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FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation

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Change History

12. Revision No.	13. Description of Change
00	Initial Issue.
00/01	Interim Change Notice. Update for no backfill base case.
01	Revision 01. Updated to include secondary REP discussions.Note: No black vertical lines were used to demote changes from the previous revisions, since extensive changes have been made since to ICN 01 of this document
02	Revision 02. Updated to conform to the Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain.Note: No black vertical lines were used to denote changes from the previous revision, because the changes were too extensive to use Step 5.6c)1) of AP-SIII.9Q
03	Revision 03. Updated for Regulatory Integration. Changes were too extensive for change bars to be practical.

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TABLES

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ACRONYMS

AC	acceptance criterion
ASM	American Society for Metals
BSC	Bechtel SAIC Company, LLC
BWR	boiling water reactor
CDSP	codisposal
CFR	Code of Federal Regulations
CRWMS	Civilian Radioactive Waste Management System
CSNF	commercial spent nuclear fuel
DC	direct current
DHLW	defense high-level radioactive waste
DOE	U.S. Department of Energy
DIRS	Document Input Reference System
DSNF	defense spent nuclear fuel
DTN	Data Tracking Number
EBS	engineered barrier system
FEP	feature, event, and process
Gy	Gray, where 1 Gray = 100 rads
HIC	hydrogen induced cracking
LA	License Application
LTCTF	Long Term Corrosion Test Facility
m	Molality, moles per kilogram of solvent
M	Molarity, moles per liter of solution
M&O	Management and Operating Contractor
MIC	microbially influenced corrosion
MTU	Metric Tons of Uranium
N/A	Not Applicable
NRC	U.S. Nuclear Regulatory Commission
Pb	Lead
PRD	Project Requirements Document
PWR	pressurized water reactor
QA	quality assurance
RMEI	reasonably maximally exposed individual

ACRONYMS (Continued)

SAW	simulated acidified water
SCC	stress corrosion cracking
SCE	saturated calomel electrode
SCW	simulated concentrated water
SDW	simulated dilute water
SNF	spent nuclear fuel
SR	Site Recommendation
TSPA	Total System Performance Assessment
TSPA-LA	Total System Performance Assessment for License Application
TWP	Technical Work Plan
UNS	Unified Numbering System
U.S.	United States
WAPDEG	Waste Package Degradation (software title)
WF	Waste Form
YMP	Yucca Mountain Project
YMRP	Yucca Mountain Review Plan

1. PURPOSE

The purpose of this report is to evaluate and document the inclusion or exclusion of features, events and processes (FEPs) with respect to drip shield and waste package modeling used to support the Total System Performance Assessment for License Application (TSPA-LA). Thirty-three FEPs associated with the waste package and drip shield performance have been identified (DTN: MO0407SEPFEPLA.000 [DIRS 170760]). A screening decision, either "included" or "excluded," has been assigned to each FEP, with the technical bases for screening decisions, as required by the Nuclear Regulatory Commission (NRC) in 10 CFR 63.114 (d, e, and f) [DIRS 156605].

The FEPs analyses in this report address issues related to the degradation and potential failure of the drip shield and waste package over the post closure regulatory period of 10,000 years after permanent closure. For included FEPs, this report summarizes the disposition of the FEP in TSPA-LA. For excluded FEPs, this report provides the technical bases for the screening arguments for exclusion from TSPA-LA.

The analyses are for the TSPA-LA base-case design (BSC 2004 [DIRS 168489]), where a drip shield is placed over the waste package without backfill over the drip shield (BSC 2004 [DIRS 168489]). Each FEP includes one or more specific issues, collectively described by a FEP name and description. The FEP description encompasses a single feature, event, or process, or a few closely related or coupled processes, provided the entire FEP can be addressed by a single specific screening argument or TSPA-LA disposition. The FEPs were assigned to associated Project reports, so the screening decisions reside with the relevant subject-matter experts.

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]).

1.2 SCOPE

The scope of this report is to identify the treatment of the FEPs affecting waste package and drip shield post closure performance over the 10,000-year regulatory period following permanent closure. Igneous and seismic effects are not treated within this report. A discussion of igneous and seismic FEPs can be found in *Features, Events, and Processes: Disruptive Events* (BSC 2004 [DIRS 170017]).

The scope for this activity involves two tasks, namely:

1. Identify the drip shield and waste package FEPs included in the TSPA-LA analysis and the reports in which they are addressed.

2. Identify the drip shield and waste package FEPs excluded from analysis in the TSPA-LA and justify why they are excluded.

This report describes and evaluates the technical bases for screening decisions associated with the drip shield and waste package FEPs. For included FEPs, this report summarizes how the FEP is addressed in TSPA-LA. For excluded FEPs, this report provides the technical basis for the decision to exclude (i.e., low probability, low consequence, or by regulation). In cases where a FEP encompasses multiple technical areas and is shared with other FEPs documents, this report provides a partial technical basis for the screening decision as it relates to drip shield and waste package. Collectively, the documents address the complete technical basis. This information is provided in Section 6.2 and subsequent sections.

An overview of the Yucca Mountain Project 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), which describes 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 drip shield and waste package FEP analysis. The list of drip shield and waste package TSPA-LA FEPs, presented in Table 1-1, was derived from DTN: MO0407SEPFEPLA.000 [DIRS 170760] with subsequent modifications to the FEP list, FEP names, and/or FEP descriptions. These modifications are documented in the "FEP History File" in the FEP database (BSC 2004 [DIRS 168706], Table 6-1) and will be incorporated into a subsequent revision of the TSPA-LA FEP list (see Section 7). Table 1-1 also includes the designation of shared FEPs.

Direct inputs supporting the screening decisions are listed in Section 4. Indirect inputs supporting the screening decisions are listed in Section 6.1. The individual FEP discussions providing identification (FEP number, name, and description) and screening information (screening decision and screening argument (if FEP is excluded) or TSPA disposition (if FEP is included)) are in Section 6.2.

Table 1-1. Drip Shield and Waste Package FEPs for TSPA-LA

FEP Number	FEP Name	Addressed in Section	Sharing With
1.1.03.01.0A	Error in Waste Emplacement	6.2.1	EBS
2.1.03.01.0A	General Corrosion of Waste Packages	6.2.2	
2.1.03.01.0B	General Corrosion of Drip Shields	6.2.3	
2.1.03.02.0A	Stress Corrosion Cracking (SCC) of Waste Packages	6.2.4	
2.1.03.02.0B	Stress Corrosion Cracking (SCC) of Drip Shields	6.2.5	
2.1.03.03.0A	Localized Corrosion of Waste Packages	6.2.6	
2.1.03.03.0B	Localized Corrosion of Drip Shields	6.2.7	
2.1.03.04.0A	Hydride Cracking of Waste Packages	6.2.8	
2.1.03.04.0B	Hydride Cracking of Drip Shields	6.2.9	
2.1.03.05.0A	Microbially Influenced Corrosion (MIC) of Waste Package	6.2.10	
2.1.03.05.0B	Microbially Influenced Corrosion (MIC) of Drip Shields	6.2.11	
2.1.03.06.0A	Internal Corrosion of Waste Packages Prior to Breach	6.2.12	WF
2.1.03.07.0A	Mechanical Impact on Waste Package	6.2.13	
2.1.03.07.0B	Mechanical Impact on Drip Shield	6.2.14	
2.1.03.08.0A	Early Failure of Waste Packages	6.2.15	
2.1.03.08.0B	Early Failure of Drip Shields	6.2.16	
2.1.03.09.0A	Copper Corrosion in EBS	6.2.17	

Table 1-1. Drip Shield and Waste Package FEPs for TSPA-LA (Continued)

FEP Number	FEP Name	Addressed in Section	Sharing With
2.1.03.10.0A	Advection of Liquids and Solids Through Cracks in the Waste Package	6.2.18	EBS
2.1.03.10.0B	Advection of Liquids and Solids Through Cracks in the Drip Shield	6.2.19	EBS
2.1.03.11.0A	Physical Form of Waste Package and Drip Shield	6.2.20	
2.1.06.06.0B	Oxygen Embrittlement of Drip Shields	6.2.21	
2.1.06.07.0B	Mechanical Effects at EBS Component Interfaces	6.2.22	EBS
2.1.07.01.0A	Rockfall	6.2.23	EBS, CLAD
2.1.07.05.0A	Creep of Metallic Materials in the Waste Package	6.2.24	
2.1.07.05.0B	Creep of Metallic Materials in the Drip Shield	6.2.25	
2.1.09.03.0B	Volume Increase of Corrosion Products Impacts Waste Package	6.2.26	
2.1.09.09.0A	Electrochemical Effects in EBS	6.2.27	CLAD
2.1.11.06.0A	Thermal Sensitization of Waste Packages	6.2.28	
2.1.11.06.0B	Thermal Sensitization of Drip Shields	6.2.29	
2.1.11.07.0A	Thermal Expansion/Stress of In-Drift EBS Components	6.2.30	EBS
2.1.12.03.0A	Gas Generation (H_2) from Waste Package Corrosion	6.2.31	EBS, CLAD
2.1.13.01.0A	Radiolysis	6.2.32	EBS, WF
2.1.13.02.0A	Radiation Damage in EBS	6.2.33	EBS, WF

NOTES: FEPs = features, events, and processes; EBS = engineered barrier system; WF = waste form; CLAD = cladding.

1.3 SCIENTIFIC ANALYSIS LIMITATION AND USAGE

This report, the documentation for inclusion or exclusion of drip shield and waste package FEPs, provides information for a database that promotes FEPs traceability and transparency. The following limitations apply to this report:

- Because this report cites other documents as direct input, its limitations inherently include limitations or constraints described within those documents.

- Where FEPs are shared, this report is limited to performance of the drip shield and waste package.

- For screening purposes, this report generally uses mean values of probabilities and either amplitudes of events or of consequences (e.g., mean time to degradation) as a basis for reaching an "include" or "exclude" decision. Mean values are determined based on the range of possible values.

- The screening results presented are specific to the Yucca Mountain Project repository design and processes. Changes in direct inputs listed in Section 4.1, in baseline conditions used for this evaluation or in other subsurface conditions, must be assessed to determine whether the changes are within the limits stated in the FEP evaluations. Engineering and design changes are reviewed to determine adverse impacts to nuclear safety, in accordance with 10 CFR 63.73, Subparts F and G [DIRS 156605], and 10 CFR 63.44 [DIRS 156605] requirements.

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2. QUALITY ASSURANCE

Development of this report is 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).

This document contributes to the analysis and modeling used to support TSPA-LA. The FEPs documented herein involve the investigations of items or barriers on the Q-list and have the potential to affect the calculation of the performance of the natural barriers and various engineered barrier system (EBS) components included on the Q-list. However, the drip shield and waste package FEPs do not qualify as "Q-list" items. The evaluations and conclusions do not directly impact engineered features important to safety, as defined in AP-2.22Q, *Classification Analyses and Maintenance of the Q-List.*

Approved quality assurance procedures identified in the technical work plan (BSC 2004 [DIRS 171583], Section 4) have been used to conduct and document the activities described in this document. In particular, this work constitutes an analysis report, and the documentation has been prepared according to AP-SIII.9Q, *Scientific Analyses,* and in accordance with related procedures and guidance documents as outlined in the technical work plan (BSC 2004 [DIRS 171583]). The technical work plan also identifies applicable controls for the electronic management of data (BSC 2004 [DIRS 171583], Section 8) during the analysis and documentation activities.

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3. USE OF SOFTWARE

This report uses no computational software; therefore, it 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 for traceability and transparency purposes, but were not used in the analyses discussed herein.

This report was developed using only commercial off-the-shelf software, Microsoft Word 2000, for word processing that is exempt from qualification requirements in accordance with LP-SI.11Q-BSC, *Software Management*. No additional applications (routines or macros) were developed using this commercial off-the-shelf software.

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4. INPUTS

AP-3.15Q, *Managing Technical Product Inputs*, categorizes technical product input as 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 Section 4.1 and indirect inputs are addressed in Section 6.1.3.

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 in 10 CFR Part 63 [DIRS 156605] and the YMRP. Section 4.3 identifies applicable codes and standards.

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 drip shield and waste package FEPs for screening in this report. The TSPA-LA FEP list identifies a FEP report or a set of sharing FEP 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 6-1).

Other direct inputs used for the FEP screening analysis are listed in Table 4-1.

Table 4-1. Direct Inputs

Technical Product Input	Specifically Used From	Specifically Used In	Input Description
ASM International 1990 [DIRS 144385]	p. 626	Table 4-1, Section 6.2.25	Handbook of metal properties
ASM International 1987 [DIRS 103753]	pp. 169, 557, 650-652	Table 4-1, Section 6.2.31	HIC of WPOB
	p.557	Table 4-1, Sections 6.2.8, 6.2.12, 6.2.27	Corrosion potential of Alloy 22, 316 Stainless Steel, and other drift materials
	pp. 169, 557, 650-652	Table 4-1, Section 6.2.8	HIC discussion. Used as input to screening argument.
	p. 681	Table 4-1, Section 6.2.21	Oxygen embrittlement of titanium
Boyer and Gall 1984 [DIRS 155318]	Section 32	Table 4-1, Section 6.2.24	Creep temperature of nickel-based alloys and titanium
BSC 2001 [DIRS 154441]	Entire	Table 4-1, Section 6.2.17	Copper in use in gantry rail system
BSC 2001 [DIRS 156807]	Section 6, Table 5-13	Table 4-1, Section 6.2.19	Crack plugging
BSC 2001 [DIRS 152655]	Section 5.3	Table 4-1, Section 6.2.30	Waste package outer barrier stress due to thermal expansion with various barrier gap sizes
BSC 2004 [DIRS 171924]	Section 8	Table 4-1, Section 6.2.28	Conclusions about aging and phase stability in the repository
BSC 2004 [DIRS 170024]	Sections 6.2.8, 6.3, 6.3.3, 6.4, 6.4.2, 6.4.7, 6.4.8, 7 (Table 22)	Table 4-1, Sections 6.2.1, 6.2.15, 6.2.16	Early failure summary for waste package and drip shield
BSC 2004 [DIRS 169766]	Table 23	Table 4-1, Section 6.2.22	Commercial SNF waste package design
	Table 1	Table 4-1, Section 6.2.1	WP spacing
	Entire	Table 4-1, Section 6.2.20	LA waste package designs
BSC 2004 [DIRS 169480]	Entire	Table 4-1, Section 6.2.30	Radial gap between the waste package inner vessel and the waste package outer barrier
BSC 2004 [DIRS 170803]	Sections 3, 4.2.2	Table 4-1, Section 6.2.12	Helium backfilling of waste packages
BSC 2004 [DIRS 169593]	Sections 5.1.4, 6.3, 6.4.1, 6.4.2, Tables 6.4-1, 6.4-3, Figure 6.4-1	Table 4-1, Sections 6.2.32, 6.2.33	Gamma dose rates to waste package materials
BSC 2004 [DIRS 166879]	Sections 7.2.1, 7.2.2	Table 4-1, Sections 6.2.14, 6.2.17, 6.2.23, 6.2.27	Deflection of the drip shield due to rockfall
BSC 2004 [DIRS 169868]	Section 6.3.3	Table 4-1, Section 6.2.18	SCC crack opening
BSC 2004 [DIRS 169845]	Sections 6.4.1, 6.5.8, 6.6.3, 6.6.4, 6.6.5, 8.2, 8.4, Tables 16 and 17	Table 4-1, Sections 6.2.1, 6.2.3, 6.2.7	Corrosion of drip shield
	Sections 6.5.3, 6.6.3, 6.7.2	Table 4-1, Section 6.2.11, 6.2.29	Corrosion of drip shield

Table 4-1. Direct Inputs (Continued)

Technical Product Input	Specifically Used From	Specifically Used In	Input Description
BSC 2004 [DIRS 169984]	Section 6.4.2	Table 4-1, Sections 6.2.2, 6.2.13	Dry air oxidation
	Sections 6.4.5, 8.2	Table 4-1, Sections 6.2.2, 6.2.10	Effect of microbial activity on corrosion
	Section 6.4.3	Table 4-1, Section 6.2.2	Aqueous corrosion rates
	Sections 6.4.4, 7.2.4, 8.3, Table 7-4	Table 4-1, Section 6.2.6	Discussion of localized corrosion models
BSC 2004 [DIRS 169847]	Sections 6.1.2, 6.1.3, 6.1.4 6.1.5, 6.1.6, 6.3.2, 8, 8.1	Table 4-1, Sections 6.2.9, 6.2.17, 6.2.31	HIC of drip shield
BSC 2004 [DIRS 169565]	Figure 6.3-56	Table 4-1, Sections 6.2.8, 6.2.21, 6.2.24, 6.2.25	Temperature history of waste packages and drip shields
BSC 2004 [DIRS 169985]	Sections 6.3.7, 6.4.2, 6.5.2	Table 4-1, Sections 6.2.5, 6.2.18, 6.2.19, 6.2.23, 6.2.26	Stress corrosion cracking of the drip shield, the waste package outer barrier
	Sections 6, 8.3, Tables 8-1 thru 8-3	Table 4-1, Section 6.2.4	Discussion of stress corrosion cracking inputs to TSPA-LA
BSC 2004 [DIRS 168489]	Table 1	Table 4-1, Section 6.2.1	WP spacing
	Entire	Table 4-1, Section 6.2.20	LA waste package designs
BSC 2004 [DIRS 169996]	Sections 6.3.2, 6.4.3	Table 4-1, Section 6.2.20	Nominal waste package configurations analyzed
CRWMS M&O 2000 [DIRS 150823]	Section 6.3	Table 4-1, Section 6.2.13	Design analysis for the defense high-level waste disposal container
CRWMS M&O 2000 [DIRS 123881]	p. II-10	Table 4-1, Section 6.2.12	Fuel drying
Etherington 1958 [DIRS 164789]	p. 10-107	Table 4-1, Section 6.2.33	Neutron fluence level thresholds for changes in mechanical properties
Haynes International 1988 [DIRS 101995]	Entire	Table 4-1, Section 6.2.24	Melting temperature
LL021012712251.021 [DIRS 163112]	Entire	Table 4-1, Sections 6.2.2, 6.2.3, 6.2.7	Long term corrosion testing results
LL021105312251.023 [DIRS 161253]	Entire	Table 4-1, Section 6.2.4	SCC of WPOB and DS
MO0003RIB00073.000 [DIRS 152926]	Entire	Table 4-1, Section 6.2.29	Physical characteristics of Titanium Grade 7
MO0407SEPFEPLA.000 [DIRS 170760]	Entire	Table 4-1, Sections 1, 1.2, 4.1, 6.1.1	FEPs database and description
Revie et al. 2000 [DIRS 159370]	Chapter 47	Table 4-1, Section 6.2.11	Titanium alloy properties
Schutz and Thomas 1987 [DIRS 144302]	Entire	Table 4-1, Sections 6.2.9, 6.2.31	Hydrogen absorption conditions in titanium alloys

The regulations in 10 CFR Part 63 [DIRS 156605] also provide direct inputs to the FEP screening process by specifying characteristics, concepts, and definitions for screening FEPs. The regulatory definitions and clarifications of concepts pertaining to the reference biosphere, geologic setting, reasonably maximally exposed individual (RMEI), and human intrusion are explained in detail in Section 4.1.3 of *The Development of the TSPA-LA Features, Events, and Processes* (BSC 2004 [DIRS 168706]). The regulations require that the standards shown in Table 4-2 are to be used in addressing these concepts.

