria

The NRC regulations and guidance specifically allow the exclusion of FEPs from the TSPA-LA if they can be shown to be of low probability or of low consequence. Additionally, FEPs can be excluded based on the constraints provided within 10 CFR 63 [DIRS 156605]. In this document, this exclusion is called “exclusion by regulation.” FEPs screening criteria are described further in the following three subsections.

4.2.2.1 Exclusion by Low Probability

The low-probability criterion is stated at 10 CFR 63.114(d) [DIRS 156605]:

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

and supported by 10 CFR 63.342 [DIRS 156605]:

DOE's performance assessments shall not include consideration of very unlikely features, events, or processes, i.e., those that are estimated to have less than one chance in 10,000 of occurring within 10,000 years of disposal.

The low-probability criterion (i.e., very unlikely FEPs) is stated as less than one chance in 10,000 of occurring in 10,000 years.

Furthermore, it is stated at 10 CFR 63.342 [DIRS 156605] that:

DOE's assessments for the human intrusion and groundwater protection standards shall not include consideration of unlikely features, events, or processes, or sequences of events and processes, i.e., those that are estimated to have less than one chance in 10 and at least one chance in 10,000 of occurring within 10,000 years of disposal.

4.2.2.2 Exclusion by Low Consequence

The low consequence criteria are stated at 10 CFR 63.114 (e and f) [DIRS 156605]:

(e) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes in the performance assessment. Specific features, events, and processes must be evaluated in detail if the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, 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 radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, would be significantly changed by their omission.

and supported by 10 CFR 63.342 [DIRS 156605]:

DOE's performance assessments need not evaluate the impacts resulting from any features, events, and processes or sequences of events or processes with a higher chance of occurrence if the results of the performance assessments would not be changed significantly.

The terms "significantly changed" and "changed significantly" are undefined terms in the NRC regulations. The absence of "significant change" is inferred for FEP 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, transport times, or other measures as well as overall expected annual dose), there is no single quantitative test of “significance.”

4.2.2.3 Exclusion by Regulation

The provisions and constraints provided within 10 CFR Part 63 [DIRS 156605] pertaining to the reference biosphere, receptor, and performance assessment serve as the basis for exclusion of some FEPs. This process of screening out the FEPs that fall outside the parameters established by 10 CFR Part 63 [DIRS 156605] is described in *Yucca Mountain Review Plan, Review Plan* (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3, Acceptance Criterion 2) together with the screening criteria of low probability and low consequence:

An acceptable justification for excluding features, events, and processes is that either the feature, event, and process is specifically excluded by regulation; probability of the feature, event, and process (generally an event) falls below the regulatory criterion; or omission of the feature, event, and process does not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment.

Exclusion of FEPs by regulation involves consideration of those portions of 10 CFR Part 63 [DIRS 156605] that define requirements and key concepts for performance assessment. In this context, portions of 10 CFR Part 63 [DIRS 156605] serve as criteria for screening related FEPs.

4.3 CODES, STANDARDS, AND REGULATIONS

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 Geological Disposal of High-Level Radioactive Waste* (ASTM 1998 [DIRS 105725]) is used to support the degradation model development methodology, categorize the model developed with respect to its usage for long-term TSPA, and relate the information or data used to develop the model to the requirements of the standard.

The applicable requirements given in 10 CFR Part 63 [DIRS 156605] are discussed in Sections 4.1 and 4.2.

5. ASSUMPTIONS

This section addresses assumptions used in the FEP screening for the cladding degradation FEPs.

5.1 ANNUAL-EXCEEDANCE PROBABILITY

Assumption: For naturally occurring FEPs, it is assumed that regulations expressed as a probability criterion can also be expressed as an annual-exceedance probability, which is defined as the probability that a specified value (such as for ground motions or fault displacement) will be exceeded during one year. More specifically, a stated probability screening criterion of one chance in 10,000 in 10,000 years ($10^{-4}/10^4$ yr) criterion is assumed equivalent to a 10^{-8} annual-exceedance probability.

Rationale: The definition of annual exceedance probability, and the following justification for this assumption is taken from *Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada* (BSC 2004 [DIRS 168030], Glossary).

The assumption of equivalence of annual-exceedance probability is appropriate if the possibility of an event is equal for any given year. This satisfies the definition of a Poisson distribution as "...a mathematical model of the number of outcomes obtained in a suitable interval of time and space, that has its mean equal to its variance..." (Merriam-Webster 1993 [DIRS 100468], p. 899). This is inferred to mean that naturally occurring, infrequent, and independent events can be represented as stochastic processes in which distinct events occur in such a way that the number of events occurring in a given period of time depends only on the length of the time period. The use of this assumption is justified in *Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada* (BSC 2004 [DIRS 168030], Section 6.4.2), which indicates that assuming the behavior of the earth is generally Poissonian or random is the underlying assumption in all probabilistic hazard analyses.

Confirmation Status: Although there may be cases where sufficient data and information exist to depart from this assumption, the Poissonian model is generally an effective representation of nature and represents a compromise between the complexity of natural processes, availability of information, and the sensitivity of results of engineering relevance. Consequently, for geologic processes that occur over long time spans, assuming annual equivalence over a 10,000-year regulatory period (a relatively short time span for geologic-related events) is reasonable and consistent with the basis of probabilistic hazard analyses. Therefore, no further confirmation is required.

Use in the Analysis: This assumption is used throughout. It is particularly germane to FEPs related to processes or phenomena that have or are relative to annual-exceedance probability.

5.2 NATURALLY OCCURRING EVENTS

Assumption: It is assumed that potential naturally occurring events, but perhaps of different magnitude, have occurred at least once in the past within the geologic record used as the basis for determining that factors that could affect the Yucca Mountain disposal system over the 10,000-year regulatory period.

Rationale: This assumption is justified because it is consistent with the regulations used as direct input. At 10 CFR 63.305(c) [DIRS 156605], DOE is directed to “vary factors related to the geology, hydrology, and climate based upon cautious, but reasonable assumptions consistent with present knowledge of factors that could affect the Yucca Mountain disposal system over the next 10,000 years.”

The implication of this assumption is that any discernible impacts or processes related to past events on the site setting are reflected in the present knowledge of natural processes that form the basis of the TSPA-LA. If the subject FEP phenomena are not reflected or discernible in the data used to describe past settings, then they are either of low consequence or of low probability and can be excluded from consideration.

Confirmation Status: Because this assumption is consistent with the regulations, no further confirmation is necessary.

Use in the Analysis: This assumption is used throughout. It is particularly germane to FEPs related to processes or phenomena that, speculatively, could affect future states of the system, but for which the magnitude and/or coupling to the effect on the repository is not well defined, or for which consequences in present time are known to be minor.

5.3 APPLICABILITY OF REGULATORY REQUIREMENTS

Assumption: It is assumed that the repository will be constructed, operated, and closed according to the regulatory requirements applicable to the construction, operation, and closure period. It is further assumed that deviations from the design that is approved by the NRC will be identified and corrected.

Rationale: Inherent in the FEPs evaluation approach is the assumption that the repository will be constructed, operated, and closed according to the design used as the basis for this FEP screening and that serves as the basis for the NRC’s review. This is inherent in performance evaluation of any engineering project, and design verification and performance confirmation are required as part of the construction and operation processes. Therefore, no further confirmation of the assumption is required.

Engineering and design changes are subject to evaluation to determine if there are any adverse impacts to safety as required by 10 CFR 63.73 and in Subparts F and G [DIRS 156605]. See also the requirements at 10 CFR 63.32, 10 CFR 63.44, and 10 CFR 63.131 [DIRS 156605].

These regulations require the DOE to develop and submit periodic and special reports regarding:

1. Progress of construction
2. Any data about the site, obtained during construction, that are not within the predicted limits on which the facility design was based
3. Any deficiencies in design and construction that, if uncorrected, could adversely affect safety at any future time

4. Results of research and development programs being conducted to resolve safety questions.

Confirmation Status: No confirmation is required for this assumption.

Use in the Analysis: Any changes in direct inputs listed in Section 4.1, in baseline conditions used for this evaluation, or in other subsurface conditions, must be evaluated to determine if the changes are within the limits stated in the FEP evaluations.

INTENTIONALLY LEFT BLANK

6. SCIENTIFIC ANALYSIS DISCUSSION

This section discusses each of the 24 FEPs dealing with cladding degradation (Table 1-1). The purpose of this report is to document the screening of the cladding degradation FEPs. For included FEPs, a summary of their implementation in TSPA-LA is presented here. For excluded FEPs, the arguments for their exclusion are documented here.

Appendix A of this report contains an evaluation of the temperatures expected for the cladding needed to evaluate some of the FEPs. For FEPs included in the TSPA-LA, alternative conceptual models are discussed in the specific report for the included FEP (BSC 2004 [DIRS 170023]). For excluded FEPs, the discussions of alternative conceptual models, where appropriate, are included in this report. The output of this report is used to support the project-specific FEP database and to promote traceability and transparency regarding FEP screening.

6.1 METHODS AND APPROACH

The identification and screening of a comprehensive list of FEPs potentially relevant to repository postclosure performance are based on site-specific information, design, and regulations. This report uses the following definitions, as taken from *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274], Glossary):

- Feature – An object, structure, or condition that has a potential to affect disposal system performance.
- Event – A natural or human-caused phenomenon that has a potential to affect disposal system performance and that occurs during an interval that is short compared to the period of performance.
- Process – A natural or human-caused phenomenon that has a potential to affect disposal system performance and that operates during all or a significant part of the period of performance.

FEP analysis for TSPA-LA is described in *The Development of the TSPA-LA Features, Events, and Processes* (BSC 2004 [DIRS 168706]). It is summarized in the following sections.

6.1.1 Features, Events, and Processes Identification and Classification

The first step of FEP analysis is FEP identification and classification, which addresses Acceptance Criterion 1 of *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3). The TSPA-LA FEP identification and classification process is described in *The Development of the TSPA-LA Features, Events, and Processes* (BSC 2004 [DIRS 168706], Section 3). This process produced the version of the LA FEP list (DTN: MO0407SEPFEPLA.000 [DIRS 170760] used as initial input in this waste form FEP report. Subsequent modifications to the FEP list from the information shown in DTN: MO0407SEPFEPLA.000 [DIRS 170760], aside from editorial corrections to FEP descriptions, are discussed later in this section. All subsequent modifications are also documented in the “FEP History File” in the FEP database (BSC 2004 [DIRS 168706], Table 6-2).

Table 6-1. Indirect Inputs

Technical Product Input	Used From	Used In	Input Description
10 CFR 60 [DIRS 103540]	Entire	Table 6-1, Appendix B	QA Requirements
10 CFR 63 [DIRS 156605]	Entire	1.3, 5.2, 5.3, 6.1.6, 6.2, 7.1, Table 6-1	FEPs screening
Adler Flitton et al. 2002 [DIRS 161991]	p. 4	Table 6-1, 6.2.4	No MIC observed
	p. 4	6.2.21,	Corrosion and soil contact
ASTM C 1174-97. 1998 [DIRS 105725]	Entire	4.3, Table 6-1	Criteria, Codes, or Standards
Baker 1992 [DIRS 149104]	pp. 2-326, Entire	Table 6-1, 6.2.24	Solubility of oxygen in zirconium
Beckman 2001 [DIRS 156122]	p. 103	Table 6-1, 6.2.11	Iodine induced SCC unlikely
Bradley et al. 1981 [DIRS 101564]	p. 38, Entire	Table 6-1, 6.2.3	Zircaloy-clad fuel rods before and after in-water spent fuel pool storage
Brossia et al. 2002 [DIRS 161988]	Entire, F3,	Table 6-1, 6.2.7, Appendix B	Pitting experiments, crevice corrosion
	F3	6.2.5, 6.2.6	Corrosion experiments, addition of H2O2
BSC 2001 [DIRS 155640]	Table 18	Table 6-1, 6.2.17	Composition of concentrated J13
BSC 2001 [DIRS 152059]	p. 36, Figure 6.1-2	Table 6-1, 6.2.15	Naval Fuels
BSC 2004 [DIRS 168030]	Entire	Table 6-1, 5.1	Seismic information
BSC 2004 [DIRS 169987]	6.2.1 & 6.2.2	6.2.23 & Table 6-1	CSNF testing, grain and gap inventory
BSC 2004 [DIRS 169868]	Entire	Table 6-1, 6.2.14	Melting temperature for glass
BSC 2004 [DIRS 169898]	Sections 6.2.8 through 6.2.10	Table 6-1, 6.2.18	EBS FEPs discussion
BSC 2004 [DIRS 167621]	Figure 6-54	Section 6.2.5	PH profiles
BSC 2004 [DIRS 169991]	Section 7.1	Table 6-1, Appendix B	Due to the environmental constraints, the microbial activity in YMP is expected to be low, and its impacts on drift chemistry can be insignificant
BSC 2004 [DIRS 168960]	Entire	Table 6-1, 6.2.21	Igneous intrusion on cladding
BSC 2004 [DIRS 168361]	Appendix A	Section 2, Table 6-1	List of Q-items
BSC 2004 [DIRS 167780]	Entire	Table 6-1, 6.2.14, 6.2.18	Information on Seismic events
BSC 2004 [DIRS 169183]	Section 6.8.2	Table 6-1, Appendix A	Discussion on seismic consequence of cladding
BSC 2004 [DIRS 171583]	Entire	Entire	Work Plan for FEPs documentation
BSC 2004 [DIRS 168706]	Entire	1.2, 4.1, 6.1, 6.1.1, 6.1.2, 6.1.6, Table 6-1	System FEPs
BSC 2004 [DIRS 170020]	Entire	1.3, Table 6-1	Addresses DSNF FEPs
Canori and Leitner 2003 [DIRS 166275]	Entire	4.2.1, Table 4-2, Table 6-1	Project requirements
Clayton 1989 [DIRS 149208]	Tables 1 through 5, p. 270	Table 6-1, 6.2.24	Hydriding of zircaloy
Cox 1973 [DIRS 152920]	Abstract	Table 6-1, 6.2.11	Passive layer of oxides on the zirconium surface
Cox 1990 [DIRS 152778]	p. 15, Figure 20	Table 6-1, 6.2.11	Threshold Intensity Factors, SCC

Table 6-1. Indirect Inputs (Continued)

Technical Product Input	Used From	Used In	Input Description
Cagnolino et al. 1999 [DIRS 152354]	p.4-13	6.2.20 & Table 6-1	Galvanic corrosion unlikely
	p. 4-15	6.2.11 & Table 6-1	Chloride-induced SCC
	p. 4-27	6.2.11 & Table 6-1	low threshold stress and failure
CRWMS M&O 1995 [DIRS 102829]	p. 15, 16	Table 6-1, 6.2.1	Standard fuel drying
CRWMS M&O 2000 [DIRS 136058]	Entire	Table 6-1, 6.2.5, 6.2.17	All corrosion mechanisms for Zircaloy cladding
	4.1.1	6.2.7	Corrosion of zirconium
	6.1.10	6.2.7	Crevice condition
	Entire	6.2.20, 6.2.21	Equilibrium-model calculations
CRWMS M&O 2000 [DIRS 151659]	Entire	1.3, Table 6-1, 6.2.6	Initial condition of the cladding
CRWMS M&O 2001 [DIRS 151662]	Entire, 6.2, Table 10B, 6.2.6	Table 6-1, 6.2.9, 6.2.11, 6.2.16	Cladding properties and creep performance
Debes 1999 [DIRS 161193]	Entire	Table 6-1, 6.2.18	No rod failures in transportation
DOE 1996 [DIRS 100320]	p. 123	Table 6-1, 6.2.14	Description of assembly
Einziger 1994 [DIRS 100442]	p. 556, eq. 14	Table 6-1, 6.2.3	Oxidation of UO_2 is slower in steam than in air
EPRI 2002 [DIRS 161421]	Entire	Table 6-1, 6.2.11	No failures from fuel assemblies and dry storage
	P. xii	6.2.9	Fuel assemblies and exposure
Farina et al. 2002 [DIRS 163639]	p.5	Table 6-1, 6.2.11	Passive layer prevents chloride-induced SCC
Garde 1991 [DIRS 101652]	Pg. 582	Table 6-1, 6.2.22	Fuel Temperatures
Garzarolli et al. 1979 [DIRS 149256]	p. 64, Entire	Table 6-1, 6.2.24	Oxygen affects on corrosion of zirconium, gas affects
Greene et al. 2000 [DIRS 145073]	F8, F11	6.2.5 & Table 6-1	Effect of adding H_2O_2
Güldner and Burtak 1999 [DIRS 171573]	Entire	Table 6-1, 6.2.3	Information on zircaloy corrosion resistance
Hansson 1984 [DIRS 101676]	Entire	Table 6-1, 6.2.8	Corrosion rates in cement pore solutions
IAEA 1998 [DIRS 150560]	Section 4.2.4	Table 6-1, 6.2.3	Oxidation of zirconium surface
	p.220, T8.4	6.2.5	Stability of H_2O_2
Jones 1992 [DIRS 169906]	Ch. 7	Table 6-1, 6.2.6	Pitting corrosion information
Knoll and Gilbert 1987 [DIRS 123682]	Entire	Table 6-1, 6.2.1	Cask drying
Kohli and Pasupathi 1986 [DIRS 131519]	p. iii, Entire	Table 6-1, 6.2.1	Water-logged spent fuel storage rods
Kreyns et al. 1996 [DIRS 100462]	p. 767, Figure 5, Entire	Table 6-1, 6.2.12	Hydrides could decrease the fracture toughness
Langmuir 1997 [DIRS 100051]	2.7.2	Table 6-1, 6.2.5	Half-life equation of radioactive decay

Table 6-1. Indirect Inputs (Continued)

Technical Product Input	Used From	Used In	Input Description
Lanning et al. 1997 [DIRS 101704]	V1, p. 3.2, Fig. 3.1)	Table 6-1, 6.2.22	Fuel center line temperatures
Lide 2002 [DIRS 160832]	Entire	Table 6-1, 6.2.1, 6.2.5	Properties of zirconium
Little and Wagner 1996 [DIRS 131533]	pp. 367 through 368, Entire	Table 6-1, 6.2.4	Microbiologically influenced corrosion
Manaktala 1993 [DIRS 101719]	Figure 3-4, pp. 3 to 12	Table 6-1, Appendix B	Helium pressure buildup
McNeil and Odom 1994 [DIRS 131537]	p.176	Table 6-1, 6.2.4	Sulfate-reducing bacteria do not affect zirconium
Merriam-Webster 1993 [DIRS 100468]	Entire	Entire	Definitions
NRC 2002 [DIRS 164593]	Entire, Appendix, p. 2	Table 6-1, 6.2.16, Appendix B	Temperature limit for Creep and Hydride reorientation
NRC 2003 [DIRS 163274]	2.2.1.2, 2.2.1.2.1.3, Entire	4.2.1, Table 4-2, 4.2.2.3, 6.1, 6.1.1, 6.1.2, Table 6-1, 7.1.1	Criteria
NRC 1997 [DIRS 101903]	Sec.8.V.1, Sec.4.V.4.b	6.2.16 & Table 6-1	Omission of water logged rods
NRC 2000 [DIRS 147797]	Entire	Table 6-1, 6.2.16	Little creep damage under 400°C
Peehs and Fleisch 1986 [DIRS 102065]	pp. 199 to 202, Entire	Table 6-1, 6.2.1	Water-logged spent fuel storage rods
Pescatore and Cowgill 1994 [DIRS 102066]	pp. 83 to 85, Entire	Table 6-1, 6.2.16	Comparison of creep models for Zircaloy.
Pourbaix 1974 [DIRS 100817]	p. 226, p. 312, Entire	Table 6-1, 6.2.3, 6.2.24	Stability of water and oxygen in zirconium. Ferric ion stability
	P. 583	6.2.17	Stability and solubility of HF
Puls 1988 [DIRS 102067]	pp. 1,507 thru 1,522, Entire	Table 6-1, 6.2.12	Series of strain tests on Zircaloy-2
Ralph et al. 2002 [DIRS 161992]	Entire, p.6	Table 6-1, 6.2.4	MIC and galvanic corrosion
Reed-Hill 1973 [DIRS 121838]	Pg. 800	Table 6-1, 6.2.11, 6.2.12	Crack intensity factor
Rothman 1984 [DIRS 100417]	Table 6	Table 6-1, 6.2.10, Appendix B	Helium pressurization
	Entire, p. 6-13	6.2.3	General corrosion of zirconium
	pp. 33-39	6.2.12	DHC not expected
Smith 1966 [DIRS 149107]	p. 325, Table 3, Entire	Table 6-1, 6.2.24	Hydrides, hydrogen absorption and oxygen dissolution in zirconium
Soderman and Jonsson 1996 [DIRS 149441]	Entire	Table 6-1, 6.2.20	Electrophoresis
Štefanic and LaVerne 2002 [DIRS 166303]	Abstract	Table 6-1, 6.2.5	Decay constants for H ₂ O ₂

Table 6-1. Indirect Inputs (Continued)

Technical Product Input	Used From	Used In	Input Description
Tasooji et al. 1984 [DIRS 102093]	Entire, F.12, p. 612	Table 6-1, 6.2.11	Iodine-induced SSC was not a major failure mechanism
Webster 1990 [DIRS 143561]	Table 3 p. 665	Table 6-1, 6.2.10	Minimum yield strength of Zircaloy-4
Wolery and Daveler 1992 [DIRS 100097]	Entire	Table 6-1, 6.2.5	Description of EQ6
Wolfram et al. 1996 [DIRS 165268]	Entire	Table 6-1, Appendix B	MIC & Zirconium
Yau 1983 [DIRS 149233]	Entire, p. 26/10, p. 26/2	Table 6-1, 6.2.4, 6.2.7, 6.2.8	Discusses pitting and corrosion in chloride solutions
Yau 1984 [DIRS 102050]	Entire	Table 6-1, 6.2.21	General, crevice, and SCC tests with zirconium
Yau and Maguire 1990 [DIRS 110761]	Entire	Table 6-1, 6.2.6	Fluorides and pitting
Yau and Webster 1987 [DIRS 100494]	Entire, Table 6, p. 709	Table 6-1, 6.2.3, 6.2.4, 6.2.5, 6.2.21 & Appendix B	Corrosion properties of zirconium

6.1.2 Features, Events, and Processes Screening

The second step of FEP analysis is screening, which addresses Acceptance Criterion 2 of *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3). The TSPA-LA FEP screening process is described in *The Development of the TSPA-LA Features, Events, and Processes* (BSC 2004 [DIRS 168706], Section 4).