The definitions and concepts discussed in *The Development of the Total System Performance Assessment-License Application Features, Events, and Processes* (BSC 2004 [DIRS 168706], Section 4.1.3.3) indicate that the RMEI is located no closer than 18 km to the south in the direction of groundwater flow and over a contaminated groundwater plume (in accordance with 10 CFR 63.312 [DIRS 156605]), and that the limit of the controlled area is no greater than 5 km from the repository in any direction other than to the south (as specified in 10 CFR 63.302 [DIRS 156605]). The location of the RMEI and the associated distance from the repository is of primary interest in evaluating potential exposure risk due to potential releases at the repository. The location of the RMEI is also important for determining exposure and is part of the technical basis for included FEPs.

As discussed in *The Development of the Total System Performance Assessment-License Application Features, Events, and Processes* (BSC 2004 [DIRS 168706], Sections 4.1.3.1), the reference biosphere must be consistent with present knowledge of conditions in the region. Changes in the biosphere (other than climate) from conditions at the time of license application should not be projected.

As discussed in *The Development of the Total System Performance Assessment-License Application Features, Events, and Processes* (BSC 2004 [DIRS 168706], Sections 4.1.3.2), the geologic setting (geology, hydrology, and climate) may evolve based upon cautious, but reasonable assumptions, consistent with present knowledge of factors that could affect the system in the 10,000 years following permanent repository closure.

By NRC’s combining of the geologic and hydrologic factors within the section addressing required characteristics of the reference biosphere, it is inferred that the listed regulatory constraint of changes in the reference biosphere are also applicable to conditions at Yucca Mountain. This approach agrees with the statement from 10 CFR 63.102(i) [DIRS 156605] that:

> Characteristics of the reference biosphere and the reasonably maximally exposed individual are to be based on current human behavior and biospheric conditions in the region, as described in §63.305 and §63.312.

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]).

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* (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 the *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 *Yucca Mountain Review Plan, Final Report* (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 the regulatory requirements 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 of Scenario Analysis and Event Probability is described in *Yucca Mountain Review Plan, Final Report* (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 model report 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 YMRP (NRC 2003 [DIRS 163274]) pertaining to identification of events with a 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] Regulatory Citation	Canori and Leitner 2003 [DIRS 166275] Associated PRD	Associated Criteria in the YMRP [DIRS 163274]
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

NOTE: 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 Part 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, Final Report* (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

This document was prepared to comply with NRC regulatory requirements presented in 10 CFR Part 63 [DIRS 156605]. Subparts of this rule that are applicable to date include Subpart B, Section 15 (Site Characterization), Subpart E, Section 114 (Requirements for Performance Assessment), Subpart F (Performance Confirmation Program) and Subpart G (Quality Assurance).

5. ASSUMPTIONS

This section addresses assumptions used in the FEP screening for the drip shield and waste package.

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]), which indicates that assuming the behavior of the earth is generally Poissonian or random is the underlying assumption in all probabilistic hazard analyses.

Although there may be cases where sufficient data and information exist to depart from this assumption, the Poissonian model is generally a reasonable 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 period (a relatively short time 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 in Sections 6.2.4.8 (FEP 1.5.03.01.0A) and 6.2.4.10 (FEP 1.5.01.01.0A).

5.2 NATURALLY OCCURRING EVENTS

Assumption: It is assumed that potential naturally occurring events, but perhaps of different magnitudes, 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 repository 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 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. Because it is consistent with the regulations, no further confirmation is necessary.

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

These types of events are known to occur. However, the effects of the phenomenon or the effects associated with varying magnitudes of the event type and probabilities are not well documented (e.g., effects of a supernova); the form of the coupling process is not well defined (e.g., changes in the earth's magnetic field); or the phenomenon has been shown to have insignificant or no impact presently (e.g., earth tides).

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. Deviations from design will be detected 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 the License Application 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. Therefore, no further confirmation of the assumption is required

Engineering and design changes are subject to evaluation to determine if there are any adverse manner impacts to nuclear safety as required in 10 CFR 63.73 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 requirements indicate that following the issuance of any license to construct, the DOE must submit periodic and special reports to the NRC regarding:

1. Progress of construction

2. Any data about the site, obtained during construction, 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.

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. This assumption is used throughout this report.

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6. SCIENTIFIC ANALYSIS DISCUSSION

The primary concern in this report is to address and document the screening decisions for the 33 waste package and drip shield degradation FEPs listed in Table 1-1. In some cases, where a FEP covers multiple technical areas shared with other FEP reports, this document provides only partial technical basis for the screening. The shared FEPs and the appropriate area of responsibility are identified in Table 1-1. The full technical basis for shared FEPs is addressed collectively by the sharing FEP reports.

6.1 METHODS AND APPROACH

The identification and screening of a comprehensive list of FEPs potentially relevant to the repository post closure performance is 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 a version of the LA FEP list (DTN: MO0407SEPFEPLA.000 [DIRS 170760]), used as initial input in this waste package and drip shield degradation 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 6-1).

6.1.2 Feature, Event, 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 4.2.2), 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 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 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 by regulation.

A FEP need only satisfy one of the exclusion screening criteria to be excluded from analysis in TSPA-LA. A FEP that does not satisfy any of the exclusion screening criteria must be included (screened in) in the TSPA-LA model. For many of the FEPs addressed in this report, it was determined that the probability of a consequential event or 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.

This report documents the screening decisions for the drip shield and waste package FEPs. In cases where a FEP covers multiple technical areas and is shared with other FEP reports, this document provides only a partial technical basis for the screening decision as it relates to drip shield and waste package issues.

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 two 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 analysis in TSPA-LA. For included FEPs, it is indicated as not applicable "N/A."

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 analysis, making reference to more detailed documentation in other supporting technical reports, as applicable. For excluded FEPs, it is indicated as not applicable "N/A."

Supporting Reports: This field is only used for included FEPs. It provides the list of supporting technical reports prepared by the Project (i.e., not all references are part of the supporting reports) that identify the FEP as an included FEP and contain information relevant to the implementation of the FEP within the TSPA-LA. This list of supporting technical reports provides traceability of the FEP through project documents. 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 input is used to provide additional information that is not used in the development of results or conclusions. Indirect inputs are listed in Table 6-1.

Table 6-1. Indirect Inputs

Technical Product Input	Specifically Used From	Specifically Used In	Input Description
10 CFR 63 [DIRS 156605]	Entire	Sections 1, 1.3, 4, 4.1, 4.2.1, 4.2.2, 4.3, 5.2, 5.3, 6.1.2, 6.1.6, 7.1, Tables 4-2, 6-1	NRC rule for Yucca Mountain Repository
ASM International 1987 [DIRS 103753]	pp. 80-82	Table 6-1, Sections 6.2.2, 6.2.4	Evidence for existence of a relative humidity threshold.
	pp. 971-974	Table 6-1, Section 6.2.33	Radiation damage discussion. Used as input to screening argument.
ASME 1995 [DIRS 141257]	Entire	Table 6-1, Section 6.2.13	Allowable tensile strength
Asphahani 1978 [DIRS 160352]	Entire	Table 6-1, Section 6.2.8	Information on HIC of nickel alloys
Beavers et al. 2002 [DIRS 158781]	Section 3.10	Table 6-1, Section 6.2.33	Possibility of radiation damage
Brossia 2001 [DIRS 159836]	Section 4.1.3	Table 6-1, Section 6.2.11	Corrosion of titanium alloys
BSC 2001 [DIRS 154441]	Section 6.3.2	Table 6-1, Section 6.2.17	Copper in gantry rail system
BSC 2001 [DIRS 152248]	Section 6.3.3	Table 6-1, Section 6.2.19	Crack sizes
BSC 2001 [DIRS 154004]	Entire	Table 6-1, Section 6.2.30	Effects of thermal expansion on WP
BSC 2001 [DIRS 152655]	Section 1, 5.3, Entire	Table 6-1, Section 6.2.30	Thermal expansion effects on WPOB
BSC 2003 [DIRS 161519]	Entire	Table 6-1, Section 6.2.30	Interlocking drip shield design
BSC 2004 [DIRS 171924]	Entire	Table 6-1, Sections 6.2.1, 6.2.28	Aging and phase stability of Alloy 22
BSC 2004 [DIRS 168030]	Entire	Section 5.1	Framework for seismicity and structural deformation at Yucca Mountain
BSC 2004 [DIRS 170019]	Entire	Table 6-1, Section 6.2.26	FEPs shared with clad degradation
BSC 2004 [DIRS 169766]	Entire	Table 6-1, Section 6.2.22	General reference
BSC 2004 [DIRS 168489]	Entire	Sections 1, 6.2.1, 6.2.9, 6.2.26, 7, Table 6-1	LA Design
	Tables 5 and 6	Table 6-1, Sections 6.2.17, 6.2.27	Design information for Alloy 22 feet and pallet
BSC 2004 [DIRS 169480]	Entire	Table 6-1, Sections 6.2.26, 6.2.30	Typical waste package components assembly
BSC 2004 [DIRS 169472]	Entire	Table 6-1, Sections 6.2.12, 6.2.30	Typical waste package components assembly
BSC 2004 [DIRS 166879]	Section 7.2.2	Table 6-1, Sections 6.2.8, 6.2.17, 6.2.18, 6.2.22, 6.2.27	Function of emplacement pallet
BSC 2004 [DIRS 169898]	Entire	Table 6-1, Sections 6.2.1, 6.2.4, 6.2.5, 6.2.13, 6.2.14, 6.2.15, 6.2.16, 6.2.20, 6.2.23, 6.2.25, 6.2.32	FEPs shared with EBS

Table 6-1. Indirect Inputs (Continued)

Technical Product Input	Specifically Used From	Specifically Used In	Input Description
BSC 2004 [DIRS 170017]	Entire	Section 1.2	General reference
BSC 2004 [DIRS 169845]	Entire	Table 6-1, Sections 6.2.3, 6.2.7, 6.2.9, 6.2.20, 6.2.22, 6.2.29, 6.1.31, 6.2.32	Corrosion of drip shield
BSC 2004 [DIRS 169984]	Section 6.4.6	Table 6-1, Sections 6.2.1, 6.2.2, 6.2.28	Effect of aging and phase instability
	Entire	Table 6-1, Sections 6.2.2, 6.2.6, 6.2.13, 6.2.20, 6.2.22, 6.2.28	Corrosion of waste package outer barrier
	Sections 6.4.3, 6.4.5, Table 6-1, Entire	Table 6-1, Sections 6.2.10, 6.2.11	MIC of waste package outer barrier
	Section 6	Table 6-1, Section 6.2.31	Corrosion of WPOB
BSC 2004 [DIRS 169847]	Entire	Table 6-1, Section 6.2.17	HIC of drip shield
	Sections 6.1.2-6.1.5, 6.3.2	Table 6-1, Section 6.2.31	HIC of drip shield
	Sections 6.1.3, 6.1.6	Table 6-1, Section 6.2.17	HIC of drip shield
BSC 2004 [DIRS 169565]	Entire	Table 6-1, Section 6.2.13	Temperature table
BSC 2004 [DIRS 168556]	Entire	Table 6-1, Section 6.2.1	FEPs that are shared or referenced from criticality
BSC 2004 [DIRS 169985]	Sections 6.3, 6.3.4	Table 6-1, Section 6.2.4	Stress corrosion cracking of the drip, slip dissolution-film rupture model shield, the waste package outer barrier
	Sections 6.2.1, 6.4.2	Table 6-1, Sections 6.2.4, 6.2.22	Stress corrosion cracking of the drip shield, the waste package outer barrier
	Section 8.3	Table 6-1, Section 6.2.22	Contact stresses
	Entire	Table 6-1, Section 6.2.15	Manufacturing defects (weld flaws) on waste package
BSC 2004 [DIRS 170265]	Entire	Table 6-1, Sections 6.2.13, 6.2.14, 6.2.23, 6.2.25	Subsurface facility description document
BSC 2004 [DIRS 171583]	Entire	Sections 1.1, 2, Table 6-1	Technical work plan for regulatory integration modeling and analysis of the waste form and waste package
	Table 3-1	Section 4.2	Yucca Mountain Review Plan acceptance criteria
BSC 2004 [DIRS 168706]	Entire	Sections 1.2, 4.1, 6.1, 6.1.1, 6.1.2, 6.1.6, Table 6-1	The development of the total system performance assessment license application features, events, and processes
BSC 2004 [DIRS 169996]	Entire	Table 6-1, Sections 6.2.2, 6.2.3, 6.2.4, 6.2.10, 6.2.15, 6.2.20	General reference to AMR
	Section 6.5	Table 6-1, Section 6.2.2	General reference to number of WPs and patches
BSC 2004 [DIRS 170020]	Entire	Table 6-1, Sections 6.2.32, 6.2.33	FEPs shared with waste form

Table 6-1. Indirect Inputs (Continued)

Technical Product Input	Specifically Used From	Specifically Used In	Input Description
Canori and Leitner 2003 [DIRS 166275]	PRD-013/T-036, PRD-013/T-045	Table 6-1, Section 6.2.12	Project Requirements Document
	Entire	Table 4-2, Sections 4.2, 4.2.1	Project Requirements Document
Cowan and Weintritt 1976 [DIRS 105212]	pp. 1 to 39 and 376 to 383	Table 6-1, Section 6.2.19	Formation of deposits in seawater
CRWMS M&O 2000 [DIRS 150823]	Table 5	Table 6-1, Section 6.2.12	Design analysis for the defense high-level waste disposal container
CRWMS M&O 2001 [DIRS 152016]	Entire	Table 6-1, Section 6.2.19	Water distribution and removal characteristics
Dunn and Brossia 2002 [DIRS 162213]	Entire	Table 6-1, Section 6.2.6	Discussion of localized corrosion
Gdowski 1991 [DIRS 100859]	Sections 5, 5.1	Table 6-1, Section 6.2.8	Degradation modes for Ni-Cr-Mo alloys
Gdowski 1997 [DIRS 102789]	p.1-8	Table 6-1, Section 6.2.29	Pd concentration in Titanium Grade 7
Hua and Gordon 2003 [DIRS 163111]	Entire	Table 6-1, Section 6.2.29	Corrosion of titanium in basic saturated water
Hua et al. 2002 [DIRS 160670]	Entire	Table 6-1, Section 6.2.29	Corrosion testing of container materials
Kohli and Pasupathi 1986 [DIRS 131519]	Entire	Table 6-1, Section 6.2.12	Investigation of water-logged spent fuel rods under dry storage conditions
Little and Wagner 1996 [DIRS 131533]	Entire	Table 6-1, Section 6.2.11	MIC of titanium alloy
Lorenzo de Mele and Cortizo 2000 [DIRS 159833]	Entire	Table 6-1, Section 6.2.3	Fluoride effects on titanium
Merriam-Webster 1993 [DIRS 100468]	Entire	Table 4-1, Section 5.1	Definitions
NRC 2003 [DIRS 163274]	Entire	Tables 4-2, 6-1, Sections 4.2, 4.2.1, 4.2.2, 6.1, 6.1.1, 6.1.2, 7.1.1	Yucca Mountain Review Plan
Pan et al. 2002 [DIRS 165536]	Entire	Table 6-1, Section 6.2.4	Stress corrosion cracking and hydrogen embrittlement of container and drip shield materials
Plinski 2001 [DIRS 156800]	Section 1	Table 6-1, Section 6.2.20	Waste package design
	Section 8.1.7	Table 6-1, Sections 6.2.4, 6.2.28	Annealing of outer cylinder

Table 6-1. Indirect Inputs (Continued)

Technical Product Input	Specifically Used From	Specifically Used In	Input Description
	Section 8.3.17	Table 6-1, Section 6.2.16	Drip shield assembly
	Section 8.1.8	Table 6-1, Sections 6.2.28, 6.2.30	Waste package design
Pulvirenti et al. 2002 [DIRS 165537]	Entire	Table 6-1, Section 6.2.4	Effects of lead, mercury, and reduced sulfur species on the corrosion of Alloy 22 in concentrated groundwaters as a function of pH and temperature
Sakai et al. 1992 [DIRS 154465]	Entire	Table 6-1, Section 6.2.4	Effect of lead water chemistry on oxide thin film of Alloy 600
Shoesmith and Ikeda 1997 [DIRS 151179]	Sections 3, 6, Entire	Table 6-1, Section 6.2.11	MIC of titanium alloys
Shoesmith and King 1998 [DIRS 112178]	pp. 5, 29 to 30	Table 6-1, Section 6.2.32	Effects of gamma irradiation on waste package materials
Shoesmith et al. 1995 [DIRS 117892]	Entire	Table 6-1, Section 6.2.11	Initiation of biofilms under crevice corrosion
Thomas 1994 [DIRS 120498]	Entire, p. 17:54	Table 6-1, Section 6.2.6	Discussion of localized corrosion

6.1.4 Qualification of Unqualified Direct Inputs

N/A. No unqualified direct inputs are used.

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 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 nuclear safety as required in 10 CFR 63.73 Subparts F and G [DIRS 156605]. See also the requirements in 10 CFR 63.44 and 10 CFR 63.131 [DIRS 156605].

6.2 WASTE PACKAGE AND DRIP SHIELD FEATURE, EVENT, AND PROCESSES SCREENING AND ANALYSES

Screening information for each of the 33 waste package and drip shield FEPs is presented in separate subsections. The FEPs are addressed in numeric order.

6.2.1 Error in Waste Emplacement (1.1.03.01.0A)

FEP Description:

Deviations from the design and/or errors in waste emplacement could affect long-term performance of the repository. A specific example of such an error would be erroneously emplacing the waste packages in a saturated or wet zone of the repository. Errors of this type would impact repository performance by affecting waste package corrosion and radionuclide transport.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

The features, events, or processes that could result in an error in waste emplacement that could lead to unanticipated operating conditions stem from administrative controls. The results of these errors could affect the waste package surface temperature and humidity history (thus impacting corrosion rates), result in placement in prohibited areas, or allow water to contact the waste package at times earlier than expected.

Accidental misloading of waste packages with fuel assemblies having a lower burnup than required by the appropriate loading curve could cause the waste package surface temperature and relative humidity to be outside the expected ranges and impact the degradation characteristics of the waste package and drip shield. *Analysis of Mechanisms for Early Waste Package Failure* (BSC 2004 [DIRS 170024], Section 6.2.8) states that the temperature at the surface of the waste package is mainly governed by the temperature in the drift where the waste package is located, rather than the heat output within the waste package (a metallic container with a rather large heat transfer area). Therefore, the increase in heat output generated by a thermally overloaded waste package (which can not be expected to exceed a few kilowatts) would be quickly dissipated into the drift and is not expected to alter the waste package surface temperature to an extent significant enough to affect post closure performance.