For FEP screening, each FEP is filtered by specified exclusion criteria (Section 4.2.3), as summarized in the three following FEP screening statements:

1. FEPs having less than one chance in 10,000 of occurring over 10,000 years may be excluded (screened out) from the TSPA-LA on the basis of low probability (as per 10 CFR 63.114(d) [DIRS 156605]).
2. FEPs whose omission would not significantly change the magnitude and time of the resulting radiological exposures to the RMEI, or radionuclide releases to the accessible environment, may be excluded (screened out) from the TSPA-LA on the basis of low consequence (as per 10 CFR 63.114 (e and f) [DIRS 156605]).
3. FEPs that are inconsistent with the characteristics, concepts, and definitions specified in 10 CFR Part 63 [DIRS 156605] may be excluded (screened out) from the TSPA-LA by regulation.

A FEP need only satisfy one of the exclusion screening criteria to be excluded from TSPA-LA. A FEP that does not satisfy any of the exclusion screening criteria must be included (screened in) in the TSPA-LA model.

This report documents the screening decisions for the cladding FEPs. In cases where a FEP covers multiple technical areas and is shared with other FEP reports, this report provides a partial technical basis for the screening decision as it relates to waste form issues. Collectively, the shared FEP reports provide the complete technical basis.

Documentation of the screening for each FEP is provided in Section 6.2. The following standardized format is used:

Section 6.2.x FEP Name (FEP Number)

FEP Description: This field describes the nature and scope of the FEP under consideration.

Screening Decision: Identifies the screening decision as one of:

- “Included”
- “Excluded – Low Probability”
- “Excluded – Low Consequence”
- “Excluded – By Regulation.”

In a few cases, a FEP may be excluded by a combination of criteria (e.g., low probability and low consequence), but the final decision can never be listed as both included and excluded.

Screening Argument: This field is used only for excluded FEPs. It provides the discussion for why a FEP has been excluded from TSPA-LA.

TSPA Disposition: This field is used only for included FEPs. It provides the consolidated discussion of how a FEP has been included in TSPA-LA, making reference to more detailed documentation in other supporting technical reports, as applicable.

Supporting Reports: This field is only used for included FEPs. It provides the list of supporting technical reports that identified the FEP as an included FEP and contain information relevant to the implementation of the FEP within the TSPA-LA model. This list of supporting technical reports provides traceability of the FEP through the document hierarchy. For excluded FEPs, it is indicated as not applicable “N/A.”

6.1.3 Supporting Reports and Inputs

The direct inputs used for the screening arguments are identified in Table 4-1. Indirect inputs are listed in Table 6-1. These inputs are primarily used for reference only and do not have direct inputs.

6.1.4 Qualification of Unqualified Direct Inputs

Direct inputs are listed in Section 4.1. Any data that required qualification are identified in Appendix B.

6.1.5 Assumptions and Simplifications

For included FEPs, the TSPA-LA dispositions may include statements regarding assumptions made to implement the FEP within the TSPA-LA model. Such statements are descriptive of the manner in which the FEP has been included and are not used as the basis of the screening decision to include the FEP with the TSPA-LA model.

As the individual FEPs are specific in nature, any discussion of applicable mathematical formulations, equations, algorithms, numerical methods, or idealizations or simplifications are provided within the individual FEP discussions in Section 6.2.

6.1.6 Intended Use and Limitations

The intended use of this report is to provide FEP screening information for a project-specific FEP database and to promote traceability and transparency regarding FEP screening. This report is used as the source documentation for the FEP database described in *The Development of the TSPA-LA Features, Events, and Processes* (BSC 2004 [DIRS 168706]). For included FEPs, this document summarizes and consolidates the method of implementation of the FEP in TSPA-LA in the form of TSPA disposition statements, based on more detailed implementation information in the listed supporting technical reports. For excluded FEPs, this document provides the technical basis for exclusion in the form of screening arguments.

Inherent in this evaluation approach is the limitation that the repository will be constructed, operated, and closed according to the design used as the basis for the FEP screening and in

accordance with NRC license requirements. This is inherent in performance evaluation of any engineering project, and design verification and performance confirmation are required as part of the construction and operation processes. The results of the FEP screening presented herein are specific to the repository design evaluated in this report for TSPA-LA.

Any changes in direct inputs listed in Section 4.1, in baseline conditions used for this evaluation, or in other subsurface conditions, will need to be evaluated to determine if the changes are within the limits stated in the FEP evaluations. Engineering and design changes are subject to evaluation to determine if there are any adverse manner impacts to safety as codified at 10 CFR 63.73 Subparts F and G [DIRS 156605]. See also the requirements at 10 CFR 63.44 and 10 CFR 63.131 [DIRS 156605].

6.2 CLADDING FEPS SCREENING AND ANALYSES

Screening information for each of the 24 cladding degradation FEPs is presented in separate subsections addressed in numeric order based on the FEP number.

For many of the FEPs addressed in this report, it was determined that the probability of a consequential event/condition occurring during the 10,000-year period following permanent closure is extremely low. However, considering the limited availability of data, analyses of the potential impacts of such low probability events or conditions was performed in lieu of a quantitative probabilistic evaluation. These analyses resulted in a determination that even in the unlikely event that on a probability weighted basis the event or condition should occur, it would not have a significant consequential effect on the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide release to the environment. As such, these FEPs have been excluded due to low consequences as permitted by 10 CFR 63.114 (e and f) [DIRS 156605]. In many of the exclusion argument discussions provided below, the low probability weighting of the FEP provides additional support to the exclusion argument and the FEP is excluded on the basis of low consequence.

6.2.1 Degradation of Cladding from Waterlogged Rods (2.1.02.11.0A)

FEP Description:

Failed fuel rods (attributed to breaches caused by manufacturing defects and reactor operations) comprise a small fraction of the fuel rods that are currently being stored in commercial reactor spent fuel pools. Failed fuel contains water in the fuel rod void space that may promote degradation of the spent fuel cladding. 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 (low consequence)

Screening Argument:

Degradation of cladding and waste form from waterlogged rods is excluded from the TSPA-LA because of low consequence.

Waterlogged rods are already failed, and further cladding failures of intact rods and waste form degradation are not expected because of the small amount of water present. It is assumed that only a few rods will be breached at the time they arrive at the repository (Section 6.2.2; BSC 2004 [DIRS 170023], Section 6.2.1). The fraction of fuel rods with breached cladding is currently estimated to be 0.01 to 1 percent (Section 6.2.2; BSC 2004 [DIRS 170023], Section 6.2.1), or about 5.6 rods in an average PWR waste package. 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 [DIRS 101903], Section 8.V.1; Knoll and Gilbert 1986 [DIRS 123682], p. iii). Moisture is effectively removed from defective rods during the cask drying operations. *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997 [DIRS 101903], Section 8.V.1) describes the cask drying criteria with reference to Knoll and Gilbert 1987 [DIRS 123682]. The residual moisture of the cask atmosphere can be estimated based on the drying conditions where the cask or waste package is vacuum dried at 5 mbars (maximum water vapor pressure, equivalent to 7.7 g or 0.43 moles of water). This amount of water produces an insignificant potential for corrosion of the cladding during dry storage or during disposal (CRWMS M&O 1995 [DIRS 102829], p. 16).

A PWR waste package contains approximately 2,500 kg (2.74×10^4 moles) of zirconium alloys (Lide 2002 [DIRS 160832]). Less than 0.43 mole (7.7 g) of H₂O is expected to be present (Knoll and Gilbert 1987 [DIRS 123682]) and general corrosion of the zirconium alloys from the water would only degrade a small fraction of the zirconium (0.2 mole). Hydrogen uptake would be insignificant compared to existing hydrides on the cladding.

Kohli and Pasupathi (1986 [DIRS 131519], p. iii) discuss removal of water from waterlogged fuel rods. Two reactor-breached fuel rods were tested, along with two fuel rods that were intentionally damaged after irradiation. A predetermined amount of moisture was added to the intentionally damaged 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 1.8 m of the furnace was heated to 400°C; the remainder was heated to 200°C. 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 1 to 1.3 hours and all measurable releases ended after 3.7 hours. This set of experiments demonstrated that standard cask drying procedures would remove water from failed rods.

Peehs and Fleisch (1986 [DIRS 102065], pp. 199 to 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. Testing results showed 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.

An alternative conceptual model for waterlogged rods would be that the failed rods contain water, which is then released into the waste package to be consumed by the carbon steel rack inside the waste package. This alternative conceptual model is possible because the rate at which carbon steel racks react with water vapor to form rust is faster than that for zirconium and UO₂. The hydrogen that is produced tends not to be absorbed by the zirconium alloys

(Section 6.2.24). It would lead to no significant damage because of the limited amount of water and large volume of steel in a waste package.

The above results have shown that the amount of water in waterlogged rods is small and that the degradation of the cladding is minimal. These results support the conclusion noted above in CRWMS-M&O 1995 [DIRS 102829].

In conclusion, cladding and waste form degradation from waterlogged fuel rods is excluded from TSPA-LA. There are few failed rods in any waste package that could be waterlogged and the volume of water inside a rod is quite limited. Because the drying procedure is effective the quantity of water is limited. Cladding failure due to rod waterlogging has a low consequence and is excluded from further consideration. The magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, would not be significantly changed by the omission of this FEP (degradation from waterlogged rods) from the TSPA-LA model.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.2 Degradation of Cladding Prior to Disposal (2.1.02.12.0A)

FEP Description:

Certain aspects of cladding degradation may 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 (i.e., from creep and from vibration and impact) and fuel handling.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

Degradation of cladding prior to disposal is included in TSPA-LA cladding degradation abstraction. The failure rate from prior degradation is based on historical data on reactor operation. It also includes failure from wet pool storage and transportation (negligible), dry storage, and handling (including spent pool events). In the TSPA, this percentage of rods is available for radionuclide release through fast release and axial splitting when the waste package is breached. It is specified as a 0.01 to 1 percent log uniform distribution (0.1 percent median) (BSC 2004 [DIRS 170023], Section 6.2.1). The TSPA-LA abstraction models that all stainless steel cladding fails and places the stainless steel cladding into waste packages as it arrives at the repository. This results in 3.5 to 7 percent (uniform distribution) of the waste packages containing stainless-steel-clad fuel rods (BSC 2004 [DIRS 170023], Section 6.2.2). These waste packages contain 15 to 30 percent stainless-steel-clad fuel rods that are failed and available for

fast release and unzipping (axial splitting) upon waste package failure (BSC 2004 [DIRS 170023], Section 8.1.2). Stainless steel cladding is modeled for TSPA-LA as failed, therefore, this FEP does not apply to stainless steel cladding.

Supporting Reports:

BSC 2004 [DIRS 170023]

6.2.3 General Corrosion of Cladding (2.1.02.13.0A)

FEP Description:

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

Screening Decision:

Excluded (low consequence)

Screening Argument:

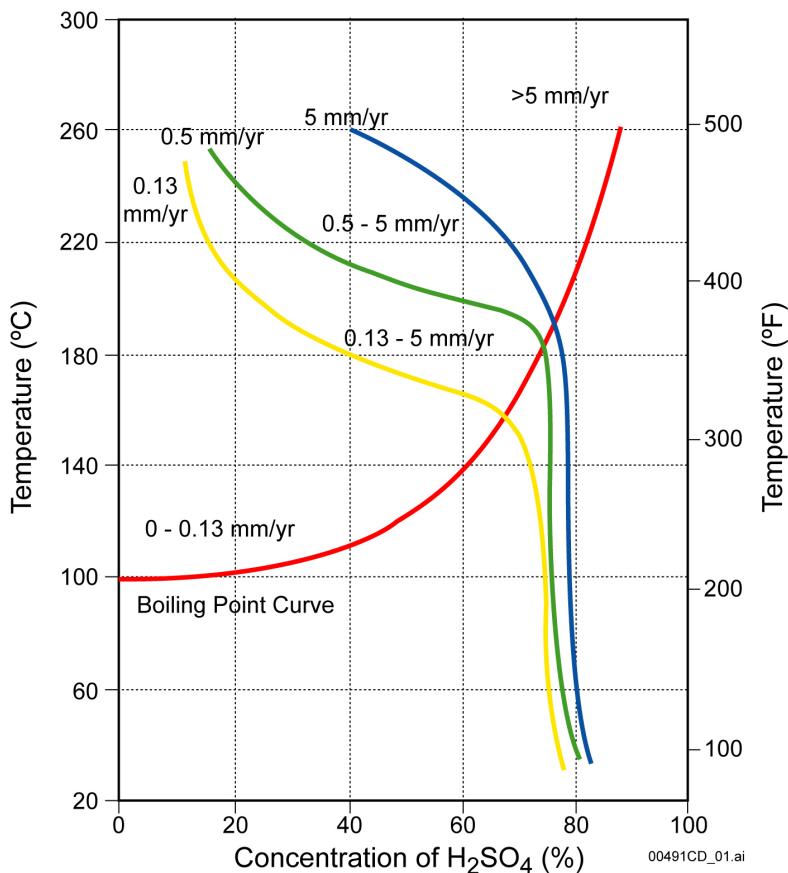
General corrosion of cladding has been excluded from the TSPA-LA on the basis of low consequence.

The in-package chemistry model (BSC 2004 [DIRS 167621]), which includes equilibrium solubilities of phases, solution complexes, and surface adsorption phenomena, estimates a minimum pH of 4.5 (lower green dashed line in Figure 6-2) for the solution in contact with the cladding. Yau and Webster (1987 [DIR 100494], pp. 709 to 710, Figs. 5, 7, and Table 6) review the corrosion potential for zirconium alloys in sulfuric acid. They note that zirconium alloys resist attack from H_2SO_4 at all concentrations up to 70 percent and at temperatures to boiling (their Figure 5 is reproduced in this report as Figure 6-1). A concentration of 70 percent H_2SO_4 represents a theoretical pH of about -1.15, well below anything expected in the waste package. In the range that zirconium alloys show corrosion resistance in H_2SO_4 , a protective film is formed on the zirconium that is predominantly cubic zirconium oxide (ZrO_2) with only traces of monoclinic phases. At concentrations higher than 70 percent, zirconium corrodes because loose films form that are zirconium disulfate tetrahydrate and partially zirconium hydrides. In concentrations of less than 60 percent H_2SO_4 , zirconium can tolerate some amounts of strong oxidizing agents such as 200 ppm Fe^{3+} and 200 ppm NO_3^- (Yau and Webster (1987 [DIRS 100494], Fig. 7). Moreover, in 20 percent or less H_2SO_4 ($pH > -0.61$), zirconium can tolerate a great amount of strong oxidizing agents. Consequently, zirconium equipment is often used in steel pickling. Zirconium alloys are used in the chemical industry under low pH conditions. In the manufacturing of H_2O_2 , zirconium alloys are used to contain up to 65 percent H_2SO_4 at up to 150°C. In the manufacturing of HNO_3 zirconium alloys are used to contain the acid up to 65 percent concentrations and temperatures to 204°C.

At low pHs (below -0.6), the ZrO_2 film will start to slowly dissolve. This is discussed in *Pitting Model for Zirconium-Alloyed Cladding at YMP* (BSC 2004 [DIRS 170043]), where there is discontinuity in the corrosion potential at about pH = -0.6; below this pH, the corrosion potential starts to rapidly decrease. This change in behavior is attributed to dissolution of the general oxide surface. Since the in-package chemistry model predicts pHs above -0.6, accelerated general corrosion is excluded from TSPA-LA performance models. The Pourbaix diagram for zirconium (Pourbaix 1974 [DIRS 100817], p. 226) shows that ZrO_2 could start to dissolve at a

pH of 4. In the text describing the diagram, the author (Pourbaix 1974 [DIRS 100817], pp. 228 and 229) recognizes that the solubility is much too high (1.2×10^{-3} moles/liter) and not consistent with experimental observations of solubility (10^{-7} moles/liter). The author attributes these high solubilities to not having modeled the dominant form of oxide (such as cubic ZrO_2), which controls the dissolution.

General corrosion is synonymous with zirconium oxidation for this repository application. The outer surface of the cladding becomes oxidized with a ZrO_2 film, which adheres to the surface and slows down further oxidation (IAEA 1998 [DIRS 150560], Section 4.2.4). The oxidation could be from O_2 consumption (dry oxidation) or H_2O consumption (wet oxidation). For the fuel in the repository, this corrosion does not occur until the waste package is penetrated.



Source: Yau and Webster 1987 [DIR 100494], Figure 5.

Figure 6-1. The Iso-Corrosion Diagram for Zirconium in Sulfuric Acid (H_2SO_4)

There are three possible effects of surface oxidation:

1. The oxidation could thin the cladding, contributing to cladding failure by creep rupture (Section 6.2.9). Wet oxidation generates hydrogen, and some of the hydrogen is absorbed into the cladding to form hydrides.
2. This hydrogen pick up could lead to delayed hydride cracking (DHC), or general hydride embrittlement (Section 6.2.12).

3. In the extreme, the oxidation could lead to cladding failure and expose the fuel pellets to the waste package environment.

The in-package chemistry model predicts that the pH will remain above 4.5. Under these nominal chemical conditions in the repository, general corrosion failures of the cladding are unlikely. *Waterside Corrosion of Zirconium Alloys in Nuclear Power Plants* (IAEA 1998 [DIRS 150560]) summarizes much of the research on zirconium corrosion. Hillner et al. (1998 [DIRS 100455], p. 9) studied corrosion of Zircaloy and published a Zircaloy corrosion correlation based on Bettis Atomic Power Laboratory experiments. Bettis Atomic Laboratory developed Zircaloy for naval reactors in the early 1950s and has an extensive database on Zircaloy performance, including 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 based on Hillner et al. calculations. The experiments are consistent with diffusion of oxygen ions through the corrosion film being the rate-limiting phenomenon. This corrosion film is generated in layers, with the physical characteristics of lower layer staying consistent (uniform). The consistency of the lower 2 μm of oxide film leads to a steady corrosion rate after a transition period.

Hillner et al. (1998 [DIRS 100455], 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. Einziger (1994 [DIRS 100442], p. 556, Equation 14) states that dry oxidation of zirconium is slightly slower than the wet corrosion rate.

There are experimental corroborating observations about the slow corrosion rates of zirconium alloys near ambient temperature (27°C). Bradley et al. (1981 [DIRS 101564], p. 38) performed metallurgical examinations of Zircaloy-clad fuel rods from two assemblies (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 [DIRS 101564], 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 assembly 0551 fuel rods in 1960) stored in water for over 20 years were unblemished and showed no evidence of reaction with water.

Rothman (1984 [DIRS 100417], pp. 6 to 13, Table 3) discusses cladding oxidation in repository conditions in great detail and 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 as the repository cools after a few hundred years), which would not lead to rod failure.

The cladding corrosion analyses (Hillner et al. 1998 [DIRS 100455], Rothman 1984 [DIRS 100417]) are based on the performance of Zircaloy-4 (Zirc-4), the cladding used in most PWRs before the year 2000. Since then, Zircaloy variants with increased corrosion resistance

(i.e., zirconium-niobium alloys) have been available for reactor reloads. The improved corrosion resistance of these cladding materials yields lower corrosion thickness compared to Zircaloy-4, even though the burnup has significantly increased over this time (Guldner and Burtak 1999 [DIRS 171573]). The hydride content of the cladding is directly proportional to the oxide thickness. Therefore, the new cladding is less likely to fail from general corrosion or hydride embrittlement (Section 6.2.12). The thinner oxide thickness produces lower fuel pellet temperatures and, thus, lowers fission gas release fractions. The remaining metal thickness is also greater. Therefore, cladding stresses are lower in advanced alloys and less creep is expected (Section 6.2.9).