Similarly, emplacement of waste packages closer than the design specifies could cause the waste package surface temperature and relative humidity to be outside the expected ranges and impact the degradation characteristics of the waste package and drip shield. The current emplacement drift design specifies that the average waste package skirt-to-skirt spacing is 0.1 m or 10 cm (BSC 2004 [DIRS 168489], Table 1) indicating that any longitudinal misplacements would be small (on the order of 10 cm). For each locally created hot zone (resulting from placing a waste package closer than intended to one of its neighbors) there necessarily would be a corresponding cooler zone (resulting from the waste package being further than intended from its other neighbor). It is reasonable to conclude that, on a larger scale, the minute variations in local temperature and relative humidity caused by longitudinal misplacement of waste packages have low consequence to waste package or drip shield degradation. This conclusion is consistent with the discussion above regarding accidental misloadings (i.e., the temperature at the surface of the waste package is mainly governed by the temperature in the drift where the waste package is located, rather than the heat output within the waste package).

In addition, no effects of aging and phase instability on the corrosion behavior of the Alloy 22 waste package outer barrier would be observed even if the waste package were maintained at less than 300ºC for a period of 500 years followed by temperatures less than 200°C for a period of 9,500 years (BSC 2004 [DIRS 171924], Section 8; BSC 2004 [DIRS 169984], Section 6.4.6). These thermal exposure conditions bound all repository-relevant thermal exposure conditions (BSC 2004 [DIRS 171924], Section 8).

Another possible consequence of accidental misloading of waste packages with fuel assemblies having a lower burnup than required by the appropriate loading curve is the increased potential for criticality. Treatment of accidental misloading of lower burnup fuel assemblies is discussed in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (BSC 2004 [DIRS 168556]).

The areas that would be restricted from waste package emplacement would be those where the waste package would straddle a fault. An error in placement then would be defined as a waste package emplaced so that it straddles a fault. The evaluation of the event is discussed in FEP 1.2.02.03.0A in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

The current engineered barrier design includes a titanium drip shield that is placed over the waste packages (BSC 2004 [DIRS 168489]). The drip shield is installed in segments and, once installed, continues along the entire length of the drift. Each segment will slightly overlap the previously emplaced segment. Human error during placement of the drip shield resulting in a gap between the drip shield segments, could lead to water contact or rockfalls on the waste package at a time earlier than would be expected under nominal in-drift conditions. The gap between two adjacent drip shields improperly interlocked is expected to be small. It is not credible that the inspection performed by remotely operated cameras would not detect an incorrect emplacement leaving a gap exceeding the length of the connecting plates of the drip shields (BSC 2004 [DIRS 170024], Section 6.4.7). Therefore, dripping water and rockfalls are not expected to fall directly onto the underlying waste package, but will hit the connecting plate and not the waste package surface (BSC 2004 [DIRS 170024], Section 6.4.7).

Similarly, erroneous emplacement of waste packages in a saturated or wet zone of the repository will not cause significant damage to the waste package outer barrier because the drip shields must fail for water to contact the underlying waste package. As discussed in FEP 2.1.03.02.0B, Stress Corrosion Cracking (SCC) of Drip Shields, SCC of drip shields does not compromise the water diversion function of the drip shield (i.e., SCC of the drip shield does not lead to water contacting the underlying waste package). As discussed in FEP 2.1.03.03.0B, Localized Corrosion of Drip Shields, localized corrosion of the drip shield will not occur under exposure conditions in the repository. Therefore, general corrosion is the only possible drip shield failure mode of any consequence to repository performance (note that seismic effects are not considered in this section). As shown in *General Corrosion and Localized Corrosion of Drip Shield* (BSC 2004 [DIRS 169845], Tables 16 and 17) the maximum general corrosion rate that can be applied to the underside of drip shield is about 1.13×10^{-4} mm/yr and the maximum general corrosion rate that can be applied to the top side of drip shield is about 3.20×10^{-4} mm/yr. Therefore, the earliest time at which the 15-mm-thick drip shield can fail as a result of general corrosion is approximately 35,000 years after final closure of the repository. With no water expected to drip on the waste package, there is no possibility of any effect of erroneous emplacement of waste packages in a saturated or wet zone of the repository on waste package degradation.

Based on the above discussion, this FEP is excluded based on low consequence to radiological exposures to the reasonable maximally exposed individual and radionuclide releases to the accessible environment.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.2 General Corrosion of Waste Packages (2.1.03.01.0A)

FEP Description:
General corrosion may contribute to waste package failure.

Screening Decision:
Included

Screening Argument:
N/A

TSPA Disposition:
General corrosion (or passive corrosion) is the uniform thinning of the waste package outer barrier at its open-circuit corrosion potential. The presence of stable "liquid" water is required to initiate corrosion processes (including general corrosion) that are supported by electrochemical corrosion reactions. Typically, a threshold relative humidity is used to simulate such a corrosion initiation condition (e.g., ASM International 1987 [DIRS 103753], pp. 80-82). Surface cleanliness, corrosion product build-ups, and/or deliquescence of hygroscopic salts can cause water film formation on surfaces at lower relative humidities than would be required on smooth clean surfaces (ASM International 1987 [DIRS 103753], p. 80-82). Conservatively, in the waste package degradation analysis for TSPA-LA, no threshold relative humidity is used (BSC 2004 [DIRS 169996], Section 5.1) (i.e., general corrosion can occur under any exposure conditions). Because general corrosion is likely to be operative for most of the repository operation period, it could lead to degradation and breach of waste packages in the repository. General corrosion due to dry-air oxidation and aqueous corrosion as well as the effects of microbially influenced corrosion and aging and phase instability on general corrosion of the waste package outer barrier are discussed in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984]).

General Corrosion and Localized Corrosion of the Waste Package Outer Barrier (BSC 2004 [DIRS 169984], Section 6.4.2) concluded that although dry air oxidation occurs, it results in a negligible amount of barrier thinning over repository time scales (only ~93 μm even if the waste package outer barrier were exposed for 10,000 years at 350°C). Therefore, dry oxidation is not considered in TSPA-LA analyses.

Penetration rates for general corrosion are provided in *General Corrosion and Localized Corrosion of the Waste Package Outer Barrier* (BSC 2004 [DIRS 169984] Section 6.4.3). General corrosion rates of the waste package outer barrier were estimated using the weight-loss of Alloy 22 crevice geometry specimens after a 5-year exposure in the LTCTF and the temperature dependence of corrosion rates measured using the polarization resistance technique (BSC 2004 [DIRS 169984], Section 6.4.3). The temperature dependence follows an Arrhenius relationship, as discussed in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984], Section 6.4.3.4).

The patch size used to analyze general corrosion of the waste packages in *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (BSC 2004 [DIRS 169996], Section 6.4.3) is four times the area of the crevice geometry specimen size used to evaluate the Alloy 22 general corrosion rates in *General Corrosion and Localized Corrosion of the Waste Package Outer Barrier* (BSC 2004 [DIRS 169984] Section 6.4.3). Therefore, the general corrosion rates are adjusted to account for the effects of this change of scale. Conceptually, the scaling method employed corresponds to using the highest of four sampled corrosion rates (from the crevice geometry specimen-based distribution) for the general corrosion rate applied to the waste package patch. The approach is conservative because it effectively uses the highest of the four corrosion rates sampled. The effect of this method is to shift the median general corrosion rate to higher values and to decrease the probability of sampling lower general corrosion rates. This general corrosion treatment applies to commercial spent nuclear fuel and codisposed waste packages.

The effects of trace metals, including arsenic, calcium, and magnesium, are included in the general corrosion rate distributions based on samples exposed in the LTCTF. Samples of Alloy 22 were immersed in SAW for five years (DTN: LL021012712251.021 [DIRS 163112]). After five years, a sample of the acidified water, including any corrosion products, was analyzed for trace metals using inductively coupled plasma and mass spectroscopy methods. Low concentrations of potentially deleterious metals, including arsenic, calcium, and magnesium were detected. Given that the corrosion rates used for the Alloy 22 waste package outer barrier are based on samples exposed in these environments, the general corrosion analyses account for the presence of these trace metals both in the solutions and in the materials used for testing. On this basis, the effects of trace metals on general corrosion of the waste package outer barrier are included in TSPA-LA although the effects are not explicitly and separately modeled.

The effect of microbial activity on the general corrosion process of the waste package outer barrier is represented in TSPA-LA analyses with a general corrosion rate-enhancement factor (BSC 2004 [DIRS 169984], Section 8.2). A more detailed discussion on the effect of microbial activity is provided in FEP 2.1.03.05.0A, Microbially Influenced Corrosion (MIC) of Waste Packages.

Comparative analysis of the corrosion rates from the polarization resistance technique showed insignificant effects of welding and thermal aging of the waste package outer barrier on the general corrosion rates. It was determined that the aging of the base metal and welds of the waste package outer barrier under the thermal conditions expected in the repository would not affect the corrosion performance of the waste package outer barrier and will not be specifically modeled in the TSPA-LA (BSC 2004 [DIRS 169984], Section 6.4.6).

The general corrosion rate distributions along with the exposure condition parameters for the waste packages and drip shields are incorporated into the integrated waste package degradation analysis (BSC 2004 [DIRS 169996]). The output from the integrated waste package degradation analysis is a set of profiles (time histories) for the failure (i.e., initial breach) and subsequent number of penetration openings in the waste package and drip shield as a function of time. The integrated waste package degradation analysis is used directly in the TSPA-LA.

The TSPA-LA waste package degradation analysis simulates the behavior of a few hundred waste packages (BSC 2004 [DIRS 169996], Section 6.5). Effects of spatial and temporal variations in the exposure conditions over the repository are modeled by explicitly incorporating relevant exposure condition histories into the analysis. The exposure condition parameters considered to vary over the repository are relative humidity and temperature at the waste package surface. Potentially variable corrosion processes for a single waste package are represented by dividing the waste package surface into subareas called "patches" and stochastically sampling the degradation parameter values for each patch. The use of patches explicitly represents the variability in degradation processes associated with a single waste package at a given time.

In the TSPA-LA analysis, uncertainty in waste package degradation is analyzed with multiple realizations of the integrated waste package degradation analysis. For each realization, values are sampled for the uncertain degradation parameters and passed to the integrated waste package degradation analysis. Each realization is a complete integrated waste package degradation analysis simulation of a given number of waste packages, explicitly considering variability in the degradation processes. Accordingly, each of the integrated waste package degradation analysis outputs (i.e., the fraction of the total number of waste packages and drip shields failed versus time and of the average number of patch and crack penetrations per failed waste package (or drip shield)) are reported as a group of "degradation profile curves" (resulting from the multiple realizations) that represent the potential range of the output parameters. For example, the waste package breach time profiles are reported with a group of "curves" representing the cumulative probability of waste package breaches as a function of time. The outputs of the integrated waste package degradation analysis are used as input for waste form degradation analysis and radionuclide release analysis from failed waste packages conducted within the TSPA-LA model. The TSPA representation of general corrosion of waste package conservatively models the phenomenon studied and data generated in *General Corrosion and Localized Corrosion of the Waste Package Outer Barrier* (BSC 2004 [DIRS 169984]) and *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (BSC 2004 [DIRS 169996]).

Supporting Reports:

BSC 2004 [DIRS 169984]
BSC 2004 [DIRS 169996]

6.2.3 General Corrosion of Drip Shields (2.1.03.01.0B)

FEP Description:

General corrosion may contribute to drip shield failure.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

General corrosion is included in drip shield degradation analysis. This includes the effects of corrosive gases. General corrosion due to dry-air oxidation, humid-air and aqueous general corrosion, microbially influenced corrosion, and aging and phase instability of the Titanium Grade 7 drip shield are discussed in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845]).

It was concluded in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Section 8.2) that although dry air oxidation occurs, it results in a negligible amount of metal thinning over repository time scales (only ~1,320 nm even if the drip shield were exposed for 10,000 years at 200°C). Therefore, dry oxidation does not need to be considered in the TSPA-LA analyses.

Penetration rates for general corrosion are provided in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Tables 16 and 17) and are used in TSPA-LA analyses. Humid-air and aqueous corrosion processes are considered part of general corrosion (BSC 2004 [DIRS 169845], Sections 6.1, 6.3.2, and 6.3.3). General corrosion rates of the drip shield were estimated using weight-loss data of Titanium Grade 16 samples after a 1-year exposure in the LTCTF and validated using 5-year Titanium Grade 16 and 2.5-year Titanium Grade 7 exposure test data (BSC 2004 [DIRS 169845], Section 7.4.2). Although analysis of this data shows that the corrosion rate of the titanium alloys decreases with time, a time-independent general corrosion rate (sampled from a distribution developed using the 1-year Titanium Grade 16 exposure data) is used in TSPA-LA (BSC 2004 [DIRS 169845], Sections 7.4.2 and 8.3). The maximum corrosion rates measured for the 2.5-year exposed Titanium Grade 7 specimens are lower than those for the 5-year exposed Titanium Grade 16 specimens. This observation indicates that the general corrosion resistance of Titanium Grade 7 is superior to that of Titanium Grade 16 and that the general corrosion rate distribution developed using the 1-year Titanium Grade 16 exposure data is a conservative representation of drip shield general corrosion rates.

The drip shield outer surface may be exposed to a more complicated chemistry and geometry than the drip shield inner surface, since dust or mineral films (from evaporation of dripping water), or both, may form crevices on the drip shield outer surfaces. In contrast, the inner surfaces of the drip shield will not be exposed to dripping water or significant dust film formation (BSC 2004 [DIRS 169845], Sections 6.3, 6.5.2, and 6.5.3). Therefore, general corrosion of the inner and outer surfaces of the drip shield are modeled by using different sets of corrosion data (BSC 2004 [169845], Sections 6.3, 6.5.2, and 6.5.3). Variations in the drip shield general corrosion rate distributions are considered to be entirely due to uncertainty (BSC 2004 [DIRS 169845], Section 6.5.4.2). For each realization of the integrated waste package degradation analysis (BSC 2004 [DIRS 169996]), a single general corrosion rate is sampled from each general corrosion rate distribution (i.e., one general corrosion rate value is sampled from the general corrosion rate distribution applicable to the drip shield underside and another general corrosion rate value is sampled from the general corrosion rate distribution applicable to the drip shield outer surface). The two sampled values are then applied one each to the outer and inner surfaces of each drip shield simulated during the given realization. On each time step, general corrosion of the drip shield occurs. Using this conceptual model for drip shield general corrosion, all drip shields in the repository fail by general corrosion at the same time. The maximum general corrosion rate for the cumulative distribution function (CDF) applied to the under side of the drip shield is approximately 1.13×10^{-4} mm/yr and the maximum general corrosion rate for the CDF applied to the top side of the drip shield is approximately 3.2×10^{-4} mm/yr (BSC 2004 [DIRS 169845], Tables 16 and 17); therefore, drip shield failure is not anticipated to occur from general corrosion until at least 35,000 years after permanent closure.

As discussed in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Section 6.5.7), the presence of dissolved fluoride can, under certain conditions, increase the general corrosion rate of titanium alloys including Titanium Grade 7. Direct incorporation of fluoride into the passive film leads to enhanced dissolution only under acidic conditions. A combination of a significant F^- concentration (> 500 µg/g) and a low pH (< 4.3) is necessary for loss of passivity (BSC 2004 [DIRS 169845], Section 6.5.7). Lorenzo de Mele and Cortizo (2000 [DIRS 159833]) conducted measurements in synthetic saliva (pH = 6.5, containing 0.2 mol/L of fluoride), which showed that if fluoride were added shortly after electrode immersion (while the oxide film was still growing and defective), the corrosion potential fell rapidly (indicating attack of the oxide). However, if the oxide were allowed to grow for four days (a sufficient period for the growth of a coherent oxide with a low defect concentration), then addition of the same amount of fluoride had no observable effect over the subsequent two days of exposure indicating that the susceptibility of titanium to corrosion in fluoride solutions is associated with defects and flaws in the oxide.

In the case of the drip shield application in the repository, there is an early period of dry air exposure. During this time, the drip shield is subjected to a long period of thermal oxidation prior to aqueous exposure. Thermal treatments not only thicken the oxide film but also significantly decrease the defect density in the film compared to films grown in aqueous environments (BSC 2004 [DIRS 169845], Section 6.5.7). The increase in thickness and improvement in film properties would lead to a significant decrease in susceptibility to fluoride-induced film breakdown and an independence of the passive corrosion rate on fluoride concentration (BSC 2004 [DIRS 169845], Section 6.5.7). Therefore, fluoride will have a low consequence to the degradation rate of titanium in the repository.

The effects of other trace metals, including arsenic, calcium, and magnesium are included in the general corrosion rate distributions based on samples exposed in the LTCTF. Samples of Titanium Grade 16, an excellent analogue for the more corrosion-resistant Titanium Grade 7, were immersed in simulated acidified water (DTN: LL021012712251.021 [DIRS 163112]) in the LTCTF. Samples of the acidified water, including any corrosion products, were analyzed for trace metals using inductively coupled plasma–mass spectrometry methods. Low concentrations of potentially deleterious metals, including arsenic, calcium, and magnesium, were detected in the sampled waters. As the corrosion rates used for the Titanium Grade 7 drip shield are based on samples exposed in these environments, the general corrosion analyses account for the presence of these trace metals (BSC 2004 [DIRS 169845], Section 6.5.1). On this basis, the effects of trace metals on general corrosion of the drip shield are included in TSPA-LA although the effects are not explicitly and separately modeled.

The general corrosion rate distributions for the drip shields are incorporated into the integrated waste package degradation analysis (BSC 2004 [DIRS 169996]). The output from the integrated waste package degradation analysis is a set of profiles (time histories) for the failure (i.e., initial breach) and subsequent number of penetration openings in the waste package and drip shield as a function of time. The integrated waste package degradation analysis is used directly in the TSPA-LA.

Supporting Reports:

BSC 2004 [DIRS 169845]
BSC 2004 [DIRS 169996]

6.2.4 Stress Corrosion Cracking (SCC) of Waste Packages (2.1.03.02.0A)

FEP Description:

Waste packages may become wet at specific locations that are stressed leading to stress corrosion cracking (SCC). The possibility of SCC under dry conditions or due to thermal stresses are also addressed as part of this FEP.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

Stress corrosion cracking of the waste package outer barrier closure weld regions is included in TSPA-LA as part of waste package degradation analyses. *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985], Section 8.3, Tables 8.1 to 8.3) provides input to the TSPA-LA treatment of waste package degradation by stress corrosion cracking.

All regions of the waste package (including the fabrication welds), except the waste package closure lid welds, are stress relief annealed before the waste packages are loaded with waste (Plinski 2001 [DIRS 156800], Section 8.1.7), and thus do not develop residual stress/stress intensity factors high enough for SCC to occur (BSC 2004 [DIRS 169985], Section 6.4.2). Welds are the most susceptible to SCC because (1) welding can produce high tensile residual stress in the weld; (2) preexisting flaws due to fabrication and welding have much higher concentration in the weld than in the base metal; and (3) welding could result in segregation and nonequilibrium brittle phases, which could enhance material susceptibility to SCC (BSC 2004 [DIRS 169985], Section 6.4.2). Plastic deformation resulting from seismic events also has the potential of leading to plastic upsets and resultant sustained residual stresses that may initiate cracks and drive them through the wall. Seismic effects are discussed 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 in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

The presence of stable "liquid" water is required to initiate corrosion processes (including SCC) that are supported by electrochemical corrosion reactions. Typically, a threshold relative humidity is used to simulate such a corrosion initiation condition (e.g., ASM International 1987 [DIRS 103753], pp. 80 to 82). Conservatively, in the waste package degradation analysis for TSPA-LA, no threshold relative humidity is used (BSC 2004 [DIRS 169996], Section 5.1) (i.e., SCC is allowed to occur, if all stress and stress intensity criteria are satisfied, under any exposure conditions).

As discussed in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985], Section 6.3), the slip dissolution–film rupture model was used to assess the breach (or lack of it) of the waste package by SCC crack propagation due to the combined effects of stress and environment, or both.

The SCC model for the waste package closure weld regions uses weld residual stress profiles and stress intensity factor profiles (BSC 2004 [DIRS 169985], Section 6). These input data were developed for the 25-mm outer lid closure weld regions (subjected to laser peening) and the as-welded 10-mm middle lid closure weld regions. The report (BSC 2004 [DIRS 169985], Section 6) also provides other input for the degradation of the waste package due to SCC including: a threshold stress for crack initiation; a threshold stress intensity factor for propagation; weld flaw size, density and orientation distributions; and an estimate of crack-opening size. This SCC treatment applies to all waste packages.

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2004 [DIRS 169985], Section 6.2.1) describes the results of SCC crack-initiation measurements under constant load conditions while immersed in a basic saturated water solution. Alloy 22 (UNS N06022) exhibits excellent SCC resistance, as cracking was not observed for any of the 120 Alloy 22 (UNS N06022) specimens covering a variety of metallurgical conditions (including the as-welded condition). The applied stress ratios used in the experiments were up to about 2.1 times the yield strength of the as-received material and up to 2.0 times the yield strength of the welded material. This stress ratio corresponds to an applied stress of about 89 to 96 percent of the ultimate tensile strength. The high degree of SCC-initiation resistance for Alloy 22 (UNS N06022) is corroborated by results of high magnification visual examination of a number of Alloy 22 (UNS N06022) U-bend specimens exposed to a range of relevant environments at 60°C and 90°C in the LTCTF (BSC 2004 [DIRS 169985] Section 6.2.1). No evidence of SCC initiation has been observed in these U-bend specimens after five years of exposure.

The discussion in FEP 2.1.11.07.0A, Thermal Expansion/Stress of In-Drift EBS Components, indicates that thermal expansion is not a source of stress and, therefore, not a driving force for SCC in the repository.

The SCC model is incorporated into the integrated waste package degradation analysis (BSC 2004 [DIRS 169996], Section 6.4). The output from the integrated waste package degradation analysis is a set of profiles (time histories) for the failure (i.e., initial breach) and subsequent number of penetration openings in the waste package and drip shield as a function of time. The integrated waste package degradation analysis is used directly in the TSPA-LA. The TSPA representation of stress corrosion cracking of waste package conservatively models the phenomenon studied and data generated in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985]) and *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (BSC 2004 [DIRS 169996]).

Lead (Pb) has been identified as a potential contributor to stress corrosion cracking in nickel-based alloys (Sakai et al. 1992 [DIRS 154465]; Pan et al. 2002 [DIRS 165536]; Pulvirenti et al. 2002 [DIRS 165537]). The results achieved by Pulvirenti et al. (2002 [DIRS 165537]) have not been able to be reproduced by the same investigators or investigators from the Center for Nuclear Waste Regulatory Analyses (Pan et al. 2002 [DIRS 165536]). The effects of lead on SCC processes were investigated by the Project by evaluating the effect of significant additions of lead nitrate ($PbNO_3$) to test solutions. Slow strain-rate stress corrosion cracking experiments were performed on specimens of Alloy 22 (UNS N06022) in lead-containing solutions, as discussed in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985], Section 6.3.4). SCC initiation test results were obtained using Slow Strain Rate Tests, at 76°C to 95°C in low pH brine solutions (pH = ~3) with and without 0.005% lead nitrate additions. These results also show no effect of Pb on SCC susceptibility. Thus, there appears to be no basis for concern that Pb will affect SCC susceptibility in these relevant concentrated brine environments over a broad range of pH values. Other experiments showed that even if SCC is forced to occur (by slow cyclic straining) before lead is added to the test solution, SCC crack-growth rates are not accelerated by subsequent lead additions (DTN: LL02110531105312251.023 [DIRS 161253]). This latter experiment used a basic saturated water solution with a pH ~ 12 at 110°C. After 8,670 hours of exposure in the lead-free solution, 1,000 ppm Pb (as $PbNO_3$) was added to the test solution and SCC crack growth rates were monitored for ~1,800 hours using extremely sensitive techniques (in situ reversing DC-potential drop technique). The presence of the lead in the test solution was concluded to have no measurable effect on the SCC growth rate (DTN: LL021105312251.023 [161253]). On this basis, any lead present in repository groundwaters is expected to have little consequence on Alloy 22 (UNS N06022) SCC processes.

Supporting Reports:

BSC 2004 [169985]
BSC 2004 [169996]

6.2.5 Stress Corrosion Cracking (SCC) of Drip Shields (2.1.03.02.0B)

FEP Description:

Drip shields may become wet at specific locations that are stressed leading to stress corrosion cracking (SCC). The possibility of SCC under dry conditions or due to thermal stresses is also addressed as part of this FEP.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

The sources of stress that could result in stress corrosion cracking in the Titanium Grade 7 drip shield are (1) weld-induced residual stress, (2) plasticity-induced residual stress caused by seismic events, and (3) residual stress produced by rock falls (BSC 2004 [DIRS 169985], Section 6.3.7).

The weld induced residual stress in the drip shields will be mitigated by annealing before placement in the drifts. Therefore, drip shields are not subject to SCC upon emplacement (BSC 2004 [DIRS 169985], Section 6.3.7). However, the drip shields are subject to SCC under the action of seismic-induced loading and rockfall. Seismic effects on drip shield degradation are discussed in FEPs 2.1.07.01.0A, Rockfall, and 1.2.03.02.0A, Seismic Ground Motion Damages EBS Components, in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

The consequence of stress corrosion cracking on drip shield performance is discussed in FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield. FEP 2.1.03.10.0B is excluded on the basis of low consequence because the amount of advective water flow through cracks in the drip shield will be severely limited based on the small aperture width (narrow opening and tight cracks), the presence of capillary forces, and the potential for plugging of the cracks due to mineral deposits. Thus, the effective water flow rate through cracks in the drip shield will be extremely low and will not contribute significantly to the overall radionuclide release rate from the repository. Therefore, since the primary role of the drip shield is to keep water from contacting the waste package, SCC of the drip shield does not compromise the design purpose of the drip shield.

Based on the above rationale, this FEP is excluded for the drip shield due to low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.6 Localized Corrosion of Waste Packages (2.1.03.03.0A)

FEP Description:

Localized corrosion (pitting or crevice corrosion) could enhance degradation of the waste packages.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

Localized corrosion is a corrosion attack at discrete sites or in a nonuniform manner. Based on a conservative linear growth model the rate of localized corrosion penetration is generally higher than the rate of general corrosion penetration, and could lead to eventual breach of waste packages.

In *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [169984], Section 6.4.4) localized corrosion of the waste package outer barrier occurs when the corrosion potential (E_{corr}), is equal to or greater than a threshold potential ($E_{critical}$), that is, $\Delta E = (E_{critical} - E_{corr}) \leq 0$. The magnitude of the ΔE is an index of the localized corrosion resistance (i.e., the larger the difference, the greater the localized corrosion resistance). The crevice corrosion initiation model components (i.e., E_{corr} and $E_{critical}$) are functions of the exposure conditions (temperature, pH, chloride ion concentration, and nitrate ion concentration). The model assumes that, once initiated, localized corrosion of the waste package outer barrier propagates at a (time independent) constant rate. As a conservative measure, the base-case localized corrosion model uses the crevice repassivation potential (E_{rcrev}) as the critical potential for the localized corrosion initiation analysis (i.e., $E_{critical} = E_{rcrev}$).

Nitrate ions have a strong inhibitive effect on localized corrosion of Alloy 22 in chloride-containing solutions (BSC 2004 [DIRS 169984], Section 6.4.4.3.2; Dunn and Brossia 2002 [DIRS 162213]). Anions containing nitrogen, phosphorus, and sulfur, which are abundant in the repository groundwater, exhibit varying degrees of inhibitive effects (Thomas 1994 [DIRS 120498]). Inhibitive anions counteract the effects of aggressive anions (e.g., chloride ions), which tend to accelerate dissolution and breakdown of the oxide passive films formed on Alloy 22. The relationship between the inhibitive and aggressive anions corresponds to competitive adsorption or ion exchange at a fixed number of sites on the oxide surface. Inhibitive anions overcome the effects of aggressive anions through participation in reversible competitive adsorption such that the adsorbed inhibitive anions reduce the surface concentration of aggressive anions below a critical value (Thomas 1994 [DIRS 120498], p. 17:54). Because only the inhibitive effect of nitrate ions is accounted for in the model, results for solutions with significant amounts of other potentially inhibitive ions such as carbonate and sulfate (in addition to nitrate ions) are conservative (BSC 2004 [DIRS 169984], Section 8.3).

The crevice repassivation potential model was constructed using data from pure chloride solutions and mixed chloride and nitrate solutions, including some data at high nitrate (up to 18 *m*) and chloride (up to 36 *m*) concentrations and high temperatures (up to 160°C). However, the long-term corrosion potential model was constructed using data from relatively dilute (in terms of chloride and nitrate ion content) mixed ionic solutions, pure chloride solutions (up to about 12 *m*), and mixed chloride (up to about 13 *m*) and nitrate (up to about 2.6 *m*) ion solutions. The maximum temperature for any data point used in construction of the corrosion potential model was 120°C. The lack of measured long-term corrosion potential data in the high temperature (> 120°C) regime indicates it would be prudent not to make use of the developed functional forms to predict localized corrosion initiation under these conditions. Localized corrosion (like general corrosion) requires the presence of a liquid water film on the waste package surface. To implement the waste package outer barrier localized corrosion initiation model, the following criteria are applied in a step-wise fashion (BSC 2004 [DIRS 169984], Section 6.4.4.6.7):

1. If aqueous brine chemistry causes the initiation of localized corrosion, then localized corrosion continues to propagate regardless of changes in the bulk chemical exposure environment. This is a conservative modeling assumption made because no detailed model of the chemistry evolution of the crevice solution is available at this time.

2. If the exposure temperature exceeds 160°C and a water film is present on the waste package surface, then localized corrosion initiates.

   Localized corrosion initiated as a result of this criterion is reevaluated in accordance with Criterion 3 (below) when the exposure temperature drops below 160°C.

3. If the exposure temperature exceeds 120°C but is less than or equal to 160°C, then

   a) If the nitrate to chloride ion ratio is 0.5 (or greater), no localized corrosion will occur, or

   b) If the nitrate to chloride ion ratio is less than 0.5, then localized corrosion initiates and continues to propagate regardless of changes in the bulk chemical exposure environment (Criterion 1).

4. If the exposure temperature is greater than or equal to 20°C and less than or equal to 120°C, then the empirical correlations for the long-term corrosion potential (E_{corr}) and crevice repassivation potential (E_{rcrev}) are evaluated in accordance with the following implementation rules. If localized corrosion is determined to initiate, then localized corrosion continues to occur regardless of changes in the bulk chemical exposure environment (Criterion 1)

   a) If the nitrate to chloride ion ratio in the environment exceeds 0.5,
      then evaluate E_{rcrev} and E_{corr} at a nitrate to chloride ion ratio of 0.5.

   b) If the molality of chloride ion in the environment exceeds 36 molal,
      then evaluate E_{rcrev} and E_{corr} at a molality of chloride ion of 36 molal.
      If the molality of chloride ion is less than 0.001 molal,
      then evaluate E_{rcrev} and E_{corr} at a molality of chloride ion of 0.001 molal.

c) If the molality of nitrate ion in the environment exceeds 18 molal,
then evaluate E_{rcrev} and E_{corr} at a molality of nitrate ion of 18 molal.
If the molality of nitrate ion is less than 0.001 molal,
then $E_{rcrev} = E^{o}_{rcrev}$ (i.e., the crevice repassivation potential in the absence of nitrate ions) and evaluate E_{corr} at a molality of nitrate ion of 0.001 molal.

d) If the pH in the environment exceeds 10.9,
then evaluate E_{rcrev} and E_{corr} at a pH of 10.9.
If the pH in the environment is less than 2.8,
then initiate localized corrosion.

In addition, Alloy 22 (UNS N06022) crevice samples were tested for over 5 years in three different solutions (simulated dilute water (SDW), simulated concentrated water (SCW) and simulated acidified water (SAW)) at 60°C and 90°C in the Project's LTCTF. None of the crevice samples has shown any indication of localized corrosion attack after being tested for over five years. The crevice corrosion initiation model also predicts no localized corrosion occurrence for these exposure conditions (BSC 2004 [DIRS 169984], Section 7.2.4, Table 7-4).

A study for the welded Alloy 22 (UNS N06022) samples in 0.5 M NaCl solutions at 95°C by the investigators at the Center for Nuclear Waste Regulatory Analysis demonstrated that nitrate ion is an effective inhibitor of localized corrosion of Alloy 22 (UNS N06022), when the nitrate to chloride concentration ratio is greater than 0.2 (Dunn and Brossia 2002 [DIRS 162213]).

As discussed in FEP 2.1.03.01.0A, General Corrosion of Waste Packages, acidified waters that contain trace metals such as lead, arsenic, calcium, and magnesium were used to develop the Alloy 22 (UNS N06022) general corrosion model used in the TSPA-LA analyses. These same solutions were considered in the analysis of localized corrosion (BSC 2004 [DIRS 169984], Section 7.2.4). The small amounts of trace metals identified in the acidified waters were appropriately considered in analyzing localized corrosion of waste packages.

Supporting Reports:

BSC 2004 [DIRS 169984]

6.2.7 Localized Corrosion of Drip Shields (2.1.03.03.0B)

FEP Description:

Localized corrosion (pitting or crevice corrosion) could enhance degradation of the drip shields.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

In *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Section 6.6.1), localized corrosion of the titanium drip shield initiates when the corrosion potential (E_{corr}) equals or exceeds the threshold potential for breakdown of the passive film ($E_{critical}$).

A correlation between exposure parameters (temperature, chloride ion concentration, and pH) and the difference between the critical potential ($E_{critical}$) and the corrosion potential (E_{corr}) (i.e., ($\Delta E = E_{critical} - E_{corr}$) was developed to show when localized corrosion could be initiated. Localized corrosion initiates when ΔE is less than or equal to zero (i.e., when E_{corr} is greater than or equal to $E_{critical}$). The results show that for the drip shield under repository conditions, ΔE is significantly greater than zero over all ranges of pH, chloride concentrations, and temperatures (BSC 2004 [DIRS 169845], Section 6.6.3) in the repository. Localized corrosion of Titanium Grade 7 will not initiate in repository-relevant environments even at pH values as high as 14 (BSC 2004 [DIRS 169845], Section 8.4).

The presence of crevices and concentrated calcium and magnesium chloride solutions and their influence on corrosion were also evaluated in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Sections 6.5.8, 6.6.4, and 6.6.5). The results in Section 6.6.5 of *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845]) show that the passive film on Titanium Grade 7 is stable in concentrated calcium and magnesium chloride solutions and general corrosion is not accelerated. Table 21 in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845]) shows that the minimum ΔE (obtained in 9 molar $CaCl_2$ at 150°C) is 1.4V, which excludes the possibility of crevice corrosion in these environments.

As discussed in FEP 2.1.03.01.0B, General Corrosion of Drip Shields, acidified waters containing trace metals such as arsenic, calcium, and magnesium were used to develop the Titanium Grade 7 general corrosion rates used in the TSPA-LA analyses. These same solutions were used in developing the localized corrosion initiation analyses (BSC 2004 [DIRS 169845], Section 6.6; DTN: LL021012712251.021 [DIRS 163112]). The small amounts of trace metals identified in the acidified waters were appropriately considered in analyzing localized corrosion of drip shields and are not considered in this discussion.

Localized corrosion of Titanium Grade 7 will not initiate under repository exposure conditions and the passive film is stable even when crevices are present and the material is exposed to concentrated calcium and magnesium chloride solutions. This FEP is excluded based on low consequence because its omission does not have a significant effect on radiological exposures to the reasonable maximally exposed individual and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.8 Hydride Cracking of Waste Packages (2.1.03.04.0A)

FEP Description:

The uptake of hydrogen and the formation of metal hydrides may mechanically weaken the waste packages and promote corrosion.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

Hydrogen generated at cathodic sites in a corroding metal may be absorbed into the metal and potentially form hydride phases. Hydrogen incorporation could lead to degradation of the mechanical properties of the material and render it susceptible to cracking even in the absence of the formation of hydride phases. Hydrogen-induced cracking (HIC) is used to refer to the impact of hydrogen on waste package materials. HIC results from the combined action of hydrogen and residual or sustained applied tensile stresses.

The Alloy 22 waste package when emplaced is protected by the Titanium Grade 7 drip shield. The drip shield is designed to protect the waste package from any fallen ground support contacting the waste package, thereby eliminating the chances of galvanic coupling. Likewise, the pallet will keep the waste package from contacting the invert (BSC 2004 [DIRS 166879], Section 7.2.2), thereby precluding any potential from galvanic coupling of the waste package contacting any material in the invert.

HIC of the waste package outer barrier (Alloy 22 (UNS N06022)) is of low consequence under repository-relevant exposure conditions for the following reasons:

- Handbook data (ASM International 1987 [DIRS 103753], pp. 650 to 652) states that fully annealed nickel-base alloys, such as Alloy 22, are essentially immune to hydrogen-induced cracking. Annealed alloys are extremely resistant to hydrogen-induced cracking, but may lack adequate strength for some structural applications. The structural requirements of the waste packages at the YMP are such that strengthening of the alloys is not required (Gdowski 1991 [DIRS 100859], Section 5). The structural requirements for the waste package are satisfied by the Stainless Steel Type 316 inner shell;

- The extremely low corrosion rates exhibited by nickel alloys are not sufficient to generate enough hydrogen to cause hydrogen-induced cracking (ASM International 1987 [DIRS 103753], p. 652);

- Ni-Cr-Mo alloys with compositions and properties similar to Alloy 22 (Alloys C-276, C-4, and 625) maintain their resistance to hydrogen-induced cracking even when heavily cold worked to yield strengths in excess of 1,240 MPa (180 ksi) (ASM International 1987 [DIRS 103753], p. 169);

- Cold-worked Ni-Cr-Mo alloys are not susceptible to hydrogen-induced cracking unless they are galvanically coupled to a less noble material (or subjected to imposed cathodic currents) and strained beyond yield (Gdowski 1991 [DIRS 100859], Section 5.1; Asphahani 1978 [DIRS 160352]).