Stainless steel cladding is modeled for TSPA-LA as failed, therefore, this FEP does not apply to stainless steel cladding.

Cladding degradation from general corrosion is excluded from TSPA-LA. The small amount of corrosion that will occur during the first 10,000 years following permanent closure of the repository will not penetrate the cladding and therefore will not affect the release of radionuclides. Cladding failure due to general corrosion has a low consequence and is excluded from further consideration. The magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, would not be significantly changed by the omission of this FEP (general corrosion of cladding) from the performance assessment (TSPA-LA) model.

TSPA Disposition:

N/A

Support Reports:

N/A

6.2.4 Microbially Influenced Corrosion (MIC) of Cladding (2.1.02.14.0A)

FEP Description:

Microbially influenced corrosion (MIC) of cladding is a potential localized corrosion mechanism where microbes produce a local acidic environment that could produce multiple penetrations through the fuel cladding.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Microbially influenced corrosion (MIC) activity is excluded as a cladding failure mechanism because of low consequence. Microbes are expected to be present in the repository, but MIC of cladding is not expected to cause cladding failure and, therefore, not have a significant effect on radionuclide exposures to the RMEI.

The term microbially influenced corrosion (MIC) is commonly used to designate corrosion caused by the presence and activities of microorganisms at the surfaces of metals.

Two studies of MIC on spent nuclear fuel have been performed. Wolfram et al. (1996 [DIRS 165268], pp. iii and iv) measured microbial activity in spent fuel pools. They concluded that all spent fuel pools tested contained microbial colonies. They also performed a literature search and concluded that “There was no evidence found in the literature that zirconium or its alloys are susceptible to MIC.”

Hillner et al. (1998 [DIRS 100455], p. 11) studied the corrosion of Zircaloy-clad fuels under repository conditions. They indicate that there are two major forms of MIC for materials being considered for waste packages. They 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 attack by SRB, Hillner et al. (1998 [DIRS 100455]) reference the work of McNeil and Odom (1994 [DIRS 131537], p. 176), which indicates by thermodynamic calculations that SRB do not affect zirconium alloys. With respect to corrosion induced by organic acids, Hillner et al. (1998 [DIRS 100455], p. 11) noted that it is most unlikely because of zirconium’s tolerance of a wide range of pHs and it is unlikely that production of weak organic acids will have an adverse effect on the passivation of Zircaloy by a ZrO₂ film. Yau and Webster (1987 [DIRS 100494], p. 717) also note that zirconium alloy 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. This supports the concept that organic solutions produced by MIC are unlikely to cause significant acceleration of the corrosion of zirconium alloys. *Evaluation of Potential Impacts of Microbial Activities on Drift Chemistry* (BSC 2004 [DIRS 169991], Section 7.1) evaluated in-drift microbial communities and concluded that the estimates of microbial masses growing in the potential repository system suggest that the effect to the in-drift geochemistry should be small.

Little and Wagner (1996 [DIRS 131533], p. 367) corroborate this information in an overview of MIC 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, under-deposit corrosion, and enhanced erosion and corrosion. Little and Wagner (1996 [DIRS 131533], pp. 367 and 368) describe several mechanisms for MIC. In addition, various case studies are presented that document MIC of alloys of iron, nickel, and copper. However, it should be noted that no indication has been found in the literature of MIC occurring on zirconium metal or alloys. Yau and Webster (1987 [DIRS 100494], p. 709) report no corrosion of zirconium alloy from marine organisms was found during seawater corrosion tests for 129 days.

MIC is excluded as a component of the localized (pitting) corrosion model where MIC could cause a localized suppression of the water pH and permit other aggressive species to attack cladding. The zirconium alloy pitting model (BSC 2004 [DIRS 170043], Section 8) shows that pitting is dependent on the concentration of chlorides, ferric ions, and hydrogen peroxide (from radiolysis).

One U.S. commercial nuclear plant spent fuel pool experienced a significant MIC event that lasted for about four years (Ralph et al. 2002 [DIRS 161992]). After an extended lay-up period, the spent fuel pool water was found to contain a significant amount of algae and bacteria. Biological agents were purged using controlled additions of chlorine and hydrogen peroxide before the pool was returned to normal operating chemistries.

The assessment revealed that the steel rack corrosion products were up to 2.5 cm thick, and they had started to engulf the individual fuel rods or flow channels of the stored assemblies in the region where the rack and plates contacted the fuel assemblies. The corrosion product had adhered to the fuel. The iron oxide was composed of FeO, Fe₂O₃, and Fe₃O₄. Ralph et al. (2002 [DIRS 161992], p. 6) state:

One fuel assembly was removed from its storage location and its channel was removed. The oxide from contact with the carbon steel rack was removed with a water lance utilizing 350 to 700 kg/cm² of water pressure. A camera with resolution of 0.025 mm was used to inspect the channel. The channel surface appeared uniform and smooth. No pitting, white discoloration or surface anomalies were observed.

The paper concluded: “The fuel cladding was not affected through any type of corrosion. Therefore, the corrosion did not change the classification of the fuel as intact or damaged.” As with the experiments by Yau (1983 [DIRS 149233]), the lack of pitting or stress corrosion cracking implies that the corrosion potential (E_{corr}) was not elevated to exceed the repassivation potential (E_{rp}) and thus cause corrosion, even with Fe₃O₄ present and adhering to the zirconium oxide film. MIC colonies could also have locally suppressed the pH but, again, no localized corrosion was observed.

Adler Flitton et al. (2002 [DIRS 161991], p. 4) buried various metal samples in an arid vadose zone environment for three years. They reported indications of pitting from MIC on some of the metals, but observed no pitting on the zirconium alloy samples.

Stainless steel cladding is modeled for TSPA-LA as failed, therefore, this FEP does not apply to stainless steel cladding.

In summary, microbes are expected to be present in the repository, but MIC of cladding is not expected to cause cladding failure. The NRC requirements in 10 CFR 63.114 (e and f) allow the exclusion of MIC from the TSPA-LA because the omission would not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.5 Localized (Radiolysis Enhanced) Corrosion of Cladding (2.1.02.15.0A)

FEP Description:

Radiolysis in a nitrogen/oxygen gas mixture with the presence of water film results in the formation of nitric acid (HNO₃). Hydrogen peroxide (H₂O₂) is formed in the water from radiolysis. These chemicals can enhance corrosion of the fuel cladding.

Screening Decision:

Excluded (low consequence)

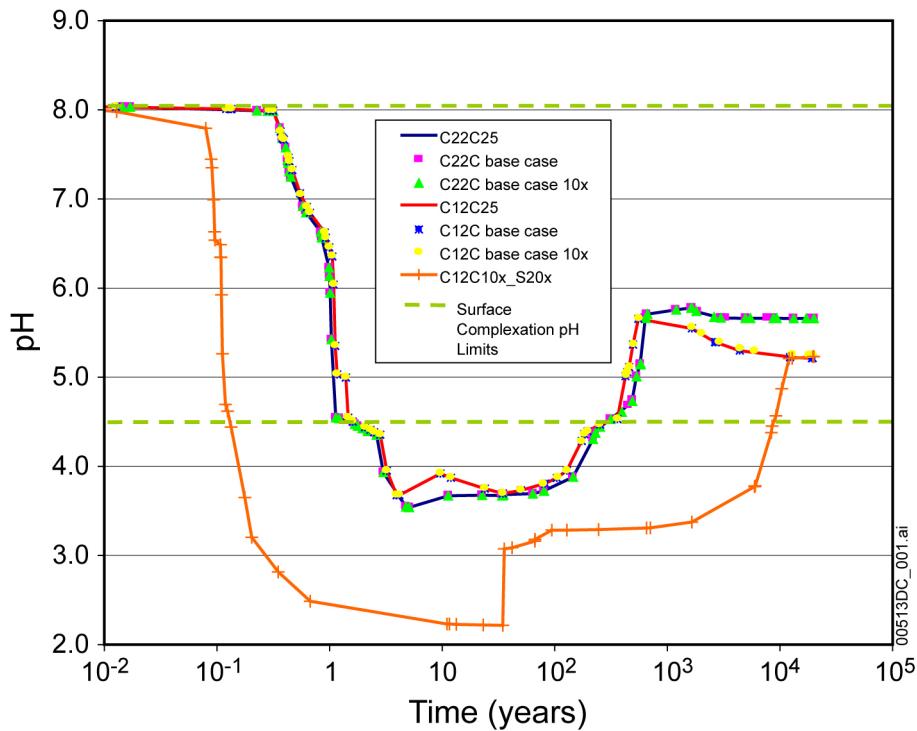
Screening Argument:

Radiolysis, as a cladding failure mechanism, is excluded from the TSPA-LA on the basis of low consequence.

The in-package chemistry model (BSC 2004 [DIRS 167621], Appendix B) addressed the change of water chemistry with the inclusion of nitric acid and hydrogen peroxide production from radiolysis. In the analysis, all of the nitric acid that can be produced in a moist waste package was absorbed into the water film on the cladding surface. The radiation field was modeled as being constant at the radiation levels expected at 500 years following repository closure, although the level would, in fact, decrease with the passing of time.

The effect of radiolysis on the in-package chemistry was handled in a nonmechanistic manner (i.e., EQ6 does not have the facility to model the radiolysis process directly), therefore, only the products of radiolysis were included as inputs in EQ6 simulations. EQ6 is a reaction path code that models water, rock or other material interactions or fluid mixing in either a pure reaction progress mode or a time mode (Wolery and Davele 1992 [DIRS 100097]). A series of runs was performed where nitric acid and hydrogen peroxide were included as inputs in EQ6 runs and the results of these simulations were compared to their nonradiolysis counterparts. Two base-case runs (C12C25 and C22C25; BSC 2004 [DIRS 167621], Section 6.3) were used to test the effects of radiolysis. These runs were chosen because they represent the median fuel exposure value and the low and median water flux values. It would be expected that the effects of nitric acid and hydrogen peroxide additions would be greater at the low end of the flux range. Two simulations were performed for each file, the first using the base-case nitric acid and hydrogen peroxide rates and the second multiplying the HNO_3 and H_2O_2 rates by a factor of ten. In addition, one simulation (C12C10x_S20x) was conducted with twenty times the sulfur in carbon steel.

The results of the simulations are displayed in Figure 6-2, which includes the results of the surface complexation pH limits, where the pH profiles for the various runs are displayed versus time. These results show that neither the base case nor the 10 \times base-case generation rates of HNO_3 and H_2O_2 had an impact on the in-package pH. Therefore, it may be concluded that if radiolysis only affects the chemistry via HNO_3 and H_2O_2 generation, then it will not be a significant process with regard to influencing the in-package chemistry. The radiolysis did not significantly affect the concentrations of Cl^- , Fe^{3+} , or H_2O_2 and therefore did not change the corrosion potential of the passive film on the zirconium alloy. The Naval Nuclear Propulsion Program cladding and SNF performance is represented by Figure 6-54 in *In-Package Chemistry Abstraction* (BSC 2004 [DIRS 167621]), and is not indicative of the surface complexation pH profiles shown on Figure 6-2.



Source: BSC 2004 [DIRS 167621], Figure B-1.

Figure 6-2. pH Profiles Showing Radiolysis Impacts on pH

Table 6-2 provides the chloride, ferric iron, and hydrogen peroxide maximum concentrations for the simulations. Again, runs with HNO_3 and H_2O_2 input show little deviation compared to the runs without.

Table 6-2. Chloride, Ferric Iron, and Hydrogen Peroxide Molality

EQ6 Input File	Maximum Molalities (m)		
	Cl^-	Fe^{+++}	$\text{H}_2\text{O}_2(\text{aq})$
C22C25	6.8E-04	5.1E-11	3.7E-19
C22CBC	6.6E-04	8.5E-11	3.7E-19
C22BC10x	6.5E-04	8.5E-11	3.7E-19
C12C25	9.7E-04	5.3E-11	3.7E-19
C12CBC	9.6E-04	5.3E-11	3.7E-19
C12C10x_S20x	9.7E-04	3.8E-06	3.7E-19

Source: BSC 2004 [DIRS 167621], Table B-2.

The Center for Nuclear Waste Regulatory Analysis (CNWRA) performed a series of corrosion tests where hydrogen peroxide was added to the ongoing test while the corrosion potential was being measured. Greene et al. (2000 [DIRS 145073], Figure 8) shows two experiments where H_2O_2 was added and the corrosion potential was measured. In one test, H_2O_2 was added two different times. In the three cases where H_2O_2 was added, the effect of the hydrogen peroxide rapidly died out. In another test (Greene et al. 2000 [DIRS 145073], Figure 11), a sample that was oxidized in air at 200°C was exposed to a solution of 1M NaCl. When 5 mM H_2O_2 was added, the corrosion potential increased by 0.275 V_{SCE} (volts, saturated calomel electrode scale) and pitting was observed. In this experiment the corrosion potential normally is nominally -0.07

V_{SCE} , and the repassivation potential is 0.04 V_{SCE} , so the increase in corrosion potential is significant. The concentrations of chloride and hydrogen peroxide in this experiment are many orders of magnitude higher than expected in the waste package (see concentrations in Table 6-2). Therefore, the corrosion potential in the repository should not increase in the same proportion as these experiments.

Brossia et al. (2002 [DIRS 161988], Figure 3) report two experiments where H_2O_2 was added to ongoing corrosion potential tests. The metal samples had oxide coatings of 1.7 μm and 3.4 μm thick. The initial solution contained 0.1 M NaCl at 95°C and 5 mM H_2O_2 was added. In both tests the corrosion potential initially increased, but later one test showed decreasing corrosion potentials. Pitting was not observed in either experiment. Again, these concentrations are higher than expected in the in-package chemistry.

Greene et al. (2000 [DIRS 145073], Figure 8) discuss the stability of H_2O_2 . International Atomic Energy Agency suggests the that H_2O_2 is known to decompose catalytically on the surfaces of various types of materials at ambient temperatures, and that zirconium oxides enhance decomposition (IAEA 1998 [DIRS 150560], p. 220). H_2O_2 decomposes in bulk solutions at elevated temperatures with a decay rate, $k = 6.5 \times 10^5 \times \exp(-71 \text{ kJ/mol}/RT)$, in units of inverse seconds (Štefanic and LaVerne 2002 [DIRS 166303], Abstract). Table 6-3 provides the half-life of H_2O_2 at various temperatures, calculated from $t_{1/2} = 0.693/k$ (Langmuir 1997 [DIRS 100051], Section 2.7.2) and $R = 0.008314 \text{ kJ/mol}\cdot\text{K}$ (Lide 2002 [DIRS 160832]), and shows that bulk decomposition during the regulatory period will be significant.

The in-package model, coupled with the pitting experiments and pitting model (Section 6.2.6 and Table 6-4), shows that radiolytic production of nitric acid and hydrogen peroxide is not sufficient to influence the corrosion potentials significantly at the steady-state concentrations expected in the repository and to produce pitting. Zirconium alloys have been shown to be relatively inert in nitric acid and hydrogen peroxide as discussed by Yau and Webster (1987 [DIRS 100494]) and in *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (CRWMS M&O 2000 [DIRS 136058], Section 6.1.6 and Appendix II.4). For example, the chemical processing industry uses peroxide strengths of 90 percent with zirconium equipment. The service life has been increased by an order of magnitude compared to graphite components previously used, which were generally considered to be inert. In nitric acid, zirconium and its alloys are inert up to acid concentrations of 65 weight percent. Since radiolysis does not produce nitric acid and hydrogen peroxide at greater concentrations, there will be no impact (low consequence) on the uniform corrosion rate as a result of radiolysis.

Table 6-3. Half-Life of Hydrogen Peroxide at Various Temperatures

Temperature (°C)	Half-Life (Days)
25	34
50	3.79
73	0.65
95	0.15
100	0.11

Source: Štefanic and LaVerne 2002 [DIRS 166303], Abstract.

Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding. Naval Nuclear Propulsion Program cladding and SNF performance is discussed in *Naval Nuclear Propulsion Program Technical Support Document for the License Application*, which is a classified document.

Cladding degradation from radiolysis-enhanced corrosion is excluded from TSPA-LA. Radiolytic production of nitric acid and hydrogen peroxide was included in the in-package chemistry model (BSC 2004 [DIRS 167621], Appendix B), which showed that radiolysis had a small effect on the chemistry. Experiments where hydrogen peroxide was added to tests show that in many cases the effect of the hydrogen peroxide quickly becomes negligible. Radiolysis by itself is not expected to damage the cladding (low consequence). Cladding failure due to radiolysis-enhanced corrosion has a low consequence, and is excluded from further consideration. The magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, would not be significantly changed by the omission of this FEP (radiolysis enhanced corrosion) from the performance assessment (TSPA-LA) model.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.6 Localized (Pitting) Corrosion of Cladding (2.1.02.16.0A)

FEP Description:

Localized corrosion in pits could produce penetrations of cladding.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Localized corrosion of the cladding from pitting has been excluded on the basis of low consequence. A comparison of the expected in-package chemistry to the chemical composition where pitting is observed shows that pitting is not expected. Therefore, the inclusion of a pitting model in TSPA-LA would not have a significant effect on the magnitude and time of radiological exposures or radionuclide releases (low consequence).

A zirconium-pitting model (BSC 2004 [DIRS 170043]) was developed to investigate the chemical conditions at which pitting occurs. Zirconium alloys are susceptible to pitting in a particularly aggressive combination of chloride (Cl^-) ions, ferric ions (Fe^{+3}), or hydrogen peroxide (H_2O_2). In order to predict cladding failure from chloride pitting, a review of the literature for pitting rates and electrochemical data for various zirconium alloys was conducted. Based on this review of the literature, failure criteria were constructed based on an electrochemical definition of pitting as the condition at which the corrosion potential for as-polished metal exceeds repassivation potential (i.e., $E_{\text{corr}} > E_{\text{rp}}$). Corrosion potential and repassivation potential values were obtained for as-polished zirconium alloys in various solution concentrations of Cl^- , Fe^{+3} , and H_2O_2 using measurements obtained from various experiments.

The model to predict repassivation potential depends only on chloride concentration in the solution. The corrosion potential for as-polished metal (E_{corr}) was modeled by performing a regression analysis to fit experimental data with varying molar concentrations of Cl^- , Fe^{+3} , and H_2O_2 . The model describes the conditions where pitting was observed in experiments. High concentrations of chlorides at extremely low pH (below -0.6) can lead to the general dissolution of the protective zirconium oxide film.

This model was evaluated using in-package chemistry, including the production of nitric acid and hydrogen peroxide from radiolysis (Section 6.2.5 and Table 6-2). Table 6-4 provides the corrosion potential for as-polished metals, repassivation potential, and potential differences for the cases described in Table 6-2. Table 6-4 shows that pitting is not expected because the repassivation potential exceeds the corrosion potential for as-polished metal.

Table 6-4. Corrosion and Repassivation Potentials for In-Package Chemistry Conditions

Case #	E_{corr} V_{SCE}	E_{rp} V_{SCE}	Difference ^a ($E_{rp}-E_{corr}$) V_{SCE}
C22C25	-0.22	0.30	0.52
C22CBC	-0.22	0.30	0.52
C22BC10x	-0.22	0.30	0.52
C12C25	-0.21	0.29	0.50
C12CBC	-0.21	0.29	0.50
C12BC10x	-0.21	0.29	0.50

Source: BSC 2004 [DIRS 170043], Table 8-1.

NOTE: ^a Negative difference implies pitting is possible.

The pitting model was generated with data for as-polished and atmospherically oxidized samples, but CSNF fuel rods are coated with thick oxides (54 μm mean, 5 to 95 percent μm range = 112 μm to 5.3 μm ; CRWMS M&O 2000 [DIRS 151659], Section 6.4). These oxides will affect the measured corrosion potential (open circuit potential) because of the high electrical resistance of the coating, but will not increase susceptibility to pitting corrosion. This sensitivity to oxide thickness was demonstrated in a series of four tests performed on zirconium samples at the Center for Nuclear Waste Regulatory Analyses. Two tests had oxide thickness of 1.7 μm and two tests had 3.4 μm coatings (Brossia et al. 2002 [DIRS 161988], Figure 3). These samples were exposed to 0.1M of NaCl. Prior to addition of oxidants to the solution, the highest corrosion potential was 0.67 V_{SCE} , well above the repassivation potential of 0.12 V_{SCE} . Two other samples also had potentials above the repassivation potential. No pitting was observed in the samples. Yau and Maguire (1990 [DIRS 110761], Fig. 4) showed that zirconium annealed in air (air oxidized) also had a corrosion potential that was above the as-polished value and the repassivation potential. Again, no pitting was observed. These observations show that the corrosion potential for oxide-coated material can exceed the repassivation potential and pitting will not occur. $E_{corr} > E_{rp}$ is a necessary but not sufficient condition for pitting. For the case where the corrosion potential is raised above the repassivation potential only due to oxide formation, pitting will not occur, as discussed below.