Aging of Ni-Cr-Mo alloys at temperatures around 500°C can lead to ordering, or grain-boundary segregation, of deleterious elements such as phosphorous and sulfur, which can increase susceptibility to hydrogen-induced cracking (ASM International 1987 [DIRS 103753], p. 169). However, since the waste package temperature does not exceed 300°C even in the low-probability-seismic collapsed-drift scenario (BSC 2004 [DIRS 169565], Figure 6.3-56), significant ordering and grain-boundary segregation will not occur. Asphahani (1978 [DIRS 160352]) tested cold worked (60 percent cold swaged) and aged (500°C for 100 hours) samples of Alloy C-276 at applied stresses up to 92% of yield and cathodic current densities of 40 mA/cm^2. No incidence of HIC was found. The conclusions reached concerning HIC of the waste package are applicable to all waste packages.

As discussed in FEP 2.1.09.09.0A, Electrochemical Effects in EBS, contact between the Alloy 22 waste package outer barrier and Titanium Grade 7 drip shield is not expected. Based on the similarity of the two materials (ASM International 1987 [DIRS 103753], p. 557) in the electrochemical series, there is a very low probability of galvanic interaction and subsequent enhanced corrosion.

As the Alloy 22 waste package outer barrier is fully annealed and Alloy 22 is highly resistant to hydrogen-induced cracking, this FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.9 Hydride Cracking of Drip Shields (2.1.03.04.0B)

FEP Description:

The uptake of hydrogen and the formation of metal hydrides may mechanically weaken the drip shields and promote corrosion.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

Hydrogen generated at cathodic sites in a corroding metal may migrate into the metal and potentially form hydride phases. Hydrogen incorporation could lead to degradation of the mechanical properties of the material and render it susceptible to cracking even in the absence of the formation of hydride phases. Hydrogen-induced cracking (HIC) is used to refer to the impact of hydrogen on waste package and drip shield materials. Hydrogen-induced cracking results from the combined action of hydrogen and residual or sustained applied tensile stresses.

Hydrogen absorption in α-titanium alloys such as Titanium Grade 7 can occur when three general conditions are simultaneously met (Schutz and Thomas 1987 [DIRS 144302]; BSC 2004 [DIRS 169847], Section 6.1.2):

1. A mechanism exists for generating nascent (atomic) hydrogen on the surface
2. Metal temperature is above approximately 80°C (175°F) where the diffusion rate of hydrogen into α-titanium is significant
3. Solution pH is less than 3 or greater than 12, or impressed potentials are more negative than -0.7 V (Saturated Calomel Electrode).

As discussed in *General Corrosion and Localized Corrosion of Drip Shield* (BSC 2004 [DIRS 169845], Section 6.6) and FEP 2.1.03.03.0B, Localized Corrosion of Drip Shields, localized corrosion (crevice corrosion and pitting) will not occur under the exposure conditions anticipated in the repository. However, in the repository design that serves as the basis for the license application (BSC 2004 [DIRS 168489]), passive general corrosion and galvanic coupling of the drip shield to less noble materials are feasible processes in the repository that could lead to hydrogen generation on the surface of the drip shield. Some of the hydrogen produced can diffuse into the metal potentially forming hydrides. The direct absorption of radiolytically produced hydrogen is insignificant except at dose rates > 10^2 Gy/hr (Grays per hour where 1 Gray = 100 rads) and temperatures > 150°C (BSC 2004 [DIRS 169847], Section 6.1.2). Satisfaction of all three general conditions for hydrogen absorption is unattainable in the repository (BSC 2004 [DIRS 169847], Section 6.1.2). At certain repository locations, where temperatures are ≥ 80°C and concentrated groundwater is present, conditions (2) and (3) may be satisfied, however, only when all three conditions are met simultaneously can hydrogen absorption can be anticipated.

For the passive noncreviced or inert crevice conditions expected to prevail, the corrosion of the titanium alloy will be sustained by reaction with water under neutral conditions ($Ti + 2H_2O \rightarrow TiO_2 + 2\ H_2$) and will proceed at an extremely slow rate. This process will generate hydrogen, which must pass through the TiO_2 film before absorption into the underlying titanium alloy (BSC 2004 [DIRS 169847], Section 6.1.2).

As stated in *Hydrogen Induced Cracking of Drip Shield* (BSC 2004 [DIRS 169847], Section 8), a simple and conservative model was developed to evaluate the effects of hydrogen-induced cracking on the drip shield. The basic premise of the model is that failure will occur once the hydrogen content exceeds a certain limit or critical value, H_C. Despite the potential occurrence of hydrogen absorption into the bulk structure, hydrogen induced cracking of Titanium Grade 7 is not expected, due to the existence of the oxide film and the high critical value, H_C, because of the addition of palladium (Pd) (BSC 2004 [DIRS 169847], Section 6.1.4). Recent analyses of published data for Titanium Grade 16, whose performance is similar to the Titanium Grade 7, suggest that the critical concentration may be well in excess of 1,000 μg/g (BSC 2004 [DIRS 169847], Section 6.1.3). Titanium Grade 7 has a higher Pd concentration than Titanium Grade 16; therefore, the H_C value for Titanium Grade 7 must be at least 1,000 μg/g and could be much higher (BSC 2004 [DIRS 169847], Section 6.1.3). The hydrogen concentration in the drip shield from passive corrosion 10,000 years after permanent closure is 510 μg/g, resulting from a conservative estimate (BSC 2004 [DIRS 169847], Section 8.1). This is below the threshold concentration and would not result in hydrogen induced cracking or any degradation of fracture toughness.

In the repository design, hydrogen generation may be caused by the galvanic couple formed between the Titanium Grade 7 Drip Shield surface and less noble structural components (such as rock bolts, wire mesh, and steel liners used in the drift), which may fall onto the drip shield surface. If temperatures are ≥ 80°C and concentrated groundwaters are present, then the formation of locally hydrided "hot spots" are possible. However, the effect of these locally hydrided regions is negligible because (BSC 2004 [DIRS 169847], Section 6.3.2):

1. The contact area is likely to be small, and the anode-to-cathode area ratio (area of steel and titanium, respectively) low leading to only limited amounts of hydrogen absorption.

2. The intermittent nature of seepage at temperatures ≥ 80°C will lead to limited periods of the aqueous conditions required to sustain an active galvanic couple, thereby limiting hydrogen absorption while temperatures are high enough to drive hydrogen transport into the metal. The intermittent wetting and drying cycles anticipated on the drip shield will lead to the ready formation of calcareous and mineral deposits, which are well known to dramatically suppress galvanic currents, thereby stifling hydrogen absorption.

3. Conditions in the repository will be oxidizing, making it less likely that the couple will sustain water reduction and, hence, hydrogen absorption.

4. Titanium drip shield and the steel component surfaces will experience a considerable period of dry temperatures ≥ 85°C. This will leave the titanium and steel in the passive state (especially titanium) and avoid galvanic contact and hydrogen absorption by titanium.

5. As discussed above, α-titanium alloys exhibit a protective oxide film and a relatively high critical value, H_C, due to palladium addition, reducing the tendency for hydrogen induced cracking (BSC 2004 [DIRS 169847], Sections 6.1.3 and 6.1.5).

The general corrosion rates of titanium alloys under the repository conditions are very low. Thus, hydrogen generation rates and, hence, hydrogen absorption rates, are expected to be very low (BSC 2004 [DIRS 169847], Section 6.1.2).

In summary, hydrogen-influenced cracking of the Titanium Grade 7 is not expected to occur under the anticipated repository conditions. Locally hydrided regions, resulting from galvanic coupling, are possible, but their effects are negligible. Therefore, this FEP can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:
N/A

6.2.10 Microbially Influenced Corrosion (MIC) of Waste Packages (2.1.03.05.0A)

FEP Description:
Microbial activity may catalyze waste package corrosion by otherwise kinetically hindered oxidizing agents. The most likely process is microbial reduction of groundwater sulfates to sulfides and reaction of iron with dissolved sulfides.

Screening Decision:
Included

Screening Argument:
N/A

TSPA Disposition:
Microbially influenced corrosion (MIC) is the contribution to the corrosion of a metal or alloy due to the presence or activity, or both, of microorganisms. Microbially influenced corrosion most often occurs due to the increase in anodic or cathodic reactions due to the direct impact of microorganisms on the alloy or by indirect chemical effects on the surrounding solution (BSC 2004 [DIRS 169984], Section 6.4.5). Waste package microbially influenced corrosion is discussed in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984], Section 6.4.5). The effect of microbially influenced corrosion on waste package corrosion is analyzed with an enhancement factor approach (i.e., microbially influenced corrosion increases the general corrosion penetration rate). In this approach, the abiotic corrosion rate is multiplied by the enhancement factor when exposure conditions in the emplacement drift are conducive to significant microbial activity (BSC 2004 [DIRS 169984], Section 6.4.5). The microbially influenced corrosion enhancement factor, $f_{MIC,}$ is uniformly distributed between 1 and 2 (BSC 2004 [DIRS 169984], Section 6.4.5) and is applied to the waste package outer barrier general corrosion rate when the relative humidity at the waste package outer barrier surface is above 90 percent (BSC 2004 [DIRS 169984], Section 6.4.5). The general corrosion rate enhancement factor is applied to the entire waste package surface (BSC 2004 [DIRS 169984], Section 6.4.5) when the relative humidity threshold is satisfied. This treatment of microbially influenced corrosion applies to all waste packages.

In the integrated waste package degradation analysis (BSC 2004 [DIRS 169996]), the general corrosion rate enhancement factor is sampled once per realization (i.e., the variation in the general corrosion rate microbially influenced corrosion enhancement factor, $f_{MIC,}$ is entirely due to uncertainty (BSC 2004 [DIRS 169984], Section 6.4.5)), and applied to the entire waste package surface. The output from the integrated waste package degradation analysis is a set of profiles (time histories) for the failure (i.e., initial breach) and subsequent number of penetration openings in the waste package and drip shield as a function of time. The integrated waste package degradation analysis is used directly in the TSPA-LA.

Supporting Reports:

BSC 2004 [DIRS 169984]
BSC 2004 [DIRS 169996]

6.2.11 Microbially Influenced Corrosion (MIC) of Drip Shields (2.1.03.05.0B)

FEP Description:

Microbial activity may catalyze drip shield corrosion by otherwise kinetically hindered oxidizing agents. The most likely process is microbial reduction of groundwater sulfates to sulfides and reaction of iron with dissolved sulfides.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

Microbially influenced corrosion (MIC) of titanium is discussed in *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Section 6.7.2). Corrosion handbooks and literature reviews generally indicate that titanium alloys are immune to microbially influenced corrosion (Revie 2000 [DIRS 159370], Chapter 47; Little and Wagner 1996 [DIRS 131533]; and Brossia et al. 2001 [DIRS 159836], Section 4.1.3) under repository-like conditions. The remarkable stability of the TiO_2 passive film formed on titanium alloys confers this immunity. While titanium is susceptible to biofouling in seawater solutions, the biofilm does not compromise the integrity of the passive film and, therefore, biofouled titanium maintains its resistance to localized corrosion processes (Revie 2000 [DIRS 159370], Chapter 47). It has been reported that production of nitrates, polythionates, thiosulfates, and oxygen associated with aerobic biologic activity does not significantly increase the corrosion rate of titanium alloys (Brossia et al. 2001 [DIRS 159836], Section 4.1.3).

Steep gradients in O_2 and pH can exist within biofilms; typically aerobic and near neutral in the outer layers becoming acidic and low in O_2 close to the metal surface (Shoesmith and Ikeda 1997 [DIRS 151179], Section 6). Hydrogen peroxide has been detected in biofilms at millimolar levels, the amount of which is thought to be controlled by bacteria enzymes during the aerobic respiration process (Shoesmith and Ikeda 1997 [DIRS 151179]). Hydrogen peroxide maintains a low pH (< 3) near the metal by oxidizing metal cations that then undergo hydrolysis. These chemical changes can lead to enoblement (a shift of the corrosion potential to more positive values) of titanium by up to 500 mV (Shoesmith and Ikeda 1997 [DIRS 151179], Section 6). As shown by Figures 19 and 20 of *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845], Section 6.6.3), ΔE far exceeds 500 mV at low pH values (i.e., localized corrosion will not initiate even if the corrosion potential is increased by 500 mV). Enoblement can also lead to several beneficial effects, including thickening of the passive film and a decrease in the number and density of film defects (Shoesmith and Ikeda 1997 [DIRS 151179], Section 3 and 6). According to Shoesmith et al. (1995 [DIRS 117892]), initiation of crevice corrosion under biofilms is highly unlikely for titanium. Lastly, microbial growth in the repository will likely be limited by the availability of nutrients (BSC 2004 [DIRS 169984], Section 6.4.5).

Microbially influenced corrosion has no significant effect on either general or localized corrosion processes of titanium alloys under the exposure conditions in the repository. Therefore, this FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.12 Internal Corrosion of Waste Packages Prior to Breach (2.1.03.06.0A)

FEP Description:
Aggressive chemical conditions within the waste package could contribute to corrosion from the inside out. Effects of different waste forms, including CSNF and DSNF, are considered in this FEP.

Screening Decision:
Excluded (Low Consequence)

Screening Argument:
Project Requirements Document (Canori and Leitner 2003 [DIRS 166275], PRD-013/T-036 and PRD-013/T-045) states that waste packages will be designed to preclude internal corrosion of the waste package and contained material. The waste package design will also preclude chemical, electrochemical, or other reactions (such as internal corrosion) of the waste package such that there will be no adverse effect on normal handling, transportation, storage, emplacement, containment, or isolation, or on abnormal occurrences such as a waste package drop accident and premature breach in the repository. This FEP pertains only to internal corrosion of the waste package.

The CSNF assemblies will be dried prior to their insertion into the waste packages (CRWMS M&O 2000 [DIRS 123881], p. II-10). After being loaded with waste, the waste packages are evacuated, to remove air and ensure the waste form is dry, then backfilled with helium to severely limit oxidizing gases (e.g., oxygen and water vapor) and help preserve the chemical and physical stability of the waste form (BSC 2004 [DIRS 170803], Section 4.2.2). The inert gas environment is intended to ensure a negligible amount of corrosion degradation prior to the breach of the waste packages. The requirements are incorporated into *DOE and Commercial Waste Package System Description Document* (BSC 2004 [DIRS 170803], Section 3). The repository will be designed and constructed in accordance with an approved QA program thereby ensuring the requirements of *DOE and Commercial Waste Package System Description Document* are implemented correctly.

Analyses performed by Kohli and Pasupathi (1986 [DIRS 131519]) suggest that the most likely cause of internal corrosion is the residual moisture remaining in the waste package at the time of emplacement. The primary source of this residual moisture is from waterlogged failed fuel rods. Analyses presented in the above reference indicate that the amount of moisture available to cause internal corrosion is very limited, and even with very conservative assumptions, the potential for degradation of the waste package materials is very remote. Additionally, all waste package types have significant amounts of internals (e.g., basket materials composed of carbon steel) (BSC 2004 [DIRS 169472]) that will corrode in preference to the Alloy 22 (UNS N06022) waste package outer barrier and the Stainless Steel Type 316 inner vessel (ASM International 1987 [DIRS 103753], p. 557). Thus, no significant corrosion damage to the waste package outer barrier and Stainless Steel Type 316 inner vessel internal surfaces will occur.

DSNF waste packages containing N Reactor spent fuel may have significant quantities of residual free and chemically bound water at the time of sealing prior to placement in storage at its present location. However, the N Reactor spent fuel cladding is significantly damaged, thus exposing chemically reactive uranium metal surfaces, which could react with residual water producing uranium oxide and uranium hydride. Other forms of DSNF are less damaged, and will contain much lower quantities of residual water due to drying prior to sealing for storage at its present location. Damaged DSNF will be placed in waste packages (CRWMS M&O 2000 [DIRS 150823], Table 5) that will not contain any residual water until breached.

As discussed in FEP 2.1.03.04.0A, Hydride Cracking of Waste Packages, HIC of the waste package outer barrier (Alloy 22) is not considered to be a credible degradation mechanism under repository-relevant exposure conditions and will not enhance internal corrosion.

In view of the above discussion, it can be concluded that insignificant corrosion damage of DSNF waste packages, DHLW glass waste packages, and CSNF waste packages will occur due to drying of the waste form before loading, backfilling of the waste packages with an inert gas, and the presence of significant amounts of waste package internals, which will corrode in preference to the Alloy 22 (UNS N06022) waste package outer barrier and the Stainless Steel Type 316 inner vessel. This FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.13 Mechanical Impact on Waste Package (2.1.03.07.0A)

FEP Description:

Mechanical impact (dynamic loading) on the waste package may be caused by internal and external forces such as internal gas pressure, forces caused by swelling corrosion products, rockfall, and possible waste package or drip shield movement. Seismic-induced impacts are addressed in a separate FEP.

Screening Decision:

Excluded (Low Probability) (Rockfall)
Excluded (Low Consequence) (Internal gas pressure and swelling of corrosion products)

Screening Argument:
The arguments put forth and the conclusions reached in this section are equally applicable to all waste packages.

Internal gas pressure: A calculation of the maximum stresses developed in the waste package due to internal pressurization as a result of fuel rod rupture at 400°C is less than *1995 ASME Boiler and Pressure Vessel Code* (ASME 1995 [DIRS 141257]) requirements for the allowable tensile strength of the materials used for the waste package (CRWMS M&O 2000 [DIRS 150823], Section 6.3). In the repository, the maximum waste package surface temperature will be less than 300°C even in the low-probability-seismic collapsed-drift scenario (BSC 2004 [DIRS 169565], Figure 6.3-56). Therefore, with the current robust waste package design, the pressurization of the internal gas under the expected repository condition would not cause mechanical damage to the waste package and can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

Swelling of corrosion products: Mechanical damage to the waste package from swelling corrosion products is discussed in greater detail under FEP 2.1.09.03.0B, Volume Increase of Corrosion Products Impacts Waste Package. Analyses cited in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984], Section 6.4.2), indicate that, even under very conservative assumptions, the growth of the corrosion product (primarily Cr_2O_3 as stated in BSC 2004 [DIRS 169984], Section 6.4.2) oxide layer (93 µm in 10,000 years) is not thick enough to produce enough pressure to cause mechanical damage to the Stainless Steel Type 316 inner shell or the Alloy 22 (UNS N06022) outer barrier. Alloy 22 (UNS N06022) contains 20.0 to 22.5 weight % chromium (BSC 2004 [DIRS 169984], Section 1.1). Therefore, waste package damage from swelling corrosion products is excluded based on low probability of occurrence under the exposure conditions in the repository.