When pitting occurs on metal surfaces, the pit behaves as an anode and the surrounding metal surface behaves as a cathode. The metal in the pit gives up electrons, becomes oxidized, and

goes into solution (Jones 1992 [DIRS 169906], Chapter 7). To support pitting, the electrons must be conducted to the free surface of the surrounding metal and be accepted by an oxidizing agent (e.g., H_2O_2 , Fe^{3+} , or O_2). The potential for pitting starts when some event creates a site with less protection than the surrounding metal. For pitting to then occur, three conditions must be satisfied:

1. The surface of the surrounding metal must be able to transport the electrons into the solution
2. Oxidizing agents in the solution must be present to accept the electrons
3. The pit site must support transport of the metal ions into solution.

Zirconium oxide is a poor conductor and a thick oxide coating restricts the current of electrons from the metal into the solution (Condition 1 above). This insulation results in the measured corrosion potential increasing from the Zircaloy corrosion potential toward the oxide-surface redox-reaction potential as the oxide insulation increases. The redox potential is not relevant to pitting susceptibility but is a consequence of inerting of the Zircaloy metal. If the oxide has a defect, the metal corrosion potential decreases to that of the bare metal, and pitting occurs only if oxidizing species exist in solution or an applied potential force the bare-metal Zircaloy corrosion potential above the repassivation potential. If there are insufficient oxidizing agents in the solution to support a critical current (Condition 2 above), then a new protective oxide surface will form on the potential pit surface and prevent dissolution of the metal (Condition 3 above).

Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding.

Although the corrosion potential (E_{corr}) is greater than the repassivation potential (E_{rp}) for zirconium with thick oxide layers, both empirical evidence and understanding of the response to local breakdown of the protective oxide layer (e.g., due to oxide layer cracking) indicate that pitting does not occur. The empirical evidence discussed in the above paragraph (Brossia et al. 2002 [DIRS 161988] Figure 3; Yau and Maguire 1990 [DIRS 110761], Fig. 4) showed that zirconium with thick oxide layers did not pit. The oxide layer on the zirconium surrounding the breakdown region prevents sufficient cathodic current to support pitting. Therefore, the anodic and cathodic reactions supporting corrosion occur in the breakdown region, a condition that cannot occur for pitting. In addition, upon breakdown of the oxide, the corrosion potential drops to about the corrosion potential of the as-polished zirconium. In effect, the breakdown region will not be influenced by the surrounding oxidized zirconium surface and will behave as as-polished zirconium. Therefore, results obtained using as-polished zirconium coupons apply in assessing the corrosion behavior in the breakdown region. For the experiments described by Brossia et al. (2002 [DIRS 161988], Figure 3), the corrosion potential for an as-polished surface (-0.11 V_{SCE}) is significantly less than the repassivation potential (0.12 V_{SCE}), and pitting was neither observed nor expected to occur.

In conclusion, cladding degradation from localized (pitting) corrosion of the cladding is excluded from TSPA-LA. A comparison of the expected in-package chemistry to the chemical composition where pitting is observed shows that pitting is not expected. Cladding failure due to

pitting has a low consequence, and is excluded from further consideration. The magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, would not be significantly changed by the omission of this FEP (localized (pitting) corrosion of the cladding) from the performance assessment (TSPA-LA) model.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.7 Localized (Crevice) Corrosion of Cladding (2.1.02.17.0A)

FEP Description:

Localized corrosion in crevices could produce penetrations of cladding.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Localized (crevice) corrosion of the cladding is excluded from the TSPA-LA on the basis of low consequence.

Yau and Webster (1987 [DIRS 100494], p. 717) report: "Of all the corrosion-resistant structural metals, zirconium and tantalum are the most resistant to crevice corrosion. In low-pH chloride solutions or chlorine gas, for example, zirconium is not subject to crevice attack." Greene et al. (2000 [DIRS 145073]) and Brossia et al. (2002 [DIRS 161988]) performed pitting and crevice corrosion tests on Zircaloy-4. They covered temperatures from 25°C to 95°C, chloride concentrations from 0.001 to 4.0 M, and pH from 2.1 to 10.7. The solutions also contained the predominant anions in the groundwater. Some of their tests had sufficiently aggressive solutions to cause pitting on exposed surfaces. Other tests had voltages applied to the sample to raise the corrosion potential above the repassivation potential and cause pitting on exposed surfaces. They report that no crevice corrosion is observed under the same environment and electrochemical conditions that promote pitting corrosion on exposed surfaces. In summary, crevice corrosion is not observed under severe conditions that promote pitting on the exposed surfaces.

Additional corroborating information is also available. More detailed information is provided by Yau (1983 [DIRS 149233]) showing that zirconium and Zr-1.5 percent Sn were resistant to crevice corrosion after 14 days exposed to boiling (107°C), saturated NaCl solution with the pH adjusted to 0 by the addition of HCl. *Clad Degradation-Local Corrosion of Zirconium and its Alloys Under Repository Conditions* (CRWMS M&O 2000 [DIRS 136058]) summarizes these experiments and shows that zirconium is not susceptible to crevice corrosion. Section 4.1.3 of that report (CRWMS M&O 2000 [DIRS 136058]) discusses the crevice corrosion resistance of zirconium in various chemical solutions, summarizes seven crevice corrosion tests, and reports that crevice corrosion was not observed. The U-bend tests discussed in Section 4.1.4 of *Clad*

Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions (CRWMS M&O 2000 [DIRS 136058]) are also designed to produce crevice corrosion under the U-bend test washers. In these tests, no crevice corrosion was reported. Section 6.1.10 of *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (CRWMS M&O 2000 [DIRS 136058]) discusses the theoretical reasons why zirconium is immune to this type of corrosion.

Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding.

In conclusion, cladding degradation from localized (crevice) corrosion is excluded from TSPA LA. Crevice corrosion of zirconium under repository in-package chemistry conditions is not expected. NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow the omission because crevice corrosion is not expected under repository conditions and therefore will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.8 Enhanced Corrosion of Cladding from Dissolved Silica (2.1.02.18.0A)

FEP Description:

High dissolved silica content of waters may enhance corrosion of cladding.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Enhanced corrosion of cladding due to high dissolved silica content in waters is excluded from the TSPA-LA on the basis of low consequence.

The potential for silica itself degrading the cladding is negligible. Hansson (1984 [DIRS 101676]) reports corrosion tests with concrete pore fluids that normally contain silica. Yau (1983 [DIRS 149233]) reports corrosion tests in seawater that also contains silica. Neither experimenter reports significant corrosion. Yau and Webster (1987 [DIRS 100494], Table 6) review the corrosion potentials for zirconium and report no corrosion with sodium silicate concentrations from 0 to 100 weight percent at ambient temperature to 100°C.

Silicon dioxide reacts with hydrofluoric acid to form fluorosilicic acid, which is highly corrosive to zirconium in high concentrations (10 weight percent) (Yau and Webster 1987, [DIRS 100494], p. 712). This reaction is not expected to occur because the pH is generally too high (pH>3.18) for hydrofluoric acid to exist and fluoride concentrations are too low (J13 well

water contains only 2.2 ppm) to form significant concentrations (less than 2×10^{-5} M) of fluorosilicic acid. The fluoride corrosion itself is addressed in Section 6.2.17 and is excluded.

Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding.

In conclusion, enhanced cladding degradation from dissolved silica is excluded from TSPA-LA. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow the omission because silica will not degrade cladding and therefore will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.9 Creep Rupture of Cladding (2.1.02.19.0A)

FEP Description:

At high temperatures (>400 degrees 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 waste package fails, the fuel can react with water and radionuclides can escape over time from the fuel rod.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Creep rupture of cladding is excluded from the TSPA-LA on the basis of low consequence. This FEP applies to the conditions within the repository. The cladding may have experienced creep rupture during its residence in the PWR, but that degradation will be covered in the percentage of damaged rods received after use and storage (i.e., degradation of cladding prior to disposal in Section 6.2.2). This FEP also addresses cladding containing small perforations that cannot allow buildup of internal pressure to cause creep.

The Spent Fuel Project Office of the NRC issued Interim Staff Guidance – 11, Revision 2 (NRC 2002 [DIRS 164593], p. 2), which set a maximum temperature limit for the cladding (400°C) to prevent damage from creep or hydride reorientation during dry storage (similar temperature histories as repository closure). Since peak cladding temperatures in the repository are expected to be less than 300°C (Appendix A), damage from creep is expected to be minimal.

Additional corroborating information is available. A statistical analysis of creep failure was performed (CRWMS M&O 2001 [DIRS 151662], Section 6.2), in which a burnup distribution (rod average = 44 MWd/kgU, range = 2 to 75 MWd/kgU) was used and a distribution of rod properties (including stress) was developed. The Murty creep correlation was selected after comparing six different correlations with results from five different experiments. It was then

modified to better predict irradiated cladding creep data. The fuel rods were exposed to two consecutive temperature histories before being placed in the repository. They were exposed to 24 hours of vacuum drying with a peak temperature of 430°C, followed by 20 years dry storage with a peak temperature of 350°C.

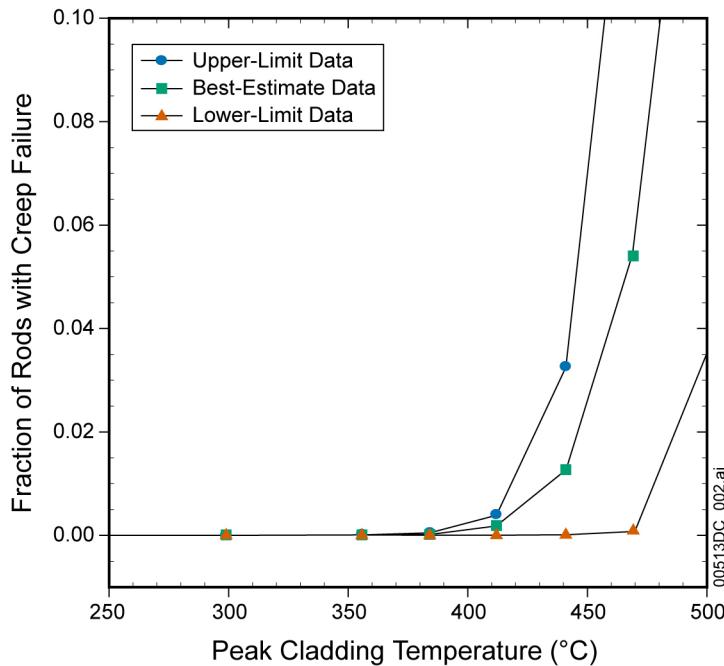
The sensitivity study considered creep strain for the 1,000 years following repository closure, because after that time, the fuel is too cool for creep to occur. Uncertainties in the temperatures and strain rate were included. The radial temperature distribution across the waste package was also modeled. The waste package surface temperature was shifted upward or downward and the fraction of rods that fail from creep was calculated. Failure was predicted when the creep strain of a rod exceeds a creep failure criterion. The upper limit was a 1-percent strain failure criterion and lower limit was 6-percent strain. The best estimate creep failure criterion was a complementary cumulative distribution function (CCDF) based on 52 failure tests. Figure 6-3 shows the results of this sensitivity study. Rod failures started to occur during repository closure at a peak cladding temperature of 400°C. Cladding failure reached 1 percent at a peak temperature of about 430°C (best estimate). In the current repository design, the cladding temperatures are below 300°C (Appendix A) and rod failures from creep are not expected.

The Dry Storage Characterization Project (EPRI 2002 [DIRS 161421], p. xii) studied the condition of fuel assemblies that were exposed to various thermal transients (peak temperatures to 415°C) followed by 15 years of dry storage (peak temperature at 342°C and slowly decreasing). These temperatures are higher than expected at the repository (Appendix A of this report). EPRI concluded “little” or no cladding creep occurred during the thermal benchmark testing and dry storage. It is anticipated that the creep would not increase significantly during additional storage due to the low temperature after 15 years, the continual decrease in temperature from the reduction of decay heat, and the concurrent reduction in pressure and stress.”

Over time, helium production from alpha decay could generate pressures that exceed the pressure at the peak temperature (see Section 6.2.10). The creep rate has a strong temperature dependency (Figures 6.9-1) and the slightly higher stress that occurs at 10,000 years does not change the creep results because the temperatures are less than 73°C and no observable creep occurs at these temperatures and strains.

Stainless steel cladding is modeled for TSPA-LA as failed, therefore, this FEP does not apply to stainless steel cladding.

In conclusion, creep rupture of cladding is excluded from the TSPA-LA on the basis of low consequence. The low repository temperatures are consistent with low to insignificant creep. The NRC requirements in 10 CFR 63.114 (e and f) allow the omission because the small amount of creep that will occur during the regulatory period will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.



Source: CRWMS M&O 2001 [DIRS 151662].

Figure 6-3. Creep Failure Fraction as a Function of Peak Cladding Temperature

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.10 Internal Pressurization of Cladding (2.1.02.20.0A)

FEP Description:

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

Screening Decision:

Excluded (low consequence)

Screening Argument:

Internal pressurization of the cladding is excluded from the TSPA-LA on the basis of low consequence. Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding.

Piron and Pelletier (2001 [DIRS 165318], Section 5.3) investigated the pressurization of the fuel rods from helium production (alpha decay). They concluded that fuel with a burnup of 47.5 MWd/kgU would produce 1,171 cm³ (STP) of helium in a rod after 10,000 years. Piron and Pelletier's (2001 [DIRS 165318], Section 5.3) values are based on having all of the helium released. This provides a total overpressure of 90 bar or 9.10 MPa at 20°C in 10,000 years after emplacement. This corresponds to 9.3 MPa at 30°C. Further discussion on potential pressure

buildup in the void space between the fuel and cladding is in Section 6.2.23. Another report (Rothman 1984 [DIRS 100417], Table 6) calculates a total pressure of 1,530 psi or 10.65 MPa at 30°C for a fuel rod having a burn-up of 36 MWd/kgU after 10,000 years. This latter calculation also includes the added pressure from Xe and Kr considered by Prion and Pelletier (2001 [DIRS 165318]). These results, considering the variability, are within an acceptable tolerance range for this evaluation. The minimum yield strength of Zircaloy-4 is about 241 MPa (Webster 1990 [DIRS 143561], Table 3, p. 665) and, with the internal gas pressure identified above of less than 11 MPa, there is minimal effect due to the internal pressurization of the cladding. Thus, this FEP is excluded based on low consequences.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.11 Stress Corrosion Cracking (SCC) of Cladding (2.1.02.21.0A)**FEP Description:**

Stress corrosion cracking (SCC) 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:

Excluded (low consequence)

Screening Argument:

Stress corrosion cracking of cladding is excluded from the TSPA-LA on the basis of low consequence.

SCC is the formation of brittle cracks on a metal surface through the simultaneous action of a tensile stress and a corrosive environment. SCC requires a certain chemical environment and sufficiently high stresses (stress intensity factors at the crack tip). Chloride-induced SCC could occur on the outside of the cladding. Chloride induced stress corrosion cracking requires that the passive layer of oxides on the zirconium surface be unstable (Cagnolino et al. 1999 [DIRS 152354], p. 4-15; Cox 1973 [DIRS 152920], Abstract; Farina et al. 2002 [DIRS 163639], p. 5; Yau and Webster 1987 [DIRS 100494], p. 718). These are the same conditions under which pitting occurs. As demonstrated in Section 6.2.6, the chemical conditions for pitting and SCC do not exist in the waste package. Therefore, chloride-induced SCC is not expected.

During the times of interest (peak temperature or peak helium pressure) the maximum stress intensity factors vary from 0.47 to 2.73 MPa-m^{0.5}. This is less than the threshold stress intensity factor for SCC, the value at which crack propagation will start. Table 10b of *Clad Degradation – Summary and Abstraction* (CRWMS M&O 2001 [DIRS 151662]) provides the threshold stress intensity factor values for various zirconium alloys in chemical solutions. The reported threshold values are greater than or equal to 4 MPa-m^{0.5}. The threshold stress intensity factor (K_{ISCC}) of 28 MPa-m^{0.5} for moist chlorine is included in this table and pertinent to the case of external cracking (Cox 1990 [DIRS 152778], Figure 20, p. 15). This value is for 70°C. The threshold stress

intensity decreases with increasing temperature (Cox 1990 [DIRS 152778], Figure 14). At a boiling point of 100°C, when water, possibly carrying chlorides, could enter the waste package, the threshold stress intensity could be about $24 \text{ MPa}\cdot\text{m}^{0.5}$. This value is still well above values reported in Table 6.10b of *Clad Degradation – Summary and Abstraction* (CRWMS M&O 2001 [DIRS 151662]). Because the estimated stress intensities are less than the threshold values, chloride-induced cracking is not expected.

Stress corrosion cracking on the interior surface of the cladding from iodine is not expected. In “NRC Issue Resolution Status Report Key Technical Issue: Container Life and Source Term” (Beckman 2001 [DIRS 156122], p. 103), the NRC concluded:

The possibility of SCC induced by iodine as discussed in the WF PMR [Process Model Report] does not appear so important because it is limited essentially by the availability of iodine. The phenomenon as such has been postulated as the cause of pellet cladding interaction failure under reactor operating conditions following steep power ramps, but it does not seem plausible under repository conditions.

Yau and Webster 1987 [DIRS 100494] identified iodine as a possible agent for SCC of zircaloy. However, anticipated iodine concentrations in the repository will be too low for SCC (Beckman 2001 [DIRS 156122], p. 103). Failure has not been observed in dry storage tests with similar conditions to early repository closure. Stress intensities for a sharp crack, (a limiting case, Reed-Hill 1973 [DIRS 121838], p. 800)(up to $2.73 \text{ MPa}\cdot\text{m}^{0.5}$) are lower than the range of threshold stress intensity factors (4.0 to $15 \text{ MPa}\cdot\text{m}^{0.5}$) identified for iodine-induced SCC (Tasooji et al. 1984 [DIRS 102093], Figure 12, p. 612). Cagnolino et al. (1999 [DIRS 152354], p. 4-27) suggested that lower threshold stress intensity factors could cause failure when considering small cracks. The Dry Storage Characterization Project (EPRI 2002 [DIRS 161421]) studied the condition of fuel assemblies that were exposed to various thermal transients (peak temperatures to 415°C) followed by 15 years of dry storage (peak temperature at 342°C and slowly decreasing). These temperatures are higher than expected at the repository (Appendix A). Rod failure during dry storage was not observed, suggesting that iodine induced SCC from small cracks did not fail rods.

Stainless steel cladding is modeled for TSPA-LA as failed, therefore, this FEP does not apply to stainless steel cladding.

Thus, the foregoing discussion has shown that SCC is not likely and hence can be excluded based on low consequence. In conclusion, the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment, would not be significantly changed by the omission of this FEP (stress corrosion cracking (SCC) of cladding) from the performance assessment (TSPA-LA) model.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.12 Hydride Cracking of Cladding (2.1.02.22.0A)

FEP Description:

Cladding contains hydrogen after reactor operation. The cladding might pick up more hydrogen from cladding general corrosion (wet oxidation) after the waste package 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 waste package 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 stress, causing hydride reorientation or delayed hydride cracking. The buildup of hydrides can cause existing cracks to propagate by delayed hydride cracking or hydride embrittlement.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Hydride cracking and embrittlement of the cladding is excluded from the TSPA-LA on the basis of low consequence.

There are several mechanisms of cladding failure that hydrogen and the formation of hydrides can sponsor. Among these are:

1. Hydride embrittlement from zirconium corrosion of cladding
2. Hydride embrittlement from galvanic corrosion
3. Hydride reorientation within cladding
4. Delayed hydride cracking of cladding
5. Hydride axial migration of cladding
6. Hydride embrittlement from fuel reactions.

As the waste package internals corrode, hydrogen is generated, although little is expected to be absorbed directly by the fuel cladding, because H_2 molecules do not migrate through the high-density ZrO_2 fuel cladding layer that is produced during the time the fuel is in the PWR (Section 6.2.24). The hydriding of BWR cladding is less likely due to the difference in waste chemistry. In addition, available data on zirconium hydriding indicate that corrosion of waste package internals will not result in hydriding of fuel cladding, as long as an oxidizing environment exists in the waste package (IAEA 1998 [DIRS 150560], p. 92).

Hydride embrittlement from galvanic corrosion of waste package contacting cladding has been excluded based on low consequence. This is justified because the thick, electrically insulating, oxide layer prevents direct absorption of hydrogen gas and galvanic coupling to dissimilar

metals. If the passive film has been mechanically removed, the unprotected cladding oxidizes within seconds and forms a passive layer when exposed to water or humid air.

Delayed hydride cracking will not be a viable mechanism for cladding failure as the stresses within the waste package are too low to sponsor crack propagation (Cagnolino et al. 1999 [DIRS 152354]; Rothman 1984 [DIRS 100417]). Cladding failure by hydride reorientation is unlikely, because the maximum temperatures are too low to dissolve much hydrogen and the fuel rods have stresses too low for reorientation. Cagnolino et al. (1999 [DIRS 152354]) concluded that hydride reorientation is likely to occur at stresses less than 100MPa at 290°C to 300°C. The cladding material will maintain sufficient strength, even if hydride reorientation were to occur, that failure would not be expected (Puls 1988 [DIRS 102067]).

Hydrogen axial migration will be limited within the temperature gradients expected (one year after closure following 50 years of ventilation) (BSC 2004 [DIRS 166692]):

Time	Centerline	Edge	Gradient
45 years	142.8°C	46.4°C	96.4°C
55 years	103.5°C	29.5°C	74.0°C

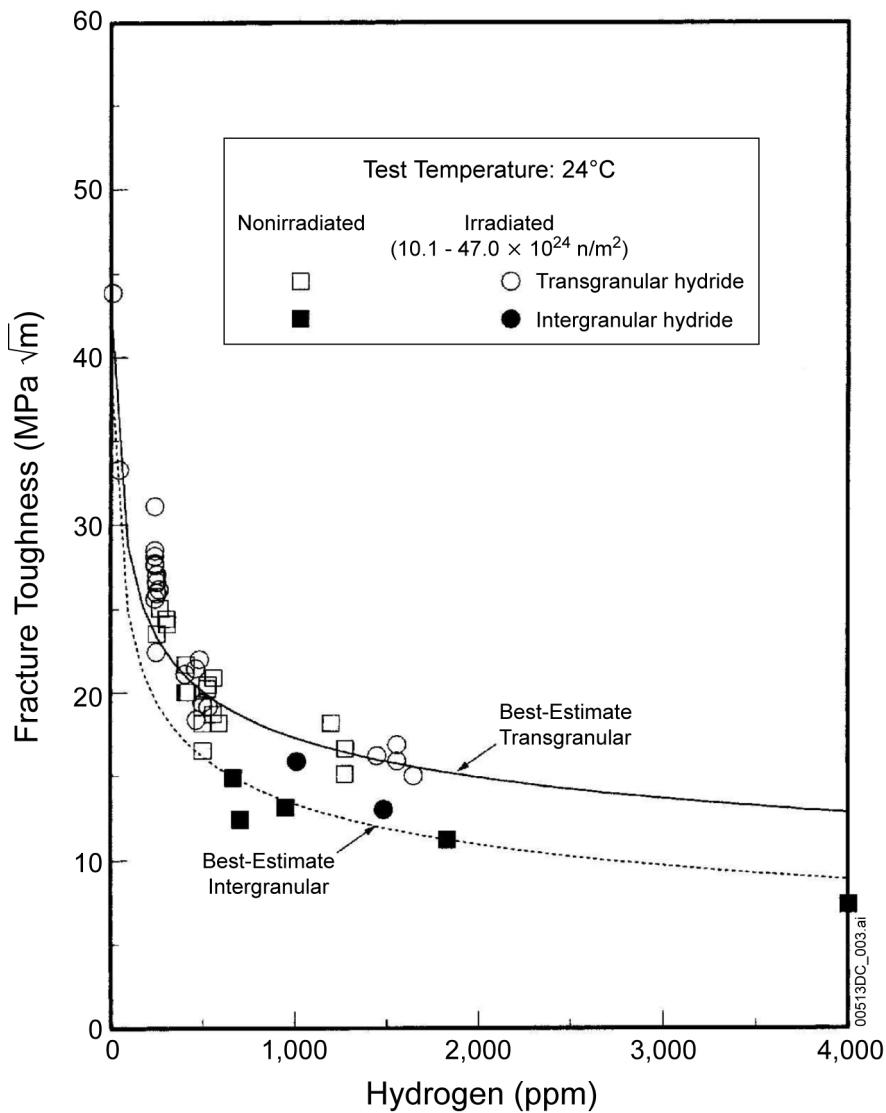
Interpolating gives the approximate temperature gradient across the fuel rod assembly at 51 years of around 83°C. Failure of the cladding by hydrogen embrittlement is unlikely. Hydrogen absorption in the cladding from UO₂ fuel corrosion only occurs in fuel with already failed cladding. Such reaction, if it should occur, has little consequence.

Hydrogen embrittlement in metals usually results in a reduced resistance to fracture. In Zircaloy, hydrogen embrittlement is caused by precipitation of intergranular zirconium hydride. Because Hydride precipitates 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 measured in terms of the critical stress intensity factor, the value that will promote the growth of a crack. This factor is proportional to the far-field stress times the square root of the crack length. Kreyns et al. (1996 [DIRS 100462], Figure 5, reproduced here as Figure 6-4) show that for both irradiated and unirradiated material, such hydrides 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 4,000 ppm. The stress intensity (K_I) maxima calculated from the formula from *Physical Metallurgy Principles* (Reed-Hill 1973 [DIRS 121838], p. 800) varies from 0.47 to 2.73 MPa m^{0.5}. Thus, failure is not expected, even with hydride concentrations as high as 4,000 ppm. The ultimate limit (100 percent hydride and no metal) has a fracture toughness of about 1 MPa·m^{0.5}. The surficial layer of the cladding could be fairly brittle (i.e., hydrogen content greater than 800 ppm), but much of the cladding thickness will have a reasonable toughness.

Stainless steel cladding is modeled for TSPA-LA as failed, therefore, this FEP does not apply to stainless steel cladding.

In conclusion, hydride cracking and embrittlement of the cladding is excluded from the TSPA-LA on the basis of low consequence. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because the small amount of hydride cracking and embrittlement

that will occur during the regulatory period will not significantly change the magnitude and time of the radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment. The stresses in the cladding are not sufficient to fail the cladding at the repository temperatures and experimental data indicate that the in-package environment and cladding stresses are not conducive to hydride cracking and embrittlement.



Source: Kreyns et al. 1996 [DIRS 100462].

Figure 6-4. Fracture Toughness vs. Hydrogen Content of Zircaloy-4

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.13 Cladding Unzipping (2.1.02.23.0A)

FEP Description:

In either dry or wet oxidizing conditions and with perforated fuel cladding, the UO₂ 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:

Included

Screening Argument:

N/A

TSPA Disposition:

Cladding axial splitting or unzipping is included in the TSPA-LA model by instantly splitting the cladding for the full length as soon as the waste package fails and the cladding is perforated. The cladding degradation abstraction models all failed rods (after initial perforation) as bare fuel pellet fragments for the full length of the fuel rod. Failed fuel rod unzipping (cladding axially splits down its length) is caused by the volume increase of corrosion products, Section 6.2.19, (fuel or cladding). It is based on experimental observations at Argonne National Laboratory, where 2 of 2 rods unzipped in less than 2 years. Unzipping leaves fuel pellets exposed to the waste package internal environment. Details of the scientific analysis that describes the disposition in greater detail is presented in *Clad Degradation – Summary and Abstraction for LA* (BSC 2004 [DIRS 170023]).

Unzipping by dry oxidation (oxidation of UO₂ to U₃O₈) of the fuel requires low humidity and high temperature conditions. Dry oxidation is expected to occur in the repository if the waste package fails at closure and the fuel is exposed to the temperatures presented in Appendix A of this report. If it should occur, it also would cause rapid unzipping and is well modeled with the instant unzipping model used in TSPA-LA. Unzipping by wet oxidation was not observed in the ANL tests because unzipping due to the fuel-side cladding corrosion occurred first. Wet oxidation is still fast and can be modeled by the instant unzipping model where necessary.

Supporting Reports:

BSC 2004 [DIRS 170023]

6.2.14 Mechanical Impact on Cladding (2.1.02.24.0A)

FEP Description:

Mechanical failure of cladding may result from external stresses, such as rockfall or impact from waste package internals. Seismic-induced impacts are addressed in a separate FEP.

Screening Decision:

Excluded – Low Probability

Screening Argument:

Mechanical failure of the cladding from external stresses originating outside the waste package has been excluded from the TSPA-LA on the basis of low probability. According to *FEPs*

Screening of Processes and Issues in Drip Shield and Waste Package Degradation (BSC 2004 [DIRS 169997], Section 6.2.23), a nominal rockfall could damage the drip shield but will not deflect the drip shield sufficiently to contact the waste package. Waste package damage from rockfall was screened out based on low probability of occurrence. Therefore, during the 10,000-year regulatory period, no damage to the CSNF is expected, as long as the waste package remains intact. Thus, the waste package prevents rock fall from contacting and degrading the waste form. Hence, damage to the CSNF (waste form and cladding) located inside the waste package would not be expected. The CSNF (waste form and cladding) is located inside the waste package and no damage is expected.

Mechanical failure of the cladding from external stresses originating inside the waste package, such as from the degradation of basket components and other waste package internals, has also been excluded from the TSPA-LA on the basis of low probability of occurrence. Volume increase from the waste package corrosion products is addressed in *EBS Radionuclide Transport Abstraction* (BSC 2004 [DIRS 169868]). Initially the waste package void fraction is 64% (4.7 m³ free volume) but after the waste package internals corrode, the void fraction is 26% (1.9 m³ free volume). Because of the high void fraction, even after corrosion, the waste package corrosion products do not mechanically load and fail the cladding and there is sufficient free volume for any helium that is released from the fuel (FEP 2.1.12.02.0A).

Another approach is to consider dimensional changes in the fuel channel that could be produced from hematite (Fe₂O₃) formation. The inside width of fuel channel is 22.64 cm and the steel component (fuel basket tube) is 0.5 cm thick (BSC 2004 [DIRS 169868]). The iron (density = 7.85 g/cm³) would change to hematite (density = 5.24 g/cm³ (BSC 2004 [DIRS 169868], Table 11). Considering the corrosion products can only form in the fuel channel, the inside dimension would decrease by 0.5 cm [(2 × 0.5 × 7.85/5.24) – 2 × 0.5 = 0.5 cm]. The standard Westinghouse 17 × 17 assembly is 21.4 cm wide (DOE 1996 [DIRS 100320], p. 123). Although the free clearance would decrease from 1.2 cm to 0.7 cm after the formation of the hematite, fuel failure is not expected. In a fuel assembly, the rods do not contact each other and there is coolant spacing between the rods. Since no fuel failure is expected from the iron corrosion, this FEP is excluded in the TSPA-LA.

Thus, mechanical failure of the cladding from external stresses has been excluded on the basis of low probability.

Effects of longer term loading due to volume increase from waste form and cladding corrosion products (i.e., internal stresses) are addressed in FEP 2.1.09.03.0A, Volume Increase of Corrosion Products Impacts Cladding. Seismic-induced rockfall, drift degradation, and ground motion are also not treated within this FEP. A full discussion of seismic effects is contained in FEPs 1.2.03.02.0A, Seismic Ground Motion Damages EBS Components; 1.2.03.02.0B, Seismic Induced Rockfall Damages EBS Components; and 1.2.03.02.0C, Seismic Induced Drift Collapse Damages EBS Components. The effect of seismic events on fuel and cladding is addressed in *Seismic Consequence Analysis* (BSC 2004 [DIRS 167780]).

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.15 Naval SNF Cladding (2.1.02.25.0B)

FEP Description:

DSNF to be disposed of in Yucca Mountain has a variety of fuel types that may not be similar to the CSNF to be disposed. Some of the fuel types may have initial cladding-degradation characteristics that are different from those for the CSNF. Therefore, the effectiveness of DSNF cladding as a barrier to radionuclide mobilization might be different from CSNF. This FEP addresses Naval SNF cladding only.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

Naval Nuclear Propulsion Program cladding and SNF performance is discussed in *Naval Nuclear Propulsion Program Technical Support Document for the License Application*, which is a classified document. Based upon the results from a sensitivity study (BSC 2001 [DIRS 152059]), waste packages containing naval fuel are conservatively modeled as containing CSNF. There are 300 waste packages containing naval SNF compared to the commercial SNF inventory of approximately 7,500 waste packages. A comparison with an equivalent amount of Zircaloy-clad CSNF indicates that the total dose from the TSPA simulation, using the commercial-fuel equivalent, is more than four orders of magnitude higher than the total dose from the source-term simulation for naval SNF (BSC 2001 [DIRS 152059], p. 36 and Figure 6.1-2). Therefore, it is conservative to model naval SNF as CSNF. Details of the scientific analysis that describes the disposition in greater detail are presented in *Clad Degradation – Summary and Abstraction for LA* (BSC 2004 [DIRS 170023], Section 6).

Supporting Reports:

BSC 2004 [DIRS 170023]

6.2.16 Diffusion-Controlled Cavity Growth in Cladding (2.1.02.26.0A)

FEP Description:

Diffusion-controlled cavity growth is a possible creep rupture mechanism that could occur under the temperature and pressure conditions that prevail during dry storage of spent fuel. It might also occur during disposal.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Diffusion-controlled cavity growth (DCCG) in cladding is excluded from the TSPA-LA on the basis of low consequence.

Applicants for dry storage licenses for CSNF were once required by the U.S. Nuclear Regulatory Commission (NRC) to model diffusion-controlled cavity growth to evaluate dry storage designs. That is, NUREG-1536 (NRC 1997 [DIRS 101903], Section 4.V.1) required the use of the DCCG method to calculate a maximum cladding temperature limit for a dry storage design. However, this design limit is overly restrictive and relatively inflexible. The literature does not support the use of this model for zirconium-based materials (Pescatore and Cowgill 1994 [DIRS 102066], pp. 83 to 85) since it has not been validated, and voids and cavities are rarely seen in irradiated Zircaloy. Pescatore and Cowgill (1994 [DIRS 102066], p. 85) recommend a methodology of calculating the amount of creep and comparing it to a creep failure criterion. The earlier NRC Interim Staff Guidance (ISG) Number 11 (NRC 2000 [DIRS 147797]) for transportation and storage recognizes the controversy with the DCCG conceptual model and permits license applicants to use other creep models in their license application. *Cladding Considerations for the Transportation and Storage of Spent Fuel* (NRC 2002 [DIRS 164593]) does not address specific creep models but concludes that creep failures are unlikely if peak temperatures are below 400°C. The temperature profiles for dry storage (time at temperature; CRWMS M&O 2001 [DIRS 151662], Figure 6) are similar to those in the early periods of repository closure when DCCG might occur. With peak cladding temperatures within the repository anticipated to be below 300°C (Appendix A), creep failure (including DCCG failure) is not expected.

Hayes et al. (1999 [DIRS 164598], Figures 2, 5, 6, 8, and 11) concluded that failure from DCCG is unlikely if the peak temperature is less than 330°C to 400°C depending on the specific DCCG model (LLNL versus PNNL), initial stress, and heat decay curve. None of their work suggests failure by DCCG at temperatures as low as 300°C. In summary, omission of this FEP is justified on the basis that DCCG has a low consequence of occurrence because DCCG, as a mechanism to fail Zircaloy cladding, is not likely to significantly degrade the cladding at the temperatures (peak temperatures below 300°C) predicted at the repository.

The creep analysis described in FEP 2.1.02.19.0A is an alternative conceptual model for DCCG model. The creep analysis was selected because of the stronger experimental basis for this type of strain.

Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding.

In conclusion, DCCG in cladding is excluded from the TSPA-LA on the basis of low consequence. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because the impact of DCCG on cladding is likely to be insignificant and therefore will not significantly change the magnitude and time of the radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.17 Localized (Fluoride Enhanced) Corrosion of Cladding (2.1.02.27.0A)

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 waste packages.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Fluoride enhanced corrosion of cladding is excluded from the TSPA-LA on the basis of low consequence.

Hydrofluoric acid can contribute to an accelerated general corrosion rate with fluoride concentrations greater than 5 ppm and pH less than 3.18 (Pourbaix 1974 [DIRS 100817], p. 583). The in-package chemistry model (BSC 2004 [DIRS 167621], Attachment III) predicts pH values greater than 4.5 and J13 well water containing only 2.2 ppm of fluoride. Even if pH dropped below 3, fluoride enhanced localized corrosion of the cladding would not be significant because the fluoride concentration would still be less than 5 ppm.

As corroborating evidence, corrosion of zirconium by fluorides is addressed in *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (CWRMS M&O 2000 [DIRS 136058], Sections 4.1, 6.1.5, 6.2.2.3, and III.4). Zirconium resists attack by most halides, including halogen acids. The major exceptions are hydrofluoric acid and ferric chloride (localized (pitting) corrosion of cladding, FEP 2.1.02.16.0A). As shown in Section 6.1.3 of *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (CWRMS M&O 2000 [DIRS 136058]), zirconium is corrosion resistant to certain fluorides when the pH is sufficiently high. Low fluoride ion concentrations (F⁻ ions), such as a few ppm, in city or ground water have little effect on zirconium's excellent corrosion resistance. However, a few ppm of hydrofluoric acid (HF molecule in solution) will noticeably increase zirconium's corrosion rate. Hydrofluoric acid only exists in solution at pH values below 3.18 (Pourbaix 1974 [DIRS 100817], p. 583).

For accelerated corrosion to occur, the fluoride must be present as free ions (i.e., not complexed as compounds), and the pH must be low. A high insoluble fluoride concentration (in essence a low fluoride ion concentration) would not be expected to have much impact on the standard zirconium corrosion rate. Section 4.1.1 of *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (CWRMS M&O 2000 [DIRS 136058], Test 12) shows that fluoride ion concentrations of less than 5 ppm, even at pH values as low as 1, produce similar corrosion rates to those with zero fluoride ion concentration. Thus, it is reasonable to conclude that low fluoride ion concentrations, as distinct from total fluoride content, will have limited impact on the uniform Zircaloy corrosion rate.

Zirconium and its alloys generally exhibit low corrosion rates in fluoride solutions, including relatively high fluoride ion content solutions, if the temperature is sufficiently low and the pH is sufficiently high. This is illustrated with the results in Section 4.1.1 of *Clad Degradation–Local*

Corrosion of Zirconium and Its Alloys Under Repository Conditions (CWRMS M&O 2000 [DIRS 136058], Test 13). However, if the metal is in contact with solutions containing HF, the corrosion rate can increase rapidly. From the Pourbaix (1974 [DIRS 100817], p. 583) diagram, HF can only exist when the pH is less than 3.18, although this does not necessarily mean that all fluoride ions are immediately converted to HF below this value. The data in Section 4.1.1 of *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions* (CWRMS M&O 2000 [DIRS 136058], Tests 10 through 13) can be divided into fluoride ion-containing solutions and HF-containing solutions using a pH of 3.18 as the demarcation point. The calcium fluoride, due to its low solubility, does not contribute to the fluoride ion concentration (about 2 ppm at 25°C and less than 3 ppm at 90°C).

If the fluoride were to react with the cladding, the amount of corrosion would be limited by the amount of fluoride entering the waste package because the fluoride is consumed in the reaction $\text{Zr} + 4\text{F} \rightarrow \text{Zr F}_4$. For example, if 1 liter of J-13 well water enters a waste package each year, the total contained fluorine content is 2.2 mg (2.2 wt. ppm). If all the available fluorine reacts with zirconium to produce ZrF_4 , the maximum quantity of zirconium that could be corroded away is 2.6 mg per year (4.0×10^{-4} cc/year). This volume represents a general corrosion mechanism and would be distributed over the entire wetted surface. If it were concentrated at one spot of cladding surface area, 0.008 square centimeters of fuel would be exposed (using nominal wall thickness of 500 μm). This quantity of fuel exposure is not significant and, therefore, fluoride corrosion would be a low consequence degradation mechanism. Much larger quantities of water containing fluorides than were used in the example must enter the waste package to fail significant quantities of fuel.

Repository conditions (as represented by J-13 well water) would not be expected to produce any significantly different corrosion rates in zirconium and its alloys than in general corrosion (FEP 2.1.02.13.0A). The well water analysis (Table 6-5) shows a neutral solution with impurity concentrations too low to be corrosive to zirconium and its alloys.

Table 6-5. Compositions of J-13 Well Water

Ion	Concentration (mg/L)
	J-13
SO_4	18.4
Cl	7.14
NO_3	8.78
F	2.18
HCO_3	128.9
Na	45.8
K	5.04
Ca	13.0
Mg	2.01
SiO_2 (aq)	61.0
pH	7.4

Source: BSC 2001 [DIRS 155640] Table 18.

Stainless steel cladding is modeled for TSPA-LA as failed; therefore, this FEP does not apply to stainless steel cladding.

In conclusion, the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment, would not be significantly changed by the omission of this FEP (fluoride enhanced corrosion) from the performance assessment (TSPA-LA) model. The impact on cladding is not expected to be significant because the pH is too high (> 3.18) for the formation of hydrofluoric acid and the concentration of fluorine is too low (< 5 ppm). Because conditions relevant to localized corrosion are not present, localized (fluoride-enhanced) corrosion of cladding has been excluded on the basis of low consequence.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.18 Rockfall (2.1.07.01.0A)

FEP Description:

Rockfalls may occur with blocks that are large enough to mechanically tear or rupture drip shields and/or waste packages. Seismic-induced rockfall is addressed in a separate FEP.