Rockfall: Mechanical damage of the waste package by rockfall is discussed in greater detail under FEP 2.1.07.01.0A, Rockfall. As indicated in *Subsurface Facility Description Document* (BSC 2004 [DIRS 170265], Section 3.1.1.4.14.1) the drip shield will be designed to protect the waste package from rockfalls during postclosure. Because the drip shield provides adequate protection to the waste packages from rockfall (FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield), the effects of rockfall on the waste package are excluded from consideration due to low probability of occurrence.

Seismic: Mechanical damage of the waste packages and drip shields by ground motion and rockfalls during seismic events is discussed 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 in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.14 Mechanical Impact on Drip Shield (2.1.03.07.0B)

FEP Description:

Mechanical impact (dynamic loading) on the drip shield may be caused by forces such as rockfall and possible waste package or drip shield movement. Seismic-induced impacts are addressed in a separate FEP.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

Mechanical damage of the drip shield by rockfall is discussed in greater detail under FEP 2.1.07.01.0A, Rockfall. That FEP discussion provides relevant references discussing the issue in greater detail. In addition, the emplacement drift system design criteria require the drip shield to be designed to protect the waste package from rockfalls during postclosure (BSC 2004 [DIRS 170265], Section 3.1.1.4.14.1). *Drip Shield and Waste Package Emplacement Pallet Design Report* (BSC 2004 [DIRS 166879], Section 7.2.1) shows that the deflection of the drip shield due to an 11.5 metric ton rockfall (which produces the maximum vertical displacement in the drip shield components) is not large enough to cause the drip shield to contact the waste package. The maximum displacement from the 11.5 metric ton rockfall event is 254 mm (BSC 2004 [DIRS 166879], Section 7.2.1). The minimum gap upon emplacement between the drip shield and waste package outer barrier is about 367 mm (BSC 2004 [DIRS 166879], Section 7.2.1).

Thus, the drip shield provides adequate protection to the waste package from rockfall. In view of the above rationale, this FEP is excluded as low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

Mechanical damage of the waste package and drip shield by ground motion during seismic events is discussed in greater detail under 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, in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.15 Early Failure of Waste Packages (2.1.03.08.0A)

FEP Description:

Waste packages may fail prematurely because of manufacturing defects, improper sealing, or other factors related to quality control during manufacture and emplacement.

Screening Decision:
Included

Screening Argument:
N/A

TSPA Disposition:
Analysis of Mechanisms for Early Waste Package/Drip Shield Failure (BSC 2004 [DIRS 170024]) evaluates several mechanisms for early breach of the waste package. Of the mechanisms evaluated, weld flaws, improper heat treatment, improper laser peening, and improper handling of waste packages were determined to be included in the TSPA-LA model (BSC 2004 [DIRS 170024], Section 7).

As discussed in FEP2.1.03.02.0A, Stress Corrosion Cracking of Waste Packages, weld flaws act as sites for initiation of SCC in the waste package closure weld regions (BSC 2004 [DIRS 169985]; BSC 2004 [DIRS 169996]). Weld flaws are included in TSPA-LA analysis through their effect on SCC of waste packages.

Improper heat treatment results primarily from improper stress-relief annealing. The consequence of improper laser peening is the introduction of unacceptable amounts of cold work in the material and increased susceptibility to stress corrosion cracking. Improper handling of the waste packages may lead to gouges in the waste package outer surface and provide sites for stress corrosion cracks. A discussion of these causes and consequences is provided in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (BSC 2004 [DIRS 170024], Section 6.4.8).

The number of early failed waste packages per realization is represented by a Poisson distribution with an uncertain intensity. The Poisson intensity is sampled from a log normal distribution with a median of 7.2×10^{-6} and an error factor of 15 (BSC 2004 [DIRS 170024], Section 7, Table 22). Improperly heat-treated waste packages might be susceptible to stress corrosion cracking or lead to the formation of grain-boundary precipitates. In lieu of trying to identify a single specific mechanism of degradation, a very conservative approach is used based on recommendations in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (BSC 2004 [DIRS 170024], Section 6.4.8) for evaluating waste package early breach:

- A failure of the waste package outer barrier shell and outer and inner closure lids should be assumed, as well as the failure of the stainless steel structural inner vessel and closure lid.

- The affected waste packages should be assumed to fail immediately upon initiation of degradation processes.

- The entire waste package surface area should be considered affected by waste package early failure.
- The materials of the entire affected area should be assumed lost upon failure of the waste package because the affected area will be subjected to stress corrosion cracking and highly enhanced localized and general corrosion.

This early breach treatment applies to all waste packages.

Waste package emplacement errors, including design deviations, improper quality control, surface contamination, and administrative errors leading to unanticipated conditions, are addressed in FEP 1.1.03.01.0A, Error in Waste Emplacement, which is shared between this report and *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

Supporting Reports:

BSC 2004 [DIRS 169996]
BSC 2004 [DIRS 170024]

6.2.16 Early Failure of Drip Shields (2.1.03.08.0B)

FEP Description:

Drip shields may fail prematurely because of manufacturing defects, improper sealing, or other factors related to quality control during manufacture and emplacement.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

Evaluation of several mechanisms for early failure of the drip shield including weld flaws, base metal flaws, improper weld material or base metal, improper heat treatment, contamination, improper handling, and drip shield emplacement error was performed in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (BSC 2004 [DIRS 170024], Section 6.3).

The consequence of weld flaws, base metal flaws, improper heat treatment, and damage by mishandling is an increased susceptibility to stress corrosion cracking (BSC 2004 [DIRS 170024], Section 6.4).

For the drip shields, fabrication welds will be fully stress-relief annealed before placement in the drifts (Plinski 2001 [DIRS 156800], Section 8.3.17). Therefore, drip shields are not subject to SCC upon emplacement (BSC 2004 [DIRS 169985], Section 6.3.7). However, the drip shields are subject to SCC as a result of seismic-induced loading and rockfalls. Seismic effects on drip shield degradation are discussed 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 (BSC 2004 [DIRS 169898]). The consequence of stress corrosion cracking (see also FEP 2.1.03.02.0B, Stress Corrosion Cracking (SCC) of Drip Shields) on drip shield performance is discussed in

FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield. FEP 2.1.03.10.0B is excluded on the basis of low consequence because the amount of advective water flow through cracks in the drip shield will be severely limited based on the small aperture width (narrow opening and tight cracks), the presence of capillary forces, and the potential for plugging of the cracks due to mineral deposits. Thus, the effective water flow rate through cracks in the drip shield will be extremely low and will not contribute significantly to the overall radionuclide release rate from the repository. Therefore, since the primary role of the drip shield is to keep water from contacting the waste package, SCC of the drip shield (which is the consequence of weld flaws, base metal flaws, improper heat treatment, and damage by mishandling) does not compromise the design purpose of the drip shield.

The use of improper weld or base metal material is possible in the repository (BSC 2004 [DIRS 170024], Section 6.3.3); however, due to the strict controls that will govern the fabrication of the drip shield, it is expected that the material composition of the improper weld or base metal material will differ only slightly from the intended composition (BSC 2004 [DIRS 170024], Section 6.4.2). In view of the high corrosion resistance of the materials in question, the consequences of improper weld or base metal will be insignificant (BSC 2004 [DIRS 170024], Section 6.4.2).

The probability of drip shield surface contamination is also evaluated in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (BSC 2004 [DIRS 170024], Section 6.3.5). It is found that the consequence of drip shield surface contamination is not significant from a corrosion standpoint (BSC 2004 [DIRS 170024], Section 6.4.5). On this basis, drip shield surface contamination is considered to be insignificant.

As discussed in FEP 1.1.03.01.0A, Error in Waste Emplacement, errors in drip shield emplacement large enough to allow dripping water to contact underlying waste package would be detected and remedied. FEP 1.1.03.01.0A is excluded based on low consequence to radiological exposures to the reasonable maximally exposed individual and radionuclide releases to the accessible environment.

Thus, manufacturing defects in the drip shield and early failure mechanisms (not including drip shield emplacement error (FEP 1.1.03.01.0A)) of the drip shield can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.17 Copper Corrosion in EBS (2.1.03.09.0A)

FEP Description:

Chemical reactions involving copper corrosion have been identified as being of potential interest for repository programs considering the use of copper containers.

Screening Decision:
Excluded (Low Consequence)

Screening Argument:
The repository does not use copper containers and thus this FEP description does not apply as written. However, consideration of copper effects is of merit since a small amount of copper may be present in the emplacement drifts as part of the gantry rail system (BSC 2001 [DIRS 154441]). The copper would have no adverse effects on the performance of the Alloy 22 (UNS N06022) waste package outer barrier or Titanium Grade 7 drip shield material as there is no potential for the waste package or the drip shield to come in contact with copper. The waste package is designed to rest on a pallet, which is constructed of Alloy 22 (BSC 2004 [DIRS 168489]) and is designed to keep the waste package from contacting other dissimilar metals (BSC 2004 [DIRS 166879], Section 7.2.2). Similarly, the drip shields are designed to contact no other material except Alloy 22 (UNS N06022) feet, which are attached to the bottom of the drip shields (BSC 2004 [DIRS 168489]). Therefore, the effect of copper corrosion is of low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

If, however, the drip shield were to come in contact with copper due to the failure of gantry system, there is a potential for galvanic interaction between the titanium and copper and subsequent hydrogen absorption. The potential for hydrogen absorption, or hydrogen induced cracking, is considered low since Titanium Grade 7 contains Pd and because a protective oxide film forms on the surface of the drip shield, which hinders hydrogen absorption (BSC 2004 [DIRS 169847], Section 6.1.3). When Titanium Grade 7 is cathodically polarized at -1.2 V (versus saturated calomel electrode), a more severe condition than galvanically coupling to copper, the hydrogen absorption efficiency was found to be 0.015 and the resulting hydrogen absorption insignificant (BSC 2004 [DIRS 169847], Section 6.1.6). Thus, corrosion due to copper in the gantry rail system may be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.18 Advection of Liquids and Solids Through Cracks in the Waste Package (2.1.03.10.0A)

FEP Description:

The presence of one or more cracks or other small openings of sufficient size in a waste package may provide a pathway for the advective flow of water (e.g., thin films or droplets) or solid material into the waste package. The resulting presence of sufficient water or solid material in the waste package may affect in-package chemistry and/or criticality. Partial or full plugging of the waste package cracks by chemical or physical reactions after their formation (i.e., healing) could also affect water flow and radionuclide transport through the waste package. Passivation by corrosion products is a potential mechanism for waste package healing.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

Formation of stress corrosion cracks in the waste packages are modeled in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985]). This document states that the through wall cracks will be located in the closure lid weld region and oriented in the radial direction. Circumferential cracks in the closure lid weld region may occur but they are not expected to penetrate through thickness of the lid because of the stress profile through the thickness of the lid (BSC 2004 [DIRS 169985]).

Typically the stress corrosion cracks are very tight and tortuous and often with many branches. *EBS Radionuclide Transport Abstraction* (BSC 2004 [DIRS 169868], Section 6.3.3) (RTA model) provides a prediction of crack aperture size for the cracks in the waste package. A range of crack aperture values are calculated using the estimated residual stress (from finite element simulations) and the crack length (approximately twice the lid thickness). *EBS Radionuclide Transport Abstraction* (BSC 2004 [DIRS 169868], Section 6.3.3) calculates the following crack widths at the inner and outer surfaces (118 μm and 196 μm, respectively) of the through wall radial cracks in the waste package lid using the method outlined in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985], Section 6.5.2).

Water vapor diffusion is treated in the RTA model as the only probable mechanism for water to enter the waste package through the SCCs. Advective flow through the cracks is of low consequence based on the following argument. First, a film that completely spans the opening of a stress corrosion crack creates a differential in the capillary forces that will prevent any further ingress of flowing water into the waste package. Second, the presence of corrosion products in the tight stress corrosion crack may provide capillary barrier for advective flow into the waste package. Third, the corrosion products in the cracks will provide resistance to flow requiring a substantial pressure gradient that is unlikely to exist. Additionally, the RTA model follows the logic that dripping water is capable of contacting a stress corrosion crack in the lid only if the waste package is tilted upward. This will not occur as the ends of the pallet designed to support the waste package are made of Alloy 22 and are expected to last beyond the regulatory period (BSC 2004 [DIRS 166879], Section 7.2.2).

Based on the above discussion, the amount of advective water flow through cracks in waste packages will be severely limited based on the low probability of SCC formation, small aperture width (narrow opening and tight cracks), and the presence of capillary forces. On these bases, advective flux of liquids and solids through cracks in the waste package is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.19 Advection of Liquids and Solids Through Cracks in the Drip Shield (2.1.03.10.0B)

FEP Description:

The presence of one or more cracks or other small openings of sufficient size in a drip shield may provide a pathway for the advective flow of water (e.g., thin films or droplets) or solid material through the drip shield. The resulting flux may affect drip shield performance and/or subsequent dripping onto the waste packages. Partial or full plugging of the drip shield cracks by chemical or physical reactions after their formation (i.e., healing) could also affect water flow through the drip shield.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

As discussed in FEP 2.1.03.03.0B, Localized Corrosion of Drip Shields, no localized corrosion of the drip shield is expected under repository exposure conditions. However, drip shields are subject to stress corrosion cracking (SCC), under seismic induced loading and rockfalls due to drift collapse. If cracks were to be produced in the drip shield due to rockfalls, they tend to be very tight in passive alloys such as Titanium Grade 7 (BSC 2004 [DIRS 169985], Section 6.3.7). Crack sizes are estimated in *Flow of Water and Pooling in a Waste Package* to be between 130 to 208 μm (BSC 2001 [DIRS 152248], Section 6.3.3). Another estimate of the crack width was made in *Plugging of Stress Corrosion Cracks by Precipitation* (BSC 2001 [DIRS 156807], Table 5-13) and suggests a value of 157 μm which is within the range provided above.

Flow of water through the stress corrosion cracks in the drip shields is addressed in *Water Distribution and Removal Model* (CRWMS M&O 2001 [DIRS 152016]). An analysis of crevice water holding capacity in that report shows that for a drip shield with a wall thickness of 15 mm, the maximum crevice width that can hold a full column of water (fully saturated crack) is 0.916 mm (CRWMS M&O 2001 [DIRS 152016], Section 6.1.4), based on the contact angle of the pendant drop on the surface. Cracks capable of supporting pendant drop contact angles of 180° or larger will not be fully saturated. This analysis implies that crevices smaller than 0.916 mm when fully saturated will be capable of supporting surface water film, and thereby not allow water to pass through the crevice. Cracks larger than 0.916 mm can be partially saturated and may allow water to pass through the crack. However, as discussed above the width of cracks in the drip shields are significantly below the aperture width (0.916 mm) needed for supporting flow through the crack.

Even if some flow through the cracks is possible, the small heat flux across the thickness of the drip shield will result in the evaporation of the slow flowing water and deposition of mineral scale deposit (principally calcium carbonate or calcite) in the crack and at the surface. This formation of calciferous deposits is well documented in seawater environments and in heat exchangers through which natural brines are forced to flow, such as in desalination plants and in potash plants (Cowan et al. 1976 [DIRS 105212], pp. 1 to 39 and 376 to 383). In both these cases, titanium surfaces are heat sources operating at about 100°C. Such deposits form rapidly under flowing conditions, and have to be removed regularly to avoid heat exchanger efficiency.

A detailed calculation of the expected rate of SCC plugging due to calcite precipitation has been performed and is documented in *Plugging of Stress Corrosion Cracks by Precipitation* (BSC 2001 [DIRS 156807], Section 6). The calculation conservatively assumes that the corrosion products generated on the crack faces, as well as colloids, particulates, and any precipitated minerals, do not help in resisting the water flow and that there is a uniform water seepage flow in space and time. It is concluded that under these assumptions, the cracks will be sealed in a few hundred years at most when water is allowed flow at a low film flow rates.

Based on the above discussion, the amount of advective water flow through cracks in drip shields will be severely limited based on the small aperture width (narrow opening and tight cracks), the presence of capillary forces, and the potential for plugging of the cracks due to mineral deposits. On these bases, advective flux of liquids and solids through cracks in the drip shields is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.20 Physical Form of Waste Package and Drip Shield (2.1.03.11.0A)

FEP Description:

The specific forms of the various drip shields, waste packages, and internal waste containers that are proposed for the Yucca Mountain repository can affect long-term performance. Waste package form may affect container strength through the shape and dimensions of the waste package and affect heat dissipation through waste package volume and surface area. Waste package and drip shield materials may affect physical and chemical behavior of the disposal area environment. Waste package and drip shield integrity will affect the releases of radionuclides from the disposal system. Waste packages may have both local effects and repository-scale effects. All types of waste packages and containers, including CSNF, DSNF, and DHLW, should be considered.

Screening Decision:

Included

Screening Argument:

N/A

TSPA Disposition:

The waste package, drip shield, and repository design configurations for the Yucca Mountain Project are shown in *D&E/PA/C IED Emplacement Drift Configuration and Environment* (BSC 2004 [DIRS 168489]). Only one drip shield configuration is used in the repository. While more than one waste package configuration will be used in the repository, they are all similar in their general design, fabrication methodology, and dimensions (Plinski 2001 [DIRS 156800], Section 1). Therefore, there will be little variation in strength, dimensions, and shape of the waste packages used in the repository. Waste package and drip shield degradation modes are modeled in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984]), *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845]), and analyzed in *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (BSC 2004 [DIRS 169996]).

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2004 [DIRS 169996], Section 6.3.2) analyzes two nominal waste package configurations. The first waste package configuration is referred to as the commercial spent nuclear fuel (CSNF) waste package configuration for which the 21-PWR waste package configuration parameters are used because the 21-PWR waste package configuration is the most common waste package configuration in the repository. The second waste package configuration analyzed is the codisposal (CDSP) waste package configuration whose length is considered to be the average length of the 5 HLW/1 DOE SNF long and short waste package configurations (which have roughly equal populations in the inventory). Because the integrated waste package degradation analysis documented in *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (BSC 2004 [DIRS 169996]) is used directly in the TSPA-LA, the effects of differences in the physical form of waste packages on corrosion processes are included in TSPA-LA.

The physical effects of degraded waste packages and drip shields on flow and transport of radionuclides are indirectly included in the selection of the EBS flow pathways, but they do not have an explicit effect because the flow pathways are modeled without regard to the detailed mechanisms of flow (FEP 2.1.08.07.0A, Unsaturated Flow in the EBS) (BSC 2004 [DIRS 169898]). Chemical effects of the waste package and drip shield materials are discussed in FEPs 2.1.09.01.0A, Chemical Characteristics of Water in Drifts; and 2.1.09.02.0A, Chemical Interactions with Corrosion Products, in *Engineered Barrier System Features, Events and Processes* (BSC 2004 [DIRS 169898]).

Supporting Reports:

BSC 2004 [DIRS 169984]
BSC 2004 [DIRS 169845]
BSC 2004 [DIRS 169996]

6.2.21 Oxygen Embrittlement of Drip Shields (2.1.06.06.0B)

FEP Description:

A potential failure mechanism for drip shields is oxygen embrittlement, resulting from the diffusion of interstitial oxygen in the titanium at high temperatures.

Screening Decision:

Excluded (Low Probability)

Screening Argument:

Oxygen embrittlement of titanium results from diffusion of interstitial oxygen into the metal at temperatures > 340°C (ASM International 1987 [DIRS 103753], p. 681). The time to failure depends on the alloy composition, material thickness, and stress state. In the repository, the maximum waste package surface temperature (because the waste package is the heat source, the waste package surface temperature would be greater than the drip shield surface temperature) is less than 300°C even in the low-probability-seismic collapsed-drift scenario (BSC 2004 [DIRS 169565], Figure 6.3-56). Because this temperature is less than the threshold temperature for oxygen embrittlement of 340°C, oxygen embrittlement of the titanium drip shields is excluded on the basis of low probability of occurrence under the exposure conditions in the repository.

Hydrogen embrittlement is discussed in FEP 2.1.03.04B, Hydride Cracking of Drip Shields. The effects of potential acidification in creviced regions such as cracks is discussed in FEP 2.1.03.03.0B, Localized Corrosion of Drip Shields.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.22 Mechanical Effects at EBS Component Interfaces (2.1.06.07.0B)

FEP Description:

Physical effects of steady-state contact (static loading) that occur at the interfaces between materials in the drift may affect the performance of the system.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

The waste package and the drip shield, as designed and emplaced, come in contact with very few other EBS components. For example, the waste package is designed to rest on a pallet, which is constructed of Alloy 22 (UNS N06022) and is designed to keep the waste package from contacting other dissimilar metals (BSC 2004 [DIRS 168489]). The pallet is also designed to keep the waste package supported in a horizontal position, and away from the invert and ground support under nonseismic scenarios (BSC 2004 [DIRS 166879], Section 7.2.2). Similarly the drip shields are designed to contact no other material except the Alloy 22 feet, which are attached to the bottom of the drip shields. These feet are in contact with the invert, which is covered by crushed tuff as ballast.

The possibility of the drip shield contacting the waste package as a result of mechanical damage caused by rockfall was considered. This possibility was, however, addressed and excluded as discussed in FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield.

The mechanical loading at the interfaces between the waste package and (degraded) pallet has been analyzed (BSC 2004 [DIRS 169766]). The contact stresses are shown to be much less (maximum stress intensity ~116 MPa; (BSC 2004 [DIRS 169766], Table 23)) than the stress threshold for initiation of stress corrosion cracking (~285 MPa) (BSC 2004 [DIRS 169985], Section 6.2.1). On this basis, no enhanced degradation due to mechanical loading at the waste package–pallet interfaces is expected. The waste package and drip shield corrosion degradation analyses include the effects of material interfaces in the repository on thermal-hydrologic-geochemical analyses (e.g., FEP 2.1.09.09.0A, Electrochemical Effects in EBS). These include the effects of materials present in the emplacement drift, including the waste package and the drip shield, which are described in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984]) and *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845]).

This treatment of mechanical effects at EBS component interfaces applies to all waste packages.

This FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.23 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 Consequence) (drip shield)
Excluded (Low Probability) (waste package)

Screening Argument:

Seismic-induced rockfall and drift degradation are not treated within this FEP discussion. 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, treated in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

The emplacement drift system design criteria require the drip shield to protect the waste package from rockfalls during postclosure (BSC 2004 [DIRS 170265], Section 3.1.1.4.14.1). *Drip Shield and Waste Package Emplacement Pallet Design Report* (BSC 2004 [DIRS 166879], Section 7.2.1), shows that the deflection of the drip shield due to rockfall is not large enough to contact the waste package. The drip shield will withstand an 11.5 metric ton rockfall without contacting the waste package. The maximum displacement from the 11.5 metric ton rockfall event is 254 mm (BSC 2004 [DIRS 166879], Section 7.2.1) and the minimum gap between the drip shield and waste package outer barrier is about 367 mm (see Section 6.2.14, FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield). Thus, the drip shield provides adequate protection to the waste package from rockfall.

The effects of rockfall on crack initiation in the drip shield are discussed in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* (BSC 2004 [DIRS 169985], Section 6.3.7). These cracks are extremely tight and, with time, become plugged with corrosion products and other mineral precipitates (BSC 2004 [DIRS 169985], Section 6.3.7; FEP 2.1.03.02.0B, Stress Corrosion Cracking (SCC) of Drip Shields). This plugging process limits water transport through the drip shield to negligible amounts, and maintains the functionality of the drip shield. Therefore, rockfall on drip shield is of low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

This document does not address multiple rockfalls. Bounding effects of multiple rockfalls and drift degradation are addressed as part of the seismic consequences in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

Stress fractures in the host rocks and rockfall events may increase the flow of water into the repository. Given that the drip shield is capable of withstanding a rockfall event the drip shield continues to keep water from falling on the waste package until the drip shield fails. Therefore, increased inflow of water related to rockfall is excluded from the TSPA-LA analysis based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

Based on the above discussion, the effects of rockfall on the waste package are excluded from consideration based on low probability of occurrence (i.e., the probability of rockfall impacting the waste package is less than 1 in 10,000 during the first 10,000 years after permanent closure).

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.24 Creep of Metallic Materials in the Waste Package (2.1.07.05.0A)

FEP Description:
Metals used in the waste package may deform by creep processes in response to deviatoric stress or internal void space.

Screening Decision:
Excluded (Low Probability)

Screening Argument:
In the repository, the maximum waste package surface temperature will be less than 300°C even in the low-probability-seismic collapsed-drift scenario (BSC 2004 [DIRS 169565], Figure 6.3-56). Adverse elevated-temperature behavior (i.e., creep deformation or creep fracture) of nickel-based alloys is not expected at temperatures under 650°C (Boyer and Gall 1984 [DIRS 155318], Section 32). No directly relevant data exist for Alloy 22 (UNS N06022) in this temperature regime; however, the melting temperature of Alloy 22 (UNS N06022) is approximately 1,370°C (1,643 K) (Haynes International 1988 [DIRS 101995]) and the maximum surface temperature (less than 300°C or 573 K) is only about 35% of the melting temperature. Creep of Alloy 22 at such low temperatures is not expected. Therefore, high temperature creep has a low probability of occurrence. This treatment of creep of metallic materials in the waste package applies to all waste packages.

External stress, by rock displacements or ground motion for example, may lead to plastic deformations and mechanical damage of the waste package and subsequent leakage of radionuclides. The drip shield is designed to protect the waste package during rockfall and ground motion events (FEP 2.1.07.05.0B, Creep of Metallic Materials in the Drip Shield). Even if mechanical damage were to occur, creep of metallic materials in the waste package will not occur unless an external factor raises the temperature of the waste package surface above 650°C (Boyer and Gall [DIRS 155318], Section 32). In view of the above rationale, this FEP is excluded based on low probability of occurrence.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.25 Creep of Metallic Materials in the Drip Shield (2.1.07.05.0B)

FEP Description:
Metals used in the drip shield may deform by creep processes in response to deviatoric stress.

Screening Decision:
Excluded (Low Probability)

Screening Argument:
In the repository, the maximum waste package surface temperature (because the waste package is the heat source, the waste package surface temperature would be greater than the drip shield surface temperature) will be less than 300°C even in the low-probability-seismic collapsed-drift scenario (BSC 2004 [DIRS 169565], Figure 6.3-56). Literature indicates that between 200°C and 315°C (400°F and 600°F), the deformation of many titanium alloys loaded to yield point does not increase with time (ASM International 1990 [DIRS 144385], p. 626). Given that creep rates decrease at lower temperatures, creep deformation will not occur to any appreciable extent under repository exposure conditions.

External stress, by rock displacements or ground motion for example, may lead to plastic deformations and mechanical damage of the drip shield. Mechanical damage of the drip shield by rockfall is discussed in greater detail under FEP 2.1.07.01.0A, Rockfall. This FEP discussion also provides relevant references discussing the issue in greater detail. In addition, the Emplacement Drift System design criteria require that the drip shield protect the waste package from rockfalls during post closure (BSC 2004 [DIRS 170265], Section 3.1.1.4.14.1). Mechanical damage of the drip shield during seismic events is discussed 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 in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]).

In view of the above rationale, this FEP is excluded based on low probability of occurrence.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.26 Volume Increase of Corrosion Products Impacts Waste Package (2.1.09.03.0B)

FEP Description:

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

Screening Decision:

Excluded (Low Probability)

Screening Argument:

In general, corrosion products have greater volume than the metal. When the corrosion products form in a tightly confined space, their increase in volume generates swelling pressures that could lead to mechanical damage of the surrounding material. Since the current design precludes the use of shrink fitting the outer and inner cylinder components, mechanical damage to the Alloy 22 (UNS N06022) waste package outer barrier due to the pressure exerted by the corrosion product (Cr_2O_3) of the inner shell (Stainless Steel Type 316) will not occur. Current waste package designs require the radial gap between the inner vessel and the outer barrier to be at least 1 mm and could be a maximum of 5 mm (BSC 2004 [DIRS 169480]). Analyses cited in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169985], Section 6.4.2), indicate that for chromia (Cr_2O_3) scale-forming alloys (e.g., Alloy 22 and Stainless Steel Type 316), even under very conservative assumptions, the growth of corrosion product will not exceed 93 μm after 10,000 years. This oxide layer is not thick enough to produce enough pressure to cause mechanical damage to the waste package outer barrier or inner vessel. In the current design of waste package and engineered barrier system in the emplacement drift (BSC 2004 [DIRS 168489]), there is no possibility of forming such a tightly confined space such that the swelling corrosion products could cause mechanical damage to the Alloy 22 (UNS N06022) outer barrier. Therefore, waste package damage from swelling corrosion products is excluded based on low probability of occurrence under the exposure conditions in the repository.

A related FEP (2.1.09.03.0A, Volume increase of corrosion products impacts cladding) is discussed in *Clad Degradation - FEPs Screening Arguments* (BSC 2004 [DIRS 170019]).

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.27 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:

Due to the large distances involved, it is reasonable to consider electrochemical effects between materials in the drift and more distant metallic materials in the natural system to be less important to waste package materials degradation than the electrochemical interactions between materials or components within the drift. Such long-range interactions are more appropriate for consideration in modeling processes, such as radionuclide transport away from the repository, rather than in consideration of relatively local phenomena, such as waste package or drip shield degradation.

The waste package and the drip shield, as designed and emplaced, come in contact with very few other EBS components. For example, the waste package is designed to rest on a pallet constructed of Alloy 22 (UNS N06022), and is designed to keep the waste package from contacting other dissimilar metals (BSC 2004 [DIRS 168489]). The pallet is also designed to keep the waste package supported in a horizontal position away from the invert and ground support under nonseismic scenarios (BSC 2004 [DIRS 166879], Section 7.2.2). Similarly, the drip shields are designed to contact no other material except the Alloy 22 feet, which are attached to the bottom of the drip shields. These feet are in contact with the invert, which is covered by crushed tuff as ballast.

The TSPA-LA waste package design (BSC 2004 [DIRS 168489]) includes an outer barrier of Alloy 22 (UNS N06022) over a Stainless Steel Type 316 inner vessel. In addition, a titanium drip shield is placed over the waste packages. Although the Stainless Steel Type 316 inner vessel provides structural stability to the Alloy 22 (UNS N06022) outer barrier, no other performance credit is taken for the waste package inner vessel. The corrosion potentials of Alloy 22 (UNS N06022) and Stainless Steel Type 316 are very close to each other under similar exposure conditions (e.g., ASM International [DIRS 103753], p. 557, Figure 10) with Alloy 22 (UNS N06022) slightly more noble than Stainless Steel Type 316. After breach of the Alloy 22 (UNS N06022) waste package outer barrier, electrochemical coupling of the Alloy 22 (UNS N06022) waste package outer barrier with the Stainless Steel Type 316 waste package inner vessel could occur. Due to the similarity in corrosion potential of Alloy 22 (UNS N06022) and Stainless Steel Type 316, any enhanced degradation of either material due to galvanic interaction would be negligible. Therefore, electrochemical coupling of the Alloy 22 (UNS N06022) waste package outer barrier and the Stainless Steel Type 316 waste package inner vessel is of low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

The possibility of the drip shield contacting the waste package as a result of mechanical damage caused by rockfall was considered. This possibility was, however, addressed and excluded as discussed in FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield. The only contact between Titanium Grade 7 and Alloy 22 (UNS N06022) occurs at the bottom of the drip shields where Alloy 22 (UNS N06022) feet are attached to prevent contact between titanium and the invert. The choice of Alloy 22 (UNS N06022) for the feet was based on similarity of the two materials in the electrochemical series (ASM International 1987 [DIRS 103753], p. 557, Figure 10), which indicates any enhanced degradation of either material due to galvanic interaction would be negligible. Therefore, electrochemical coupling of the Alloy 22 (UNS N06022) waste package outer barrier and the Titanium Grade 7 drip shield is of low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.28 Thermal Sensitization of Waste Packages (2.1.11.06.0A)

FEP Description:
Phase changes in waste package materials can result from long-term storage at moderately hot temperatures in the repository. Stress corrosion cracking (SCC), intergranular corrosion, or mechanical degradation may ensue.

Screening Decision:
Excluded (Low Probability)

Screening Argument:
Alloy 22 (UNS N06022) could to be subject to "aging" and phase instability when exposed to elevated temperatures that are well above the anticipated maximums expected in the repository (BSC 2004 [DIRS 171924], Section 6.3). The processes involve precipitation of different secondary phases and restructuring of the microstructure. The affected material exhibits increased brittleness and decreased resistance to corrosion, especially to localized corrosion and SCC (BSC 2004 [DIRS 171924], Section 1.1).

Before waste loading, the waste packages (base-metal and fabrication welds) are fully solution annealed (Plinski 2001 [DIRS 156800], Section 8.1.7). After waste loading, the closure lids are welded onto the waste package (Plinski 2001 [DIRS 156800], Section 8.1.8). No effects of aging and phase instability on the Alloy 22 waste package outer barrier would be observed even if the waste package were maintained at less than 300ºC for a period of 500 years followed by temperatures less than 200°C for a period of 9,500 years (BSC 2004 [DIRS 171924], Section 8; BSC 2004 [DIRS 169984], Section 6.4.6). These thermal exposure conditions bound all repository-relevant thermal exposure conditions (BSC 2004 [DIRS 171924], Section 8).

The closure lid welds cannot be solution annealed without risking damage to the waste form. Therefore, the closure welds of the waste package outer barrier could be more prone to thermal aging and phase instability than the base metal under long-term thermal exposure in the repository (BSC 2004 [DIRS 169984], Section 6.4.6). Analyses conducted in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2004 [DIRS 169984], Section 6.4.6) studied the effect of thermal aging on corrosion of Alloy 22. Three metallurgical conditions of Alloy 22 (UNS N06022) were studied using the multiple crevice assembly samples: mill annealed, as-welded, and as-welded plus thermally aged (at 700°C for 173 hours). The samples were tested in 5 M $CaCl_2$ solutions with and without $Ca(NO_3)_2$ additions (up to 0.5 m) at test temperatures varying from 45°C to 120°C. Comparison of the calculated corrosion rates of the mill annealed, as-welded, and as-welded plus thermally aged samples showed no apparent enhancement of the corrosion rate due to the presence of welds or thermal aging of the welded samples for the tested conditions.

Based on this analysis, insignificant aging and phase stability will occur under the thermal conditions expected in the repository (BSC 2004 [DIRS 169984], Section 6.4.6) and the corrosion performance of the waste package outer barrier material is not expected to be affected by aging and phase stability in the repository. This treatment of thermal sensitization of waste packages applies to all waste packages. Thermal sensitization of waste packages is excluded on the basis of low probability of occurrence.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.29 Thermal Sensitization of Drip Shields (2.1.11.06.0B)

FEP Description:

Phase changes in drip shield materials can result from long-term storage at moderately hot temperatures in the repository. Stress corrosion cracking (SCC), intergranular corrosion, or mechanical degradation may ensue.

Screening Decision:

Excluded (Low Probability)

Screening Argument:

Aging and phase stability of the drip shield is considered in Section 6.5.3 of *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2004 [DIRS 169845]) that states Titanium Grade 7 is a stabilized alpha (α) phase alloy and possesses outstanding phase stability. While Titanium Grade 7 contains small amounts of alloying elements (DTN: MO0003RIB00073.000 [DIRS 152926]), most notably palladium, it is essentially a pure titanium alloy that has no capability of forming intermetallic compounds under the thermal exposure conditions in the repository.

The solubility of palladium in Titanium Grade 7 is about 1 weight percent at 400°C. The nominal concentration of palladium in Titanium Grade 7 is well below the solubility limit at this temperature (Gdowski 1997 [DIRS 102789], pp. 1 to 8). Titanium–palladium intermetallic compounds capable of being formed in the Ti-Pd system have not been reported to occur in Titanium Grade 7 with normal heat treatments. Hua et al. (2002 [DIRS 160670]) tested both the base metal and welded metal of Titanium Grade 7 in a highly concentrated basic environment at 60°C, 70°C, 80°C, 90°C, 100°C, and 105°C for up to eight weeks (Hua et al. 2002 [DIRS 160670]; Hua and Gordon 2003 [DIRS 163111]). No difference in weight loss and, therefore, in corrosion rate was observed between the base metal and welds. The boundaries between the welds and heat-affected zone and between the heat-affected zone and base metal were not visibly attacked. Therefore, based on the above experimental evidence, thermal sensitization of the drip shield can be excluded based on low probability of occurrence. Also, because thermal sensitization of the drip shield will not occur, stress corrosion cracking (SCC), intergranular corrosion, or mechanical degradation will not ensue due to thermal sensitization of the drip shield.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.30 Thermal Expansion/Stress of In-Drift EBS Components (2.1.11.07.0A)

FEP Description:
Repository heat at Yucca Mountain could result in thermally induced stress changes that would affect the mechanical and chemical evolution of the repository. These stress changes could affect the EBS components, thus causing the formation of pathways for groundwater flow through the EBS or altering and/or enhancing existing pathways. Relevant processes include changes in physical properties of the drip shields, waste packages, pallet, and invert.