Screening Decision:

Excluded – Low Probability

Screening Argument:

As stated in *FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation* (BSC 2004 [DIRS 169997], Section 6.2.23), nominal rockfall could damage the drip shield but will not deflect the drip shield sufficiently to contact the waste package. Therefore, during the 10,000-year regulatory period, no damage to the waste package is expected. The CSNF (waste form and cladding) is located inside the waste package and no damage is expected. While the fuel and cladding might experience some minor vibration from rockfall, no damage is expected. Fuel and cladding are exposed to vibration during transportation and damage is not observed (Debes 1999 [DIRS 161193]).

Seismic induced rockfall and drift degradation are not considered within this FEP. A full discussion of seismic effects is contained in *Engineered Barrier System Features, Events, and Processes* [DIRS 169898], Sections 6.2.8 through 6.2.10. The effect of seismic events on fuel and cladding is addressed in *Seismic Consequence Analysis* (BSC 2004 [DIRS 167780]). Because the drip shield protects the waste package against rock fall, rock fall damage to the cladding has been excluded based on low probability.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.19 Volume Increase of Corrosion Products Impacts Cladding (2.1.09.03.0A)

FEP Description:

Corrosion products have a higher molar volume than the intact, uncorroded material. Increases in volume during waste form and cladding corrosion could change the stress state in the material being corroded and lead to cladding unzipping.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

The volume increase of corrosion products causes cladding axial splitting, or unzipping, and is included in the TSPA cladding degradation abstraction (BSC 2004 [DIRS 170023], Section 6.2.4). This FEP applies to failed cladding where water or moist air can interact with the fuel or cladding interior. The volume increase of corrosion products inside the cladding causes stress on the cladding and the cladding to tear open. This tearing is modeled to be instantaneous. All failed rods contain fuel pellet fragments for the full length of the fuel rod that are available for dissolution. Failed fuel rod unzipping (cladding axially splits down its length) is caused by the volume increase of corrosion products (fuel or cladding). It is based on experimental observations of two rods at ANL where both rods unzipped in less than two years. Unzipping leaves the fuel pellets exposed to the waste package internal environment. The scientific analysis that describes the disposition in greater detail is presented in *Clad Degradation – Summary and Abstraction for LA* (BSC 2004 [DIRS 170023], Section 6.2.4).

Unzipping by dry oxidation (oxidation of UO_2 to U_3O_8) of the fuel requires low humidity and high temperature conditions. It is expected to occur in the repository if the waste package fails at closure and the fuel is exposed to the temperatures given in Appendix A. If dry oxidation should occur, it also would cause rapid unzipping and is well modeled with the instant unzipping model used in TSPA-LA.

The effects of basket component degradation on external cladding integrity have been evaluated in FEP 2.1.02.24.0A, Mechanical Impact on the Cladding.

Supporting Reports:

BSC 2004 [DIRS 170023]

6.2.20 Electrochemical Effects in EBS (2.1.09.09.0A)

FEP Description:

Electrochemical effects may establish an electric potential within the drift or between materials in the drift and more distant metallic materials. Migration of ions within such an electric field could affect corrosion of metals in the EBS and waste, and could also have a direct effect on the transport of radionuclides as charged ions.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Electrochemical effects (electrophoresis and galvanic coupling) are excluded from the TSPA-LA on the basis of low consequence.

Cladding, with its thick oxide layer produced in reactor operation, is kinetically noble (Yau and Webster 1987 [DIRS 100494], Table 15, p. 718). Any unprotected cladding (due to mechanical removal of the protective oxide layer) oxidizes and forms a passive layer within seconds if exposed to water or humid air (Hansson 1984 [DIRS 101676], p. 6). Inside the waste package, the cladding would behave as a cathode and the carbon steel fuel box would be an anode if they were electrically coupled. The cladding cannot undergo hydrogen charging because the oxide layer prevents hydrogen absorption in the metal (FEP 2.1.12.03.0A).

A report by the Center for Nuclear Waste Regulatory Analysis (Cagnolino 1999 [DIRS 152354], p. 4-13) surveyed various corrosion mechanisms for zirconium cladding under repository conditions. They concluded:

Zr is not susceptible to galvanic corrosion because the protective ZrO_2 passive film leads to E_{corr} values in the galvanic series in flowing seawater close to those of noble metals and graphite but slightly lower than that of Ag (Yau and Webster, 1987). However, local corrosion promoted by galvanic coupling to a more noble metal may occur if the film is mechanically disrupted. Nevertheless, the repassivation rate of Zr and its alloys is sufficiently fast in many aqueous solutions that unless fretting is continuously occurring no substantial corrosion can be expected.

Possible effects of galvanic coupling between cladding and the carbon steel fuel boxes have been considered. After reactor operation, fuel assemblies are coated with a thick oxide, which has a high electrical resistance and would minimize the galvanic effect. After the waste package is breached, the carbon steel fuel box would behave as an anode if coupled with the cladding and could corrode more quickly. An increase in corrosion of the carbon steel fuel box at assembly contact points would have a low consequence. Stainless steel and aluminum plates would maintain the geometry for some time. Galvanic corrosion of cladding or carbon steel fuel boxes is excluded from the TSPA because of low consequence.

Galvanic coupling could theoretically affect the rate of redox reactions involved in waste form matrix degradation when the waste form matrix is an electronic conductor. For example, galvanic coupling between the CSNF matrix (a semiconductor) and cladding could influence the corrosion potential, and hence, the rate of oxidative dissolution of the CSNF matrix. However, data from tests performed on CSNF rod segments (which include the cladding) and on CSNF fragments in Zircaloy holders indicate that such hypothetical galvanic coupling has a negligible effect on the rate of corrosion of the fuel matrix and the associated radionuclide release rate. The effects of galvanic coupling on the corrosion and associated radionuclide release rate from DSNF are also negligible because the base case model for DSNF corrosion causes it to corrode completely in one TSPA-LA time step. Electrochemical effects on DHLW degradation are

negligible because the principal reactions involved are glass network hydrolysis reactions that are not influenced by electrochemical effects.

Electrochemical effect on the waste form will also be minimal. The CSNF waste form is surrounded by the split zirconium cladding and would not be in direct contact with the waste package internals. If the UO₂ did contact the steel fuel boxes and galvanic coupling did occur, the steel boxes would see the accelerated reaction and not the fuel pellets. As the UO₂ corrodes, it coats itself with reaction products, which will minimize galvanic effects. Omission of electrochemical effects (electrophoresis and galvanic coupling) in the waste would not have a significant effect on the resulting radionuclide exposures to the RMEI because those effects are much smaller than the effects of modeling with a minimum flow rate through a failed container of 15 l/yr used in performing the equilibrium-model calculations (CRWMS M&O 2000 [DIRS 136058]). This flow rate has a much greater effect than can be created by electrophoresis or electro-osmosis (Soderman and Jonsson 1996 [DIRS 149441]).

Electrochemical effects of various solutions that might cause pitting of the cladding are addressed in Section 6.2.6. The pitting model (BSC 2004 [DIRS 170043]) is an electrochemical model and the in-package chemistry model (BSC 2004 [DIRS 167621]) addresses the effect of EBS and waste package corrosion on chemistry. Figure 6-2 shows pH variation with time, including the effects of radiolysis. The depression of pH in this figure is caused by waste package steel corrosion. The result from Sections 6.2.5 and 6.2.6 is that no failure is expected.

Telluric currents can, under some conditions, affect corrosion rates in carbon steels, copper alloys and other less corrosion resistant alloy systems. This is especially true for cathodically protected buried carbon steel pipeline systems. However, in the case of the highly passive Alloy 22 waste package outer barrier material and the Titanium Grade 7 drip shield material, telluric current effects would not be expected to affect corrosion response since these materials remain highly passive over a very broad range of applied potentials. Thus small, varying changes in local surface potentials that could potentially result from telluric currents or other stray current sources would not measurably affect the very low corrosion rates of these materials.

In conclusion, electrochemical effects (electrophoresis and galvanic coupling) are excluded from the TSPA-LA on the basis of low consequence. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because electrochemical effects are expected to be small and will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.21 Chemical Effects of Waste-Rock Contact (2.1.09.11.0A)

FEP Description:

Waste (CSNF, DSNF, and HLW) and rock may be placed in direct contact by mechanical failure of the drip shields and/or waste packages. Chemical effects on the waste (e.g. dissolution) may be enhanced or altered in a system where waste, rock minerals, and water are all in physical contact with one another, relative to a system where only waste and water are in physical contact.

Screening Decision:

Excluded (low consequence)

Screening Argument:

The chemical effects of waste–rock contact are excluded from the TSPA-LA on the basis of low consequence.

Waste–rock contact will have no effect on CSNF, DSNF, and HLW dissolution. Water that contacts SNF will have previously been in contact (i.e., equilibrium with the host rock) and therefore, placement of the rock in physical contact with the SNF will not affect the waste–rock–water system chemistry. Furthermore, *In-Package Chemistry Abstraction* (BSC 2004 [DIRS 167621]) demonstrated that variations in the chemistry of the water contacting the waste package internal components, including the SNF, had an insignificant effect on the pH and total carbonate, the two key chemical parameters controlling the solubility of CSNF, DSNF, and HLW, of the in-package fluids. Waste form and cladding interaction with igneous intrusions are addressed in FEP 1.2.04.04.0A, Igneous intrusion interactions with EBS components. This FEP is addressed in *Igneous Intrusion Impacts on Waste Package and Waste Forms* (BSC 2004 [DIRS 168960]).

The chemical effect of waste–rock contact has been excluded from TSPA-LA because of low consequence. Even if near-field chemistry contacts the cladding or waste form, accelerated corrosion is not expected. Tests involving the contact of zirconium with soils have not shown corrosion. Adler Flitton et al. (2002 [DIRS 161991], p. 4) buried various metal samples including zirconium in an arid vadose zone environment for three years. These samples were in intimate contact with the vadose (unsaturated) zone environment. They reported indications of pitting on some of the other metals, but observed no pitting on the zirconium samples when in contact with rock. Other corroborating information is also available. The waste form is protected from the rocks by both the drip shield and waste package during the 10,000-year regulatory period. Zirconium is impervious to most chemical solutions (CRWMS M&O 2000 [DIRS 136058]) and only severe chemicals will pit cladding (BSC 2004 [DIRS 170043]). Yau (1984 [DIRS 102050]) performed corrosion tests with zirconium in geothermal fluids and reports no corrosion. Yau and Webster (1987 [DIRS 100494]) review the limited conditions where corrosion of zirconium is observed. This review suggests that contact with groundwater from YMP tuff would not corrode the cladding.

The CSNF and DSNF waste forms will have little contact with the rock because the stainless steel inner barrier and Alloy 22 outer barrier waste package will prevent contact. For the CSNF and some forms of DSNF, the zirconium alloy cladding surrounding the UO₂ pellets will prevent contact. Contact of the UO₂ pellets with the rock would have little effect. The uranium in the

UO₂ first oxidizes (U⁴⁺ to U⁶⁺), dissolves, and then precipitates immediately on the pellet surface as a U⁶⁺ mineral. The rock would have little contact with the UO₂ pellets and little effect on the CSNF corrosion. DSNF and DHLW are also unlikely to contact the rock and direct effects of such contact, were it to occur, are also negligible. Indirect effects that could occur through the water chemistry (e.g., effects on dissolved silicon concentration) which would feed back into the rate of glass dissolution are small compared to the effects of the glass dissolution, as indicated by the results of tests conducted in J-13 water (i.e., J-13 well water preconditioned by reaction with tuff).

In conclusion, the chemical effects of waste–rock contact are excluded from the TSPA-LA on the basis of low consequence. NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because contact is not expected. Even if it did occur, the resulting chemical effects would not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment. If contact were to occur in the first 10,000 years following permanent closure, no chemical effect is expected.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.22 Thermal Expansion and Stress of In-Package EBS Components (2.1.11.05.0A)

FEP Description:

Thermally induced stresses could alter the performance of the waste or EBS. For example, thermal stresses could cause the waste form to develop cracks and create pathways for preferential fluid flow and, thereby, accelerate degradation of the waste.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Thermal expansion and stresses of in-package EBS components, including the waste form, are excluded from the TSPA-LA on the basis of low consequence.

The waste package and its internals are designed for the thermal expansion from the temperatures discussed in Appendix A. The CSNF and DSNF are designed for the thermal cycles expected in reactors, which are more severe than the conditions anticipated to occur in the repository. As discussed in Section 6.2.1 of *CSNF Waste Form Degradation: Summary Abstraction* (BSC 2004 [DIRS 169987]), the in-reactor thermal cycles (principally that associated with the initial power escalation) result in extensive cracking of the fuel matrix. The effects of this cracking are included in the specific surface area parameter. Glass logs crack because of the cooldown during manufacturing. The cracking that results from this cooldown is included in the DHLW model surface area parameter. This cooldown (from molten glass, about 950°C, BSC 2004 [DIRS 169988], Section 7.5.3) is more severe than repository conditions.

Commercial nuclear fuel operates at higher temperatures than expected during the post-closure period at the repository. Under normal conditions, typical cladding operates at about 320°C (Garde et al. 1991 [DIRS 101652], p. 582) with fuel centerline temperatures reaching 1,800°C (Lanning et al. 1997 [DIRS 101704], V3, p. 3.2, Figure 3.1). Fuel is also designed to undergo anticipated operating occurrences (off normal transients that occur during the design life) without damage. These are more severe thermal cycles than normal reactor operation or repository closure. Every time a reactor shuts down and goes to cold shutdown, the fuel is cooled to below 100°C (coolant is less than boiling). These temperature transients are more severe than repository closure. DSNF is also exposed to reactor transients more severe than the post-closure cooldown. Because the temperature transients for spent fuel from normal in-reactor operations and for DHLW from normal manufacturing cooldown are more severe than the transient associated with repository closure, no further degradation (cracking) is expected from thermal expansion or secondary thermal stresses of in-package EBS components.

In conclusion, thermal expansion and stress of in-package EBS components, including the waste form is excluded from the TSPA-LA on the basis of low consequence. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because thermal expansion will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.23 Gas Generation (He) From Waste Form Decay (2.1.12.02.0A)**FEP Description:**

Helium (He) gas production may occur by alpha decay in the waste. Helium production might cause local pressure buildup in cracks in the fuel and in the void between fuel and cladding, leading to cladding and waste package failure.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Effects of helium gas generation from alpha decay on the cladding in spent fuel are excluded from the TSPA-LA on the basis of low consequence.

Piron and Pelletier (2001 [DIRS 165318], Section 5.3) investigated pressurization of the fuel rods from alpha decay helium production and concluded that fuel with a burn-up of 47.5 MWd/kgU would produce 1,171 cm³ (STP) of helium in a rod after 10,000 years, too low a quantity to damage the fuel. The internal pressurization of less than 11 MPa (Section 6.2.10) due to gas build up would have to be significantly higher for the cladding to fail from delayed hydride cracking.

For cladding, delayed hydride cracking, stress corrosion cracking, 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. For failed rods, the helium would be released into the waste package, possibly increasing the pressure there. Using complete helium release from the fuel is conservative at these temperatures. Helium is an inert gas and will not chemically react with in-package components (internals, cladding or UO₂ pellets). After waste package failure, it will also tend to diffuse out. The helium would have little effect on the corrosion of the UO₂ pellets. UO₂ corrosion is an oxidation-dissolution phenomenon and the presence of traces of an inert gas will not affect this. The effects of microcracking due to pressure buildup in gas bubbles within the fuel matrix on the rate of matrix corrosion are expected to be negligible. As described in Sections 6.2.1 and 6.2.2 of *CSNF Waste Form Degradation: Summary Abstraction* (BSC 2004 [DIRS 169987]), evidence from CSNF testing indicates that the corrosion process is a general corrosion process occurring predominantly at the periphery of the fuel fragments. The effective specific surface area of the corroding fuel is comparable to the geometric surface area and is not sensitive to the internal grain boundary decohesion or microcracking that may result from helium buildup in the matrix. Likewise, because DHLW dissolution occurs at the periphery of glass fragments, the internal microstructural features (including helium gas bubbles) do not affect the dissolution rate.

For an intact waste package, the free volume is 4.7 m³ (Section 6.2.14). If the waste package has failed, any helium that is released from the failed fuel can escape the waste package. It is expected that less than 1 percent of the fuel is failed (Section 6.2.2). If all of the rods were considered failed and all of the helium was released from the fuel matrix, then approximately (rounding up) 1,500 cm³ (STP) of gas (Section 6.2.10) would be released per rod. With 5,544 rods per waste package (BSC 2004 [DIRS 170023], Table 8-1), there would be 8.3×10^6 cm³ (STP) of helium available. When distributed into the free volume, 4.7 m³ (Section 6.2.14), of the waste package at 73°C (nominal peak temperature at 10,000 years; BSC 2004 [DIRS 169565] Figure 6.3-67), the pressure would increase by 0.22 MPa (2.2 atm). This pressure rise is negligible and, thus, would have no affect on the waste package. With most of the fuel rods intact, and most of the helium tied up in the fuel matrix, the actual pressure rise would be even smaller.

In conclusion, effects of helium gas generation from alpha decay on the cladding are excluded from the TSPA-LA on the basis of low consequence. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because the pressures generated by helium generation are insufficient to damage the cladding and, therefore, will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.24 Gas Generation (H_2) From Waste Package Corrosion (2.1.12.03.0A)

FEP Description:

Gas generation can affect the mechanical behavior of the host rock and engineered barriers, chemical conditions, and fluid flow, and, as a result, the transport of radionuclides. Gas generation due to oxic corrosion of waste packages, cladding, and/or structural materials will occur at early times following closure of the repository. Anoxic corrosion may follow the oxic phase if all oxygen is depleted. The formation of a gas phase around the waste package may exclude oxygen from the iron, thus inhibiting further corrosion.

Screening Decision:

Excluded (low consequence)

Screening Argument:

Hydrogen gas (H_2) generation from waste package corrosion, as a degradation mechanism for cladding, is excluded from the TSPA-LA on the basis of low consequence. The cladding absorbs little or no hydrogen from the outside environment, so waste package corrosion and hydride embrittlement will not occur.

Many investigators have considered the hydriding of zirconium alloys (for example, the papers cited by Clayton (1989 [DIRS 149208], 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. *Waterside Corrosion of Zirconium Alloys in Nuclear Power Plants* (IAEA 1998 [DIRS 150560], 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 1 ppm of the hydrogen were from dissolved tritium gas, the radioactivity of T_2 would result in thousands of decays per second per one square centimeter of surface. This level of activity would be readily detected. The IAEA (1998 [DIRS 150560], 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 this quotation, the hydrogen pressures required to cause hydriding are quite large. By comparison, the highest hydrogen pressure in a breached waste package is pure hydrogen gas at atmospheric pressure, or about 0.1 MPa.

Additional corroborating information follows. The oxide film on the surface of the metal is important in preventing the uptake of hydrogen from the surrounding environment. Under specific experimental conditions, this oxide layer can be removed. At 200°C, the solubility of oxygen in zirconium is greater than 1 percent by weight, and the solubility of oxygen increases

with increasing temperature (Baker 1992 [DIRS 149104], p. 2-326). As a result, the oxide film on the surface of a piece of zirconium is normally not thermodynamically stable. In his discussion of experimental procedure, Smith (1966 [DIRS 149107], p. 325) notes, "Zirconium samples were first annealed at 700°C under vacuum ($\sim 10^{-3}$ mm Hg) to remove any oxide film. The film dissolved into the samples, 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 [DIRS 100817], p. 226). Spent nuclear fuel has a robust oxide layer from reactor operation. Under the oxidizing and humid environment of a failed waste package, it is expected that the oxide layer will be preserved.

Garzarolli et al. (1979 [DIRS 149256], p. 64) studied the effect of the oxygen supply and stated:

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.

Hydriding of zirconium by absorption of gas has been observed in the laboratory. For example, Smith (1966 [DIRS 149107], Table 3) provides data on hydrogen absorption. However, the environment for these experiments was hot, extremely pure hydrogen. Smith (1966 [DIRS 149107], Table 3) states that the temperatures for the hydrogen absorption experiments were 210°C to 700°C. Smith (1966 [DIRS 149107], p. 325) notes, "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, 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.

Hydrogen would have no effect on in-package chemistry because it would immediately diffuse out of the package before reacting with anything. Also, hydrogen is a reduced gas and is, therefore, unlikely to form in an oxidizing environment. The uranium in the UO_2 first oxidizes (U^{4+} to U^{6+}), dissolves, and then precipitates immediately on the pellet surface as a schoepite-type mineral. Although the kinetics of hydrogen gas reactions are sluggish under the conditions expected in a breached waste package, the presence of hydrogen would reduce the concentration of oxidizing agents and hence the corrosion rate of the UO_2 ; neglecting this effect is, therefore, conservative. The presence of some hydrogen gas would not affect the hydrolysis reactions that control the rate of DHLW dissolution. The hydrogen would be generated for the first 500 years while the waste package internals are corroding so this conservatism exists for a limited time.

In conclusion, hydrogen gas (H_2) generation from waste package corrosion, as a degradation mechanism for cladding, is excluded from the TSPA-LA on the basis of low consequence. The

cladding absorbs little or no hydrogen from the outside environment such as waste package corrosion and hydride embrittlement will not occur. The NRC requirements in 10 CFR 63.114 (e and f) [DIRS 156605] allow this omission because H₂ will not damage the cladding and therefore will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

INTENTIONALLY LEFT BLANK

7. CONCLUSIONS

This report summarizes the screening of 24 waste form and cladding degradation-related FEPs for possible inclusion in the TSPA-LA. The FEPs and the result of their screening are delineated in Table 7-1. This table provides the FEP number, name, screening decision (included/excluded), and basis for the exclusion (i.e., low probability or low consequence). Low consequence means that omitting the FEP will not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual (RMEI), or radionuclide releases to the accessible environment (10 CFR 63.114 e and f [DIRS 156605]).

For the first nineteen FEPs in Table 7-1, the FEP screening applies to CSNF and naval spent fuel. The last five FEPs in Table 7-1 address cladding and all waste forms including DOE-owned spent nuclear fuel (DSNF) and defense high-level waste (HLW). Uncertainties pertaining to the implementation of the included FEPs are identified in the supporting reports listed at the end of each included FEP description. The results of this screening process are documented in the screening arguments and TSPA-LA disposition statements in Section 6.2.

For the repository, cladding is classified as a barrier to release as part of the waste form. However, it is not designed nor controlled by the project. The behavior can be modeled but the design characteristics are not within the project controls.

Any restrictions to the use of this document are inherently captured in the discussion of the included FEP and supporting reports. No other restrictions or limitations are identified.

Table 7-1. Summary of Cladding Degradation and Waste Form FEPs

FEP Number	FEP Name	Screening Decision and Basis	Section
2.1.02.11.0A	Degradation of cladding from waterlogged rods	Excluded (Low Consequence)	6.2.1
2.1.02.12.0A	Degradation of cladding prior to disposal	Included	6.2.2
2.1.02.13.0A	General corrosion of cladding	Excluded (Low Consequence)	6.2.3
2.1.02.14.0A	Microbially influenced corrosion (MIC) of cladding	Excluded (Low Consequence)	6.2.4
2.1.02.15.0A	Localized (radiolysis enhanced) corrosion of cladding	Excluded (Low Consequence)	6.2.5
2.1.02.16.0A	Localized (pitting) corrosion of cladding	Excluded (Low Consequence)	6.2.6
2.1.02.17.0A	Localized (crevice) corrosion of cladding	Excluded (Low Consequence)	6.2.7
2.1.02.18.0A	Enhanced corrosion of cladding from dissolved silica	Excluded (Low Consequence)	6.2.8
2.1.02.19.0A	Creep rupture of cladding	Excluded (Low Consequence)	6.2.9
2.1.02.20.0A	Internal pressurization of cladding	Excluded (Low Consequence)	6.2.10
2.1.02.21.0A	Stress corrosion cracking (SCC) of cladding	Excluded (Low Consequence)	6.2.11
2.1.02.22.0A	Hydride cracking of cladding	Excluded (Low Consequence)	6.2.12
2.1.02.23.0A	Cladding unzipping	Included	6.2.13
2.1.02.24.0A	Mechanical impact on cladding	Excluded (Low Probability)	6.2.14
2.1.02.25.0B	Naval SNF cladding	Included	6.2.15
2.1.02.26.0A	Diffusion-controlled cavity growth in cladding	Excluded (Low Consequence)	6.2.16
2.1.02.27.0A	Localized (fluoride enhanced) corrosion of cladding	Excluded (Low Consequence)	6.2.17
2.1.07.01.0A	Rockfall	Excluded (Low Probability)	6.2.18
2.1.09.03.0A	Volume increase of corrosion products impacts cladding	Included	6.2.19

Table 7-1. Summary of Cladding Degradation and Waste Form FEPs (Continued)

FEP Number	FEP Name	Screening Decision and Basis	Section
2.1.09.09.0A	Electrochemical effects in EBS	Excluded (Low Consequence)	6.2.20
2.1.09.11.0A	Chemical effects of waste-rock contact	Excluded (Low Consequence)	6.2.21
2.1.11.05.0A	Thermal expansion/stress of in-package EBS components	Excluded (Low Consequence)	6.2.22
2.1.12.02.0A	Gas generation (He) from waste form decay	Excluded (Low Consequence)	6.2.23
2.1.12.03.0A	Gas generation (H ₂) from waste package corrosion	Excluded (Low Consequence)	6.2.24

The conclusions from this document (FEP screening decision, TSPA-LA disposition for included FEPs, or screening argument for excluded FEPs), along with any modifications to the FEP list, names, or descriptions are incorporated in the Yucca Mountain TSPA-LA FEP database. The FEP database contains all Yucca Mountain FEPs considered for TSPA-LA with FEP number, name, description, and relevant reports where the documentation of the screening of specific FEPs is summarized. The FEP database also contains screening decisions (included or excluded), screening arguments, and TSPA-LA dispositions quoted from this and other FEP reports.

7.1 YMRP ACCEPTANCE CRITERIA

The purpose of this report is to evaluate and document the inclusion or exclusion of cladding degradation FEPs with respect to modeling used to support the TSPA-LA. A screening decision, either *Included* or *Excluded*, was given for each FEP along with the corresponding technical basis for the excluded FEPs and the descriptions of how the included FEPs were incorporated in the TSPA-LA. This information was required by the U.S. Nuclear Regulatory Commission (NRC) regulations at 10 CFR 63.114 (d, e, and f) [DIRS 156605].

The acceptance criteria, identified as applicable to this analysis in Section 4.2, that are related to the FEPs screening process were addressed in this analysis. The following acceptance criteria are based on meeting the requirements at 10 CFR 63.114(e) and (f) [DIRS 156605].

7.1.1 Scenario Analysis Acceptance Criteria

The following acceptance criteria (AC) from *Yucca Mountain Review Plan, Review Plan* (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3) were addressed in this report:

Acceptance Criterion 1—The Identification of a List of Features, Events, and Processes Is Adequate

(1) *The Safety Analysis Report contains a complete list of features, events, and processes, related to the geologic setting or the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers), that have the potential to influence repository performance. The list is consistent with the site characterization data. Moreover, the comprehensive features, events, and processes list includes, but is not limited to, potentially disruptive events related to igneous activity (extrusive and intrusive); seismic shaking (high-frequency-low magnitude, and rare large-*

magnitude events); tectonic evolution (slip on existing faults and formation of new faults); climatic change (change to pluvial conditions); and criticality.

Documentation of the origin of the FEPs list is provided in Section 6.1.1; FEP descriptions are provided in Section 6.2. This analysis contains a list of cladding degradation-related FEPs (Tables 1-1 and 7-1). This list of FEPs includes those related to the degradation, deterioration, or alteration of engineered barriers. The list of cladding degradation-related FEPs is consistent with the site characterization data.

Acceptance Criterion 2—Screening of the List of Features, Events, and Processes Is Appropriate

(1) The U.S. Department of Energy has identified all features, events, and processes related to either the geologic setting or to the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) that have been excluded;

The relevant FEPs related to the degradation, deterioration, or alteration of engineered barriers were identified. These FEPs were screened for inclusion in the TSPA-LA. See Table 7-1 for a list of excluded cladding degradation-related FEPs.

(2) The U.S. Department of Energy has provided justification for those features, events, and processes that have been excluded. An acceptable justification for excluding features, events, and processes is that either the feature, event, and process is specifically excluded by regulation; probability of the feature, event, and process (generally an event) falls below the regulatory criterion; or omission of the feature, event, and process does not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment; and

For the FEPs that were excluded from TSPA-LA by either low probability, low consequence, or by regulation, the justification was provided in the appropriate subsections of Section 6.2, documenting the basis for the exclusion. See the method and approach discussion provided in Section 6.1.2 for an explanation of the use of various types of justification.

(3) The U.S. Department of Energy has provided an adequate technical basis for each feature, event, and process, excluded from the performance assessment, to support the conclusion that either the feature, event, or process is specifically excluded by regulation; the probability of the feature, event, and process falls below the regulatory criterion; or omission of the feature, event, and process does not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment.

For the FEPs that were excluded from TSPA-LA by either low probability, low consequence, or by regulation, the appropriate technical basis was provided in the subsections of Section 6.2, documenting the basis for the exclusion. See the method and approach discussion provided in Section 6.1.2 for an explanation of the use of various types of justification.

INTENTIONALLY LEFT BLANK

8. INPUTS AND REFERENCES

8.1 DOCUMENTS CITED

161991 Adler Flitton, M.K.; Mizia, R.E.; and Bishop, C.W. 2002. "Underground Corrosion of Activated Metals in an Arid Vadose Zone Environment." *Corrosion/2002*, [57th Annual Conference & Exposition, April 7-11, 2002, Denver, Colorado]. Paper No. 02531. Houston, Texas: NACE International. TIC: 253838.

103753 ASM International. 1987. *Corrosion*. Volume 13 of *Metals Handbook*. 9th Edition. Metals Park, Ohio: ASM International. TIC: 209807.

149104 Baker, H. 1992. "Binary Alloy Phase Diagrams." Volume 3 of *ASM Handbook*. Page 2.326. Materials Park, Ohio: ASM International. TIC: 247622.

156122 Beckman, D.A. 2001. "NRC Issue Resolution Status Report on Container Life and Source Term, Revision 3." Memorandum from D.A. Beckman (BSC) to Distribution, February 21, 2001, LV.LAP.AH.02/2001-236, with enclosure. ACC: MOL.20010418.0048.

101564 Bradley, E.R.; Bailey, W.J.; Johnson, A.B., Jr.; and Lowry, L.M. 1981. *Examination of Zircaloy-Clad Spent Fuel After Extended Pool Storage*. PNL-3921. Richland, Washington: Pacific Northwest Laboratory. TIC: 230049.

161988 Brossia, C.S.; Cagnolino, G.A.; and Dunn, D.S. 2002. "Effect of Oxide Thickness on the Localized Corrosion of Zircaloy." *Corrosion/2002*, [57th Annual Conference & Exposition, April 7-11, 2002, Denver, Colorado]. Paper No. 02549. Houston, Texas: NACE International. TIC: 253839.

155640 BSC (Bechtel SAIC Company) 2001. *Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier*. ANL-EBS-MD-000001 REV 00 ICN 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010724.0082.

152059 BSC 2001. *Performance Assessment of U.S. Department of Energy Spent Fuels in Support of Site Recommendation*. CAL-WIS-PA-000002 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010627.0026.

168030 BSC 2004. *Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada*. ANL-CRW-GS-000003 REV 00 [Errata 001]. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20000510.0175; DOC.20040223.0007.

170023 BSC 2004. *Clad Degradation – Summary and Abstraction for LA*. ANL-WIS-MD-000021 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041004.0002.

169987 BSC 2004. *CSNF Waste Form Degradation: Summary Abstraction*. ANL-EBS-MD-000015 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040908.0001.

166692 BSC 2004. *D&E/PA/C IED Typical Waste Package Components Assembly*. 800-IED-WIS0-00206-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040202.0014.

169988 BSC 2004. *Defense HLW Glass Degradation Model*. ANL-EBS-MD-000016, Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company.

169868 BSC 2004. *EBS Radionuclide Transport Abstraction*. ANL-WIS-PA-000001, Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company.

169898 BSC 2004. *Engineered Barrier System Features, Events, and Processes*. ANL-WIS-PA-000002, Rev. 03. Las Vegas, Nevada: Bechtel SAIC Company.

169991 BSC 2004. *Evaluation of Potential Impacts of Microbial Activities on Drift Chemistry*. ANL-EBS-MD-000038, Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company.

169997 BSC 2004. *FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation*. ANL-EBS-PA-000002 REV 03. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041012.0004.

168960 BSC 2004. *Igneous Intrusion Impacts on Waste Packages and Waste Forms*. MDL-EBS-GS-000002 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040421.0002.

167621 BSC 2004. *In-Package Chemistry Abstraction*. ANL-EBS-MD-000037, Rev. 03. Las Vegas, Nevada: Bechtel SAIC Company.

169565 BSC 2004. *Multiscale Thermohydrologic Model*. ANL-EBS-MD-000049, Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company.

170043 BSC 2004. *Pitting Model for Zirconium-Alloyed Cladding*. MDL-WIS-MD-000001 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040827.0009.

168361 BSC 2004. *Q-List*. 000-30R-MGR0-00500-000-000 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040721.0007.

167780 BSC 2004. *Seismic Consequence Abstraction*. MDL-WIS-PA-000003 REV 0 Errata 1. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20030818.0006; DOC.20040218.0002.

169183 BSC 2004. *Seismic Consequence Abstraction*. MDL-WIS-PA-000003, Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company.

171583 BSC 2004. *Technical Work Plan For: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package*. TWP-WIS-MD-000009 REV 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040910.0001.

168706 BSC 2004. *The Development of the Total System Performance Assessment-License Application Features, Events, and Processes*. TDR-WIS-MD-000003, Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company.

170020 BSC 2004. *Waste-Form Features, Events, and Processes*. ANL-WIS-MD-000009, Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company.

166275 Canori, G.F. and Leitner, M.M. 2003. *Project Requirements Document*. TER-MGR-MD-000001 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031222.0006.

149208 Clayton, J.C. 1989. "Internal Hydriding in Irradiated Defected Zircaloy Fuel Rods." *Zirconium in the Nuclear Industry: Eighth International Symposium*, held June 19-23, 1988 San Diego, California. ASTM STP 1023. Pages 266-288. Philadelphia, Pennsylvania: American Society for Testing and Materials. TIC: 241414.

152920 Cox, B. 1973. "Stress Corrosion Cracking of Zircaloy-2 in Neutral Aqueous Chloride Solutions at 25 C." *Corrosion*, 29, (4), 157-166. Houston, Texas: National Association of Corrosion Engineers. TIC: 248988.

152778 Cox, B. 1990. "Environmentally-Induced Cracking of Zirconium Alloys – A Review." *Journal of Nuclear Materials*, 170, 1-23. [Amsterdam, The Netherlands]: North-Holland. TIC: 234774.

152354 Cragnolino, G.A.; Dunn, D.S.; Brossia, C.S.; Jain, V.; and Chan, K.S. 1999. *Assessment of Performance Issues Related to Alternate Engineered Barrier System Materials and Design Options*. CNWRA 99-003. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. TIC: 248875.

102829 CRWMS M&O 1995. *Analysis of Degradation Due to Water and Gases in MPC*. BB0000000-01717-0200-00005 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19960419.0202.

136058 CRWMS M&O 2000. *Clad Degradation–Local Corrosion of Zirconium and Its Alloys Under Repository Conditions*. ANL-EBS-MD-000012 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000405.0479.

151659 CRWMS M&O 2000. *Initial Cladding Condition*. ANL-EBS-MD-000048 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001002.0145.

151662 CRWMS M&O 2001. *Clad Degradation – Summary and Abstraction*. ANL-WIS-MD-000007 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010214.0229.

161193 Debes, M. 1999. "Transportation in France." Letter from O. Dekens (EDF) to A.J. Machiels (EPRI), November 15, 1999, with attachment. TIC: 249077.

100320 DOE (U.S. Department of Energy) 1996. *Spent Nuclear Fuel Discharges from U.S. Reactors 1994*. SR/CNEAF/96-01. Washington, D.C.: U.S. Department of Energy. TIC: 232923.

100442 Einziger, R.E. 1994. "Preliminary Spent LWR Fuel Oxidation Source Term Model." High Level Radioactive Waste Management, Proceedings of the Fifth Annual International Conference, Las Vegas, Nevada, May 22-26, 1994. 2, 554-559. La Grange Park, Illinois: American Nuclear Society. TIC: 210984.

161421 EPRI (Electric Power Research Institute) 2002. *Dry Cask Storage Characterization Project*. EPRI TR-1002882. Palo Alto, California: Electric Power Research Institute. TIC: 253737.

163639 Farina, S.B.; Duffo, G.S.; and Galvele, J.R. 2002. "Stress Corrosion Cracking of Zirconium and Zircaloy-4 in Iodine Containing Solutions." Corrosion/2002, [57th Annual Conference & Exposition, April 7-11, 2002, Denver, Colorado]. Paper No. 02436. Houston, Texas: NACE International. TIC: 253841.

101652 Garde, A.M. 1991. "Enhancement of Aqueous Corrosion of Zircaloy-4 Due to Hydride Precipitation at the Metal-Oxide Interface." Zirconium in the Nuclear Industry, Ninth International Symposium, Kobe, Japan, November 5-8, 1990. Eucken, C.M. and Garde, A.M., eds. ASTM-STP-1132, 566-594. Philadelphia, Pennsylvania: American Society for Testing and Materials. TIC: 246113.

149256 Garzarolli, F.; von Jan, R.; and Stehle, H. 1979. "The Main Causes of Fuel Element Failure in Water-Cooled Power Reactors." Atomic Energy Review, 17, (1), 31-128. [Vienna, Austria]: International Atomic Energy Agency. TIC: 221365.

145073 Greene, C.A.; Brossia, C.S.; Dunn, D.S.; and Cagnolino, G.A. 2000. "Environmental and Electrochemical Factors on the Localized Corrosion of Zircaloy-4." Corrosion/2000, [55th Annual Conference & Exposition, March 26-31, 2000, Orlando, Florida]. Paper No. 00210. Houston, Texas: NACE International. TIC: 246988.

171573 Güldner, R. and Burtak, F. 1999. "Contribution of Advanced Fuel Technologies to Improved Nuclear Power Plant Operation." The Uranium Institute Twenty Fourth Annual International Symposium 1999. [London, England]: Uranium Institute. Accessed September 13, 2004. TIC: 256420. <http://www.world-nuclear.org/sym/1999/guldner.htm> Copyright Requested

101676 Hansson, C.M. 1984. *The Corrosion of Zircaloy 2 in Anaerobic Synthetic Cement Pore Solution*. SKB TR-84-13. Stockholm, Sweden: Svensk Kärnbränsleförsörjning A.B. TIC: 206293.

164598 Hayes, T.A.; Rosen, R.S.; and Kassner, M.E. 1999. *Critical Analysis of Dry Storage Temperature Limits for Zircaloy-Clad Spent Nuclear Fuel Based on Diffusion Controlled Cavity Growth*. UCRL-ID-131098. Livermore, California: Lawrence Livermore National Laboratory. TIC: 254551.

100455 Hillner, E.; Franklin, D.G.; and Smee, J.D. 1998. *The Corrosion of Zircaloy-Clad Fuel Assemblies in a Geologic Repository Environment*. WAPD-T-3173. West Mifflin, Pennsylvania: Bettis Atomic Power Laboratory. TIC: 237127.

150560 IAEA (International Atomic Energy Agency) 1998. *Waterside Corrosion of Zirconium Alloys in Nuclear Power Plants*. IAEA-TECDOC-996. Vienna, Austria: International Atomic Energy Agency. TIC: 248234.

169906 Jones, D.A. 1992. *Principles and Prevention of Corrosion*. 1st Edition. New York, New York: Macmillan. TIC: 242631.

123682 Knoll, R.W. and Gilbert, E.R. 1987. *Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel*. PNL-6365. Richland, Washington: Pacific Northwest Laboratory. TIC: 213789.

131519 Kohli, R. and Pasupathi, V. 1986. *Investigation of Water-logged Spent Fuel Rods Under Dry Storage Conditions*. PNL-5987. Richland, Washington: Pacific Northwest Laboratory. TIC: 246472.

100462 Kreyns, P.H.; Bourgeous, W.F.; White, C.J.; Charpentier, P.L.; Kammenzind, B.F.; and Franklin, D.G. 1996. "Embrittlement of Reactor Core Materials." Zirconium in the Nuclear Industry, Eleventh International Symposium held in Garmisch-Partenkirchen, Germany, September 11-14, 1995. Bradley, E.R. and Sabol, G.P., eds. ASTM STP 1295. Pages 758-782. West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 237256.

100051 Langmuir, D. 1997. *Aqueous Environmental Geochemistry*. Upper Saddle River, New Jersey: Prentice Hall. TIC: 237107.

101704 Lanning, D.D.; Beyer, C.E.; and Painter, C.L. 1997. *FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application*. NUREG/CR-6534. Volume 1. Richland, Washington: Pacific Northwest National Laboratory. TIC: 238923.

160832 Lide, D.R., ed. 2002. *CRC Handbook of Chemistry and Physics. 83rd Edition*. Boca Raton, Florida: CRC Press. TIC: 253582.

131533 Little B. and Wagner P. 1996. "An Overview of Microbiologically Influenced Corrosion of Metals and Alloys Used in the Storage of Nuclear Wastes." Canadian Journal of Microbiology, 42, (4), 367-374. Ottawa, Canada: National Research Council of Canada. TIC: 246614.

101719 Manaktala, H.K. 1993. *Characteristics of Spent Nuclear Fuel and Cladding Relevant to High-Level Waste Source Term*. CNWRA 93-006. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. TIC: 208034.

131537 McNeil, M. and Odom, A. 1994. "Thermodynamic Prediction of Microbiologically Influenced Corrosion (MIC) by Sulfate-Reducing Bacteria (SRB)." *Microbiologically Influenced Corrosion Testing*. ASTM STP 1232. Pages 173-179. Philadelphia, Pennsylvania: American Society for Testing and Materials. TIC: 246989.

100468 Merriam-Webster. 1993. *Merriam-Webster's Collegiate Dictionary, 10th Edition*. Springfield, Massachusetts: Merriam-Webster. TIC: 8883.

101903 NRC (U.S. Nuclear Regulatory Commission) 1997. *Standard Review Plan for Dry Cask Storage Systems*. NUREG-1536. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20010724.0307.

147797 NRC 2000. "Interim Staff Guidance - 11. Storage of High Burnup Spent Fuel." ISG - 11. Washington, D.C.: U.S. Nuclear Regulatory Commission. Accessed March 21, 2000. TIC: 247227. <http://www.nrc.gov/OPA/reports/isg11.htm>

164593 NRC 2002. *Interim Staff Guidance - 11, Revision 2. Cladding Considerations for the Transportation and Storage of Spent Fuel*. ISG-11, Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 254441.

163274 NRC 2003. *Yucca Mountain Review Plan, Final Report*. NUREG-1804, Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards. TIC: 254568.

102065 Peehs, M. and Fleisch, J. 1986. "LWR Spent Fuel Storage Behaviour." *Journal of Nuclear Materials*, 137, (3), 190-202. Amsterdam, The Netherlands: North-Holland Publishing Company. TIC: 235595.

102066 Pescatore, C. and Cowgill, M. 1994. *Temperature Limit Determination for the Inert Dry Storage of Spent Nuclear Fuel*. EPRI TR-103949. Palo Alto, California: Electric Power Research Institute. TIC: 102933.

165318 Piron, J.P. and Pelletier, M. 2001. "State of the Art on the Helium Issues." Section 5.3 of *Synthesis on the Long Term Behavior of the Spent Nuclear Fuel*. Poinssot, C., ed. CEA-R-5958(E). Volume 1. [Paris], France: Commissariat à l'Énergie Atomique. TIC: 253976.

100817 Pourbaix, M. 1974. *Atlas of Electrochemical Equilibria in Aqueous Solutions*. Houston, Texas: National Association of Corrosion Engineers. TIC: 208955.

102067 Puls, M.P. 1988. "The Influence of Hydride Size and Matrix Strength on Fracture Initiation at Hydrides in Zirconium Alloys." *Metallurgical Transactions A*, 19A, (6), 1507-1522. Metals Park, Ohio: American Society for Metals. TIC: 237143.

161992 Ralph, R.; Purohit, A.; and Dille, E. 2002. "Characterization of Oxide Formation on Dresden Unit 1 Fuel Storage Pool Racks." Tenth International Conference on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors, August 5 to 9, 2001, Lake Tahoe, Nevada. Houston, Texas: NACE International. TIC: 252999.

121838 Reed-Hill, R.E. 1973. *Physical Metallurgy Principles*. Second Edition. 797-801. New York, New York: D. Van Nostrand Company. TIC: 237154.

100417 Rothman, A.J. 1984. *Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Nuclear Fuel in a Tuff Repository*. UCID-20172. Livermore, California: Lawrence Livermore National Laboratory. ACC: NNA.19870903.0039.

149107 Smith, T. 1966. "Kinetics and Mechanism of Hydrogen Permeation of Oxide Films on Zirconium." *Journal of Nuclear Materials*, 18, 323-336. Amsterdam, The Netherlands: Elsevier. TIC: 247624.

149441 Soderman, O. and Jonsson, R. 1996. "Electro-Osmosis: Velocity Profiles in Different Geometries with Both Temporal and Spatial Resolution." *Journal of Chemical Physics*, 105, (23), 10300-10311. New York, New York: American Institute of Physics. TIC: 247669.

166303 Štefanic I. and LaVerne, J.A. 2002. "Temperature Dependence of the Hydrogen Peroxide Production in the γ -Radiolysis of Water." *Journal of Physical Chemistry*, 106, (2), 447-452. [Washington, D.C.]: American Chemical Society. TIC: 255323.

102093 Tasooji, A.; Einziger, R.E.; and Miller, A.K. 1984. "Modeling of Zircaloy Stress-Corrosion Cracking: Texture Effects and Dry Storage Spent Fuel Behavior." *Zirconium in the Nuclear Industry, Sixth International Symposium*, Vancouver, British Columbia, June 28- July 1, 1982. Franklin, D.G. and Adamson, R.B., eds. ASTM STP 824. Pages 595-626. Philadelphia, Pennsylvania: American Society for Testing and Materials. TIC: 241417.

143561 Webster, R.T. 1990. "Zirconium and Hafnium." Volume 2 of *Metals Handbook*. 10th Edition. Pages 661-669. Materials Park, Ohio: ASM International. TIC: 241059.

100097 Wolery, T.J. and Daveler, S.A. 1992. *EQ6, A Computer Program for Reaction Path Modeling of Aqueous Geochemical Systems: Theoretical Manual, User's Guide, and Related Documentation (Version 7.0)*. UCRL-MA-110662 PT IV. Livermore, California: Lawrence Livermore National Laboratory. ACC: MOL.19980701.0459.

165268 Wolfram, J.H.; Mizia, R.E.; Jex, R.; Nelson, L.; and Garcia, K.M. 1996. *The Impact of Microbially Influenced Corrosion on Spent Nuclear Fuel and Storage Life*. INEL-96/0335. Idaho Falls, Idaho: Idaho National Engineering Laboratory, Lockheed Martin Idaho Technologies Company. ACC: MOL.20030925.0039.

102050 Yau, T. L. 1984. "Zirconium Versus Corrosive Species in Geothermal Fluids." Paper 140 of Corrosion 84. Houston, Texas: National Association of Corrosion Engineers. TIC: 238987.

110761 Yau, T.L. and Maguire, M. 1990. "Control of Localized Corrosion of Zirconium in Oxidizing Chloride Media." Advances in Localized Corrosion, Proceedings of the Second International Conference on Localized Corrosion, June 1-5, 1987, Orlando, Florida. Isaacs, H.S.; Bertocci, U.; Kruger, J. and Smialowska, S., eds. Pages 311-319. Houston, Texas: National Association of Corrosion Engineers. TIC: 247083.

100494 Yau, T.L. and Webster, R.T. 1987. "Corrosion of Zirconium and Hafnium." In Corrosion, Volume 13, Pages 707-721 of ASM Handbook. Formerly 9th Edition, Metals Handbook. [Materials Park, Ohio]: ASM International. TIC: 240704.

149233 Yau, T-L. 1983. "Corrosion Properties of Zirconium in Chloride Solutions." Corrosion 83, International Corrosion Forum, Anaheim, California, April 18-22, 1983. Paper No. 26. Pages 26/1 - 26/13. Houston, Texas: National Association of Corrosion Engineers. TIC: 243849.

8.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

103540 10 CFR 60. Energy: *Disposal of High-Level Radioactive Wastes in Geologic Repositories*. Readily available.

156605 10 CFR 63. Energy: *Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada*. Readily available.

AP-2.22Q, Rev. 1. ICN 1 *Classification Analyses and Maintenance of the Q-List*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040714.0002.

AP-2.27Q, Rev. 1, ICN 4. *Planning for Science Activities*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC. 20040610.0006.

AP-3.15Q, Rev. 4, ICN 5. *Managing Technical Product Inputs*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC. 20040812.0004.

AP-SIII.2Q Rev. 1. ICN 2 *Qualification of Unqualified Data*. Washington, DC: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040127.0008.

AP-SIII.9Q, Rev. 1. ICN 7. *Scientific Analyses*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040920.0001.

AP-SV.1Q, Rev. 1, ICN 1. *Control of the Electronic Management of Information*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040308.0001.

LP-SI.11Q-BSC, Rev. 0, ICN 1. *Software Management*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20041005.0008.

105725 ASTM C 1174-97. 1998. *Standard Practice for Prediction of the Long-Term Behavior of Materials, Including Waste Forms, Used in Engineered Barrier Systems (EBS) for Geological Disposal of High-Level Radioactive Waste*. West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 246015.

8.3 SOURCE DATA, LISTED BY DATA TRACKING NUMBER

170760 MO0407SEPFEPLA.000. *LA FEP List*. Submittal date: 07/20/2004.

INTENTIONALLY LEFT BLANK

APPENDIX A

CLADDING TEMPERATURE ESTIMATES FOR POSTCLOSURE

INTENTIONALLY LEFT BLANK

A. CLADDING TEMPERATURE ESTIMATES FOR POSTCLOSURE

Many of the FEPs in this document depend on temperature. Peak temperatures are important for some FEPs such as delayed hydride cracking. Other FEPs such as general corrosion or creep depend on both the temperature levels and how long the temperatures stay elevated.

The cladding temperatures are predicted by adding the interior temperature rise across the waste package to the waste package surface temperature. The waste package surface temperatures (BSC 2004 [DIRS 169565], Section 6.2) vary with waste package location in the repository, water infiltration flux, and time (heat from radioactive decay). The temperature rise across the waste package (BSC 2004 [DIRS 166692]) depends on the thermal loading (decay heat power) of the waste package and the loading pattern. The peak cladding temperature for a specific rod will also depend on the location of the rod within the waste package. The maximum cladding temperatures are in the center location and the coolest locations are along the outer row of locations.

A.1 EVALUATION OF CLADDING TEMPERATURE

The temperature histories from *Multiscale Thermohydrologic Model* (BSC 2004 [DIRS 169565], Figure 6.3-67) were developed based on all waste packages and accounting for the uncertainty of the host-rock thermal conductivity and percolation flux. The maximum waste package surface temperature of around 205°C (BSC 2004 [DIRS 169565], Figure 6.3-67) occurs between 65 and 70 years after emplacement (15 to 20 years after ventilation period). The waste package surface temperatures for this time period varies from around 100°C to 205°C due to the uncertainties.

The internal waste package temperature difference (the temperature difference from the waste package surface to a specific location inside the waste package) decreases with time after emplacement (BSC 2004 [DIRS 166692]). Also, the decrease in the internal waste package temperature difference is less than the waste package surface temperature increase from the time after ventilation until the peak waste package temperature peaks between 15 to 20 years afterwards (BSC 2004 [DIRS 169565], Figure 6.3-67). Thus, maximum cladding temperature can be estimated at 65 years after emplacement, which results in an estimate of 296°C (205°C waste package surface temperature + 91.1°C waste package internal temperature increase (BSC 2004 [DIRS 166692]), using the internal temperature difference (T) from the surface of the 21-PWR waste package with 50 years of ventilation for 65 years in the center at the hottest location). After the peak at 65 years both the waste package surface and internal temperature increase will continue to decrease.

This evaluation shows that with the predicted waste package surface temperatures, using the wide range of uncertainties, and the predicted waste package internal temperatures at the time when the peak temperatures are occurring, the temperature for the cladding remains below 300°C.

The above represents the TSPA nominal scenario repository environment where the drift remains open. Higher waste package surface and consequently cladding temperatures are possible in the case of a complete drift collapse early in the postclosure period. This could only occur as a result of a seismic event. The seismic event probabilities, consequences, and treatment for the TSPA-LA model are discussed in *Seismic Consequence Abstraction* (BSC 2004 [DIRS 169183]).

INTENTIONALLY LEFT BLANK

APPENDIX B

QUALIFICATION OF DATA THAT SUPPORTS CLAD DEGRADATION - FEPs SCREENING ARGUMENTS (ANL-WIS-MD-000008)

INTENTIONALLY LEFT BLANK

B. QUALIFICATION OF DATA THAT SUPPORTS *CLAD DEGRADATION - FEPs SCREENING ARGUMENTS* (ANL-WIS-MD-000008)**B.1 PURPOSE**

The “Clad Degradation - FEPs (Features, Events and Processes) Screening Arguments” (analysis) relies on data that may not have been collected under an approved quality assurance program that meets the requirements of 10 CFR Part 63, Subpart G [DIRS 156605] or its predecessor, 10 CFR Part 60 [DIRS 103540]. This appendix qualifies those data that the cladding degradation analysis uses as direct input. The appendix was prepared according to AP-SIII.9Q, *Scientific Analyses*. The outside sources of data that are used as direct inputs to the screening of FEPs being qualified for use within this report are addressed in this appendix.

B.2 METHODS AND CRITERIA

The data obtained from outside sources that are not established facts are documented in this appendix for use within this technical report. The extent to which the data demonstrate the properties of interest must be addressed.

One or more of the following factors from AP-SIII.9Q, Section 5.2.1 shall be used to qualify data obtained from an outside source:

- Reliability of data source
- Qualifications of personnel or organizations generating the information,
- Prior uses of the data, and
- Availability of corroborating data.

B.3 QUALIFICATION OF DATA

The procedure does not apply to “Established Fact” data or numerical data obtained from an established/authoritative data source. Because some of the data comes from authoritative sources (other than the U.S. Department of Energy, Office of Civilian Radioactive Waste Management), it does not need to be qualified. The remaining information is qualified only for its intended use (i.e., to justify excluding from performance assessment twenty-one modes of cladding degradation). Some of the exclusionary arguments depend on qualitative arguments from the open literature that the analysis cites as data.

Table B-1 lists sources of data and the factors used to qualify the information. The table designates the attributes or factors that are used to qualify the information. The data to be qualified is described in the following subsections.

Table B-1. Sources of Technical Information and Criteria Used to Qualify the Information

Sources	Factors
Greene, C.A.; Brossia, C.S.; Dunn, D.S.; and Cragnolino, G.A. 2000. "Environmental and Electrochemical Factors on the Localized Corrosion of Zircaloy-4." Corrosion/2000, [55th Annual Conference & Exposition, March 26-31, 2000, Orlando, Florida]. Paper No. 00210. Houston, Texas: NACE International. TIC: 246988. DIRS 145073	Properties of Interest Availability of corroborating data.
Hansson, C.M. 1984. <i>The Corrosion of Zircaloy 2 in Anaerobic Synthetic Cement Pore Solution</i> . SKB TR-84-13. Stockholm, Sweden: Svensk Kärnbränsleförsörjning A.B. TIC: 206293. DIRS 101676	Properties of Interest Availability of corroborating data.
Hayes, T.A.; Rosen, R.S.; and Kassner, M.E. 1999. <i>Critical Analysis of Dry Storage Temperature Limits for Zircaloy-Clad Spent Nuclear Fuel Based on Diffusion Controlled Cavity Growth</i> . UCRL-ID-131098. Livermore, California: Lawrence Livermore National Laboratory. TIC: 254551. DIRS 164598	Properties of Interest Availability of corroborating information.
Hillner, E.; Franklin, D.G.; and Smee, J.D. 1998. <i>The Corrosion of Zircaloy-Clad Fuel Assemblies in a Geologic Repository Environment</i> . WAPD-T-3173. West Mifflin, Pennsylvania: Bettis Atomic Power Laboratory. TIC: 237127. DIRS 100455	Properties of Interest Availability of corroborating data.
Piron, J.P. and Pelletier, M. 2001. "State of the Art on the Helium Issues." Section 5.3 of <i>Synthesis on the Long Term Behavior of the Spent Nuclear Fuel</i> . Poinssot, C., ed. CEA-R-5958(E). Volume 1. [Paris], France: Commissariat à l'Énergie Atomique. TIC: 253976. DIRS 165318	Properties of Interest Availability of corroborating information
Wolfram, J.H.; Mizia, R.E.; Jex, R.; Nelson, L.; and Garcia, K.M. 1996. <i>The Impact of Microbially Influenced Corrosion on Spent Nuclear Fuel and Storage Life</i> . INEL-96/0335. Idaho Falls, Idaho: Idaho National Engineering Laboratory, Lockheed Martin Idaho Technologies Company. ACC: MOL.20030925.0039. DIRS 165268	Properties of Interest Availability of corroborating data

Both the discussions above and in the following sections fulfill requirements of AP-SIII.9Q to discuss the data for qualification, the method of qualification, and the evaluation criteria (AP-SIII.9Q, Section 5.2.1, paragraph 1)).

B.3.1 Qualification of Data from Greene, C.A.; Brossia, C.S.; Dunn, D.S.; and Cragnolino, G.A. 2000. [DIRS 145073]

Properties of Interest—This analysis (Section 6.7) evaluates the potential for localized (crevice) corrosion of the cladding. The conclusion of this source was that crevice corrosion was not observed. These data are integral to evaluation of the cladding degradation FEPs.

Extent and Quality of Corroborating Information—Two sources corroborate the above source. Yau and Webster (1987 [DIRS 100494], p. 717), an authoritative source (ASM International 1987 [DIRS 103753], p. 717; an established fact), reported that zirconium resists crevice corrosion. In low-pH, chloride solutions zirconium is not subject to crevice attack. Brossia et al. (2002 [DIRS 161988]) performed pitting and crevice corrosion tests on Zircaloy-4. They report that no crevice corrosion is observed under the same environment and electrochemical conditions that promote pitting corrosion on exposed surfaces.

B.3.2 Qualification of Data from Hansson 1984 [DIRS 101676]

Properties of Interest — This analysis assesses the extent to which silica could affect cladding corrosion. The analysis considers high dissolved silica content in water, the source considers the silica content in concrete, and an authoritative source (Yau and Webster 1987 [DIRS 100494]) considers the silica concentration in sea water. All three anticipate little cladding degradation that can be attributed to silica concentrations that far exceed those that the cladding would encounter in the repository.

Extent and Quality of Corroborating Information — An authoritative source, Yau and Webster (1987 [DIRS 100494]) corroborates this source's assessment of cladding's resistance to enhanced corrosion at the silica concentrations estimated for the repository.

B.3.3 Qualification of Data from Hayes et al. 1999 [DIRS 164598]

Properties of Interest — This analysis assesses creep as a mechanism for failure of Zircaloy cladding. The analysis does not expect creep at peak repository temperatures (<300°C), and others expect little creep damage under 400°C. Evaluation of failure via this mechanism is essential to the screening of cladding degradation FEPs.

Availability of Corroborating Information — An authoritative source (NRC 2002 [DIRS 164593]), corroborates this source's assessment of creep. Both state that cladding failure attributed to creep is unlikely for peak temperatures below 400°C.

B.3.4 Qualification of Data from Hillner et al. 1998 [DIRS 100455]

Properties of Interest — This analysis evaluates the potential for microbes to cause cladding failure. The corrosion of Zircaloy-clad fuels via microbes under repository conditions and the organic acids produced by microbes are unlikely to significantly accelerate zirconium corrosion. Microbially induced corrosion is one of the FEPs that requires evaluation therefore, the data presented in this reference is essential.

Availability of Corroborating Information — Three independent sources corroborate Hillner et al. (1998). Wolfram et al. (1996 [DIRS 165268]) found no evidence in the literature that zirconium or its alloys are susceptible to microbially induced corrosion. Yau and Webster (1987 [DIRS 100494]), an authoritative source, states that zirconium resists organic acids that microbes produce. And the Yucca Mountain Project (BSC 2004 [DIRS 169991], Section 7.1) evaluated in-drift microbial communities and concluded that they would have little effect on the in-drift geochemistry.

B.3.5 Qualification of Data from Piron and Pelletier 2001 [DIRS 165318]

Properties of Interest — This analysis considers whether helium could pressurize the cladding and cause it to fail. They concluded that fuel with a burnup of 47.5 MWd/kgU would produce 1,171 cm³ (STP) of helium in a rod after 10,000 years. Piron and Pelletier's (2001 [DIRS 165318], Section 5.3) values are based on having all of the helium released. This provides a total overpressure of 90 bar or 9.10 MPa at 20°C in 10,000 years after emplacement. This corresponds to 9.3 MPa at 30°C.

Availability of Corroborating Information — Manaktala (1993 [DIRS 101719]) and Rothman (1984 [DIRS 100417]) corroborate this source's assertion that helium production would not fail the cladding. Both found that cooling of the rods would offset helium pressure buildup. Consequently, the pressure at 10,000 years would be less than what this source predicted, and the likelihood of cladding failure would be even less likely.

B.3.6 Qualification of Data from Wolfram et al. 1996 [DIRS 165268]

Properties of Interest — This analysis does not expect microbes to cause cladding failure. It found no evidence in the literature that zirconium or its alloys are susceptible to microbial-induced corrosion. Because the evaluation of microbially induced corrosion is one of the FEPs being screened, these data are essential.

Availability of Corroborating Information — Three independent sources corroborate Wolfram et al. (1996 [DIRS 165268]). Hillner et al. (1998 [DIRS 100455]) supports the concept that organic acids produced by microbes are unlikely to significantly accelerate zirconium corrosion. Yau and Webster, 1987 [DIRS 100494], an authoritative source, states that zirconium resists organic acids that microbes produce. And the Yucca Mountain Project (BSC 2004 [DIRS 169991], Section 7.1) evaluated in-drift microbial communities and concluded that they would have little effect on the in-drift geochemistry.

B.4 CONCLUSION

In conclusion, the data are qualified for their intended use.