Screening Decision:
Excluded (Low Consequence)

Screening Argument:
The coefficient of thermal expansion for Stainless Steel Type 316L (an analogue for the Stainless Steel Type 316 used for the waste package inner shell) is larger than the coefficient of thermal expansion for Alloy 22 (UNS N06022). Thus, changes in temperature could lead to contact stresses between the waste package barriers. In *Waste Package Outer Barrier Stresses Due to Thermal Expansion with Various Barrier Gap Sizes* (BSC 2001 [DIRS 152655]), the maximum tangential stress at the waste package outer barrier inner and outer surfaces were evaluated for several waste package types (21-PWR, 44-BWR, 12-PWR Long, 5 DHLW/DOE SNF – Short, 2-MCO/2-DHLW, and naval SNF Long) as a function of temperature and barrier gap size (difference in radius of the two barriers evaluated at room temperature (BSC 2001 [DIRS 152655], Section 5.3). An earlier calculation (BSC 2001 [DIRS 154004]) using a barrier gap size of zero, showed that under thermal expansion loading, tangential stresses are significantly higher than radial stresses (BSC 2001 [DIRS 152655], Section 1.0). The conclusion of these studies was that a barrier gap size of at least 1 mm would result in no tangential stresses due to thermal expansion. Current waste package designs require the radial gap between the waste package inner vessel and the waste package outer barrier to be at least 1 mm and could be a maximum of 5 mm (BSC 2004 [DIRS 169480]).

Waste Package Operation Fabrication Process Report (Plinski 2001 [DIRS 156800], Section 8.1.8) requires a loose fit between the outer barrier (Alloy 22 [UNS N06022]) and the inner vessel (Stainless Steel Type 316) to accommodate the differing thermal expansion coefficients. Typical waste package designs also require large longitudinal barrier gaps (~30 mm) (BSC 2004 [DIRS 169472], Table 1A). Therefore, although thermal expansion of waste package components occurs, no significant stresses due to differing thermal expansion between the barriers develop. This FEP is excluded for the waste packages based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

In the LA drip shield design (BSC 2003 [DIRS 161519]), the drip shield connectors are designed to allow for thermal expansion with no effect on drip shield performance. The drip shield segments are interlocked with a significant amount of freedom to expand and still maintain their intended purpose. The space between the drip shield and waste package is large enough to accommodate deflection due to rockfall (see FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield). The space required for thermal expansion is very small by comparison. Therefore, this FEP can be excluded for the drip shields based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.31 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:

The materials selected for waste package outer barrier and the drip shield are highly corrosion resistant materials (BSC 2004 [DIRS 169845], Section 6; BSC 2004 [DIRS 169984], Section 6). These form a thin, highly protective oxide layer (passive film) that protects the materials from further corrosion. As a result, limited corrosion of these materials leads to negligible gas generation. In addition, a drift in the unsaturated zone of the repository is expected to be connected to the atmosphere and to be operating under oxidizing conditions. Therefore, any gases generated by metal corrosion would escape from the drifts.

Hydrogen generated at cathodic sites in a corroding metal may migrate into the metal and potentially form hydride phases. Hydrogen incorporation could lead to degradation of the mechanical properties of the material and render it susceptible to cracking even in the absence of the formation of hydride phases. Hydrogen-induced cracking (HIC) is one of the results that may happen from the combined action of hydrogen and residual or sustained applied tensile stresses.

As discussed in FEP 2.1.03.04.0A, Hydride Cracking of Waste Packages, HIC of the waste package outer barrier (Alloy 22 (UNS N06022)) is not considered to be an effective degradation mechanism under repository-relevant exposure conditions based on analyses of handbook data (ASM International. 1987 [DIRS 103753], pp. 169, 557, 650-652). As discussed in FEP 2.1.03.04.0B, Hydride Cracking of Drip Shields, HIC of the drip shields is not considered to be an effective degradation mechanism under repository-relevant exposure conditions based on analyses of handbook data (Schutz and Thomas 1987 [DIRS 144302]) and Project analyses (BSC 2004 [DIRS 169847], Sections 6.1.2, 6.1.3, 6.1.4, 6.1.5, 6.3.2). This treatment of gas generation from waste package corrosion applies to all waste packages. Overall, even though gas generation is possible, it is of low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:
N/A

Supporting Reports:
N/A

6.2.32 Radiolysis (2.1.13.01.0A)

FEP Description:
Alpha, beta, gamma, and neutron irradiation of water can cause disassociation of molecules, leading to gas production and changes in chemical conditions (potential, pH, and concentration of reactive radicals).

Screening Decision:
Excluded (Low Consequence)

Screening Argument:
Alpha, beta, gamma, and neutron irradiation of water leads to formation of highly reactive excited and ionized species, which in turn can undergo various reactions. In pure water, the final products are hydrogen and oxidants. This section addresses only the effects of radiolysis of water on degradation of the waste package outer barrier. Radiolysis effects on waste form are addressed in *Waste-Form Features, Events, and Processes* (BSC 2004 [DIRS 170020]). Chemical effects from radiolysis are addressed in *Engineered Barrier System Features, Events, and Processes* (BSC 2004 [DIRS 169898]). Gamma radiation is the dominant contributor to dose rate at the waste package surface because the neutron dose rate is relatively small by comparison (BSC 2004 [DIRS 169593], Section 6.3) and alpha and beta radiation will not penetrate the container. Analogous to the discussion in FEP 2.1.12.03.0A, Gas Generation (H_2) from Waste Package Corrosion, a drift in the unsaturated zone of the Yucca Mountain repository is expected to be operating under oxidizing conditions and will be connected to the atmosphere allowing any gases generated by radiolysis to escape from the drifts (Shoesmith and King 1998 [DIRS 112178], p. 5). The effects of radiation on the corrosion of the waste package materials differ depending on the amount of liquid present on their surfaces (i.e., humid air or aqueous

conditions). Under humid air conditions, a thin film of liquid forms possibly containing trace constituents (e.g., dissolved gases). Irradiation of these films could lead to acidic conditions and to enhanced corrosion rates. Under aqueous conditions (bulk solutions), anodic shifts in the open circuit potential of stainless steel in gamma-irradiated solutions have been experimentally observed (BSC 2004 [DIRS 169845], Section 6.7.1). These shifts in potential have been shown to be due to the formation of hydrogen peroxide (BSC 2004 [DIRS 169845], Section 6.7.1).

Calculations of the expected radiation levels at the surface of the waste package have been performed in (BSC 2004 [DIRS 169593], Section 6.4.2). For a bounding-case waste package containing 21-PWR spent fuel assemblies (80 GWd/MTU burnup, and 5-year decay), the maximum surface radiation level was calculated to be about 1,170 rad/hr at the waste package outer barrier surface and 1,650 rad/hr at the outer lid of the waste package (BSC 2004 [DIRS 169593], Table 6.4-3). During the 50-year ventilation period, an aqueous or humid air environment will not exist and, as a result, radiolysis is not anticipated. After 50 years, the maximum surface radiation level decreases to about 100 rad/hr for the waste package outer barrier surface and 80 rad/hr for the outer lid (BSC 2004 [DIRS 169593], Figure 6.4-1). One hundred years after emplacement, the maximum calculated levels reduce to about 35 rad/hr for the waste package outer barrier surface and 20 rad/hr for the outer lid region (BSC 2004 [DIRS 169593], Figure 6.4-1). The bounding radiation levels for the highest burnup spent fuel are well below the levels at which the effect of radiation has been observed.

Although there is little information available in the literature on the effects of radiation on Alloy 22 (UNS N06022), some data are available on the corrosion of Alloy C-4, which is compositionally similar to Alloy 22. Gamma irradiation in aggressive $MgCl_2$ brines showed that below ~100 rad/hour (Shoesmith and King 1998 [DIRS 112178], p. 29) irradiation has no observable influence on the corrosion behavior of Alloy C-4. In this same environment, it was found, even at dose rates above 1,000 rad/hr, only a minor enhancement of film growth rates on Titanium Grade 7 was observed and passivity was not threatened (Shoesmith and King 1998 [DIRS 112178], p. 30). Based on this data it is concluded that, even in aggressive $MgCl_2$ brines, radiation levels in the repository are not high enough to result in an enhancement of corrosion processes on Alloy 22 (UNS N06022) or Titanium Grade 7. On this basis, radiolysis is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

6.2.33 Radiation Damage in EBS (2.1.13.02.0A)

FEP Description:

Radiolysis due to the alpha, beta, gamma-ray, and neutron irradiation of water could result in enhancement of the radionuclide migration from the surface of a degraded waste form into groundwater. When radionuclides decay, the emitted high-energy particle could result in the production of radicals in the water or air surrounding the spent nuclear fuel. If these radicals migrate (diffuse) to the surface of the fuel, they may then enhance the degradation/corrosion rate of the fuel (UO_2). This effect would increase the dissolution rate for radionuclides from the fuel material (fuel matrix) into the groundwater. Strong radiation fields could lead to radiation damage to the waste forms (CSNF, DSNF, DHLW), waste packages, drip shields, seals, and surrounding rock.

Screening Decision:

Excluded (Low Consequence)

Screening Argument:

The enhanced degradation or corrosion of the fuel is covered in *Waste-Form Features, Events, and Processes* (BSC 2004 [DIRS 170020]). The cumulative neutron fluence at the waste package emplacement pallet as a function of years after emplacement is determined in *Dose Rate Calculation for 21-PWR Waste Package* (BSC 2004 [DIRS 169593], Section 6.4.1). The calculation uses the design-basis source term (60 GWd/MTU burnup, 10-year decay time) (BSC 2004 [DIRS 169593], Section 5.1.4). At 290 years and 340 years after emplacement (the two longest times reported), the cumulative neutron fluence at the emplacement pallet location was determined to be 3.78×10^{14} n/cm^2 and 3.90×10^{14} n/cm^2, respectively (where n stands for neutrons) (BSC 2004 [DIRS 169593], Table 6.4-1). The cumulative neutron fluence at the emplacement pallet location can be approximated to be increasing at a rate of $(1.2 \times 10^{13})/50$ n/cm^2/yr. Using this approximation, the cumulative neutron fluence at the emplacement pallet location after 10,000 years would be about 3×10^{15} n/cm^2. Due to the close proximity of the waste package and the waste package emplacement pallet, it is reasonable to equate the cumulative neutron fluence at the emplacement pallet location to the cumulative neutron fluence at the waste package location. *Nuclear Engineering Handbook* (Etherington 1958 [DIRS 164789], p. 10-107) identifies neutron fluence levels below which there is no change in the mechanical properties of Stainless Steel Type 316 (5×10^{19} n/cm^2), nickel (10^{20} n/cm^2), and molybdenum (10^{20} n/cm^2). Because Alloy 22 is a nickel-based-molybdenum-containing alloy, there is no reason that the neutron fluence threshold for Alloy 22 should be more than four orders of magnitude lower than for these materials (as it would have to be in order for the estimated neutron fluence to exceed the neutron fluence threshold for changes in mechanical properties). Therefore, while it is likely that some radiation damage occurs, because the estimated neutron fluence is below the neutron fluence threshold by a considerable margin, this FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

Furthermore, the Waste Package Materials Performance Peer Review Panel addressed the possibility of radiation damage in the repository in their final report (Beavers et al. 2002 [DIRS 158781], Section 3.10). They stated that the waste package will be subjected to a flux of neutrons and gamma rays from the stored radioactive waste (Beavers et al. 2002

[DIRS 158781]). These fluxes could cause the following: 1) neutrons could produce atomic displacement damage in the metal, 2) neutrons could produce atomic displacement damage and gamma rays will cause electron-hole pairs in the passive film, and 3) gamma rays could cause radiolysis of the surrounding environment (Beavers et al. 2002 [DIRS 158781], Section 3.10). The peak neutron flux has been calculated to be about 5×10^4 n/cm^2·sec) in the repository environment (Beavers et al. 2002 [DIRS 158781], Section 3.10). The total neutron fluence, taking the most conservative estimate with no nuclear decay of the waste, will be about 1.5×10^{16} n/cm^2 in 10,000 years in the repository environment (Beavers et al. 2002 [DIRS 158781], Section 3.10). The report concluded that there is no evidence to suggest that radiation damage to the waste package will alter its mechanical properties (Beavers et al. 2002 [DIRS 158781], Section 3.10). The drip shield is located farther away from the source of radiation (the waste form) than is the waste package barrier. On this basis, the drip shield material will be subject to less radiation damage than the waste package (i.e., radiation damage is of even less consequence to drip shield performance than it is to waste package performance).

Therefore, the only potential effect of radiation will be the change in external environment due to groundwater radiolysis (e.g., ASM International 1987 [DIRS 103753], pp. 971 to 974), which is addressed in FEP 2.1.13.01.0A, Radiolysis.

Based on the above rationale, this FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

TSPA Disposition:

N/A

Supporting Reports:

N/A

7. CONCLUSIONS

The analyses documented in this report are for the TSPA-LA design in which a drip shield is placed over the waste package (BSC 2004 [DIRS 168489]).

Thirty-three FEPs relevant to waste package and drip shield degradation processes were screened against criteria presented in Section 4.2.2. The results of this screening process are documented in the screening arguments and the TSPA disposition statements in Section 6.2. The screening basis for each FEP is summarized in Table 7-1. This table shows the FEP number, name, screening decision (included or excluded in TSPA-LA), and basis for the screening decision (i.e., low probability of occurrence or low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment).

Table 7-1. Summary of Waste Package FEPs

FEP Number	FEP Name	Screening Decision and Basis	Addressed in Section
1.1.03.01.0A	Error in Waste Emplacement	Excluded (Low Consequence)	6.2.1
2.1.03.01.0A	General Corrosion of Waste Packages	Included	6.2.2
2.1.03.01.0B	General Corrosion of Drip Shields	Included	6.2.3
2.1.03.02.0A	Stress Corrosion Cracking (SCC) of Waste Packages	Included	6.2.4
2.1.03.02.0B	Stress Corrosion Cracking (SCC) of Drip Shields	Excluded (Low Consequence)	6.2.5
2.1.03.03.0A	Localized Corrosion of Waste Packages	Included	6.2.6
2.1.03.03.0B	Localized Corrosion of Drip Shields	Excluded (Low Consequence)	6.2.7
2.1.03.04.0A	Hydride Cracking of Waste Packages	Excluded (Low Consequence)	6.2.8
2.1.03.04.0B	Hydride Cracking of Drip Shields	Excluded (Low Consequence)	6.2.9
2.1.03.05.0A	Microbially Influenced Corrosion (MIC) of Waste Package	Included	6.2.10
2.1.03.05.0B	Microbially Influenced Corrosion (MIC) of Drip Shields	Excluded (Low Consequence)	6.2.11
2.1.03.06.0A	Internal Corrosion of Waste Packages prior to breach	Excluded (Low Consequence)	6.2.12
2.1.03.07.0A	Mechanical Impact on Waste Package	Excluded (Low Consequence and Low Probability)	6.2.13
2.1.03.07.0B	Mechanical Impact on Drip Shield	Excluded (Low Consequence)	6.2.14
2.1.03.08.0A	Early failure of Waste Packages	Included	6.2.15
2.1.03.08.0B	Early failure of Drip Shields	Excluded (Low Consequence)	6.2.16
2.1.03.09.0A	Copper corrosion in EBS	Excluded (Low Consequence)	6.2.17
2.1.03.10.0A	Advection of liquids and solids through cracks in the waste package	Excluded (Low Consequence)	6.2.18
2.1.03.10.0B	Advection of liquids and solids through cracks in the drip shield	Excluded (Low Consequence)	6.2.19
2.1.03.11.0A	Physical form of Waste Package and Drip Shield	Included	6.2.20
2.1.06.06.0B	Oxygen embrittlement of Drip Shields	Excluded (Low Probability)	6.2.21
2.1.06.07.0B	Mechanical effects at EBS component interfaces	Excluded (Low Consequence)	6.2.22

Table 7-1. Summary of Waste Package FEPs (Continued)

FEP Number	FEP Name	Screening Decision and Basis	Addressed in Section
2.1.07.01.0A	Rockfall	Excluded (Low Consequence) and Excluded (Low Probability)	6.2.23
2.1.07.05.0A	Creep of metallic materials in the Waste Package	Excluded (Low Probability)	6.2.24
2.1.07.05.0B	Creep of metallic materials in the Drip Shield	Excluded (Low Probability)	6.2.25
2.1.09.03.0B	Volume increase of Corrosion products impacts Waste Package	Excluded (Low Probability)	6.2.26
2.1.09.09.0A	Electrochemical effects in EBS	Excluded (Low Consequence)	6.2.27
2.1.11.06.0A	Thermal sensitization of Waste Packages	Excluded (Low Probability)	6.2.28
2.1.11.06.0B	Thermal sensitization of Drip Shields	Excluded (Low Probability)	6.2.29
2.1.11.07.0A	Thermal expansion/stress of In-Drift EBS components	Excluded (Low Consequence)	6.2.30
2.1.12.03.0A	Gas generation (H_2) from Waste Package corrosion	Excluded (Low Consequence)	6.2.31
2.1.13.01.0A	Radiolysis	Excluded (Low Consequence)	6.2.32
2.1.13.02.0A	Radiation damage in EBS	Excluded (Low Consequence)	6.2.33

The conclusions from this document (FEP screening decision, TSPA disposition for included FEPs, or screening argument for excluded FEPs), along with any modifications to the FEP list, FEP names, and/or FEP 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 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 waste package and drip shield 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 is 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* (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 waste package and drip shield-related FEPs (Table 1-1). This list of FEPs includes those related to the degradation, deterioration, or alteration of engineered barriers. The list of waste package and drip shield-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. Table 7-1 provides a list of excluded waste package and drip shield-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.

8. INPUTS AND REFERENCES

8.1 DOCUMENTS CITED

144385 ASM International 1990. *Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, Specific Metals and Alloys*. Volume 2 of Metals Handbook. 10th Edition. Page 666. Materials Park, Ohio: American Society for Metals. TIC: 239807.

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

160352 Asphahani, A.I. 1978. "Hydrogen Cracking of Nickel-Base Alloys." *Hydrogen in Metals, Proceedings of the 2nd International Congress, Paris, France, 6-10 June, 1977*. 1, Paper 3C2. New York, New York: Pergamon. ACC: NNA.19910419.0006.

158781 Beavers, J.A.; Devine, T.M., Jr.; Frankel, G.S.; Jones, R.H.; Kelly, R.G.; Latanision, R.M.; and Payer, J.H. 2002. *Final Report, Waste Package Materials Performance Peer Review Panel, February 28, 2002*. Las Vegas, Nevada: Waste Package Materials Performance Peer Review Panel. ACC: MOL.20020614.0035.

155318 Boyer, H.E. and Gall, T.L., eds. 1984. *Metals Handbook*. Desk Edition. 10th Printing 1997. Metals Park, Ohio: American Society for Metals. TIC: 250192.

159836 Brossia, C.S.; Browning, L.; Dunn, D.S.; Moghissi, O.C.; Pensado, O.; and Yang, L. 2001. *Effect of Environment on the Corrosion of Waste Package and Drip Shield Materials*. CNWRA 2001-003. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. TIC: 252324.

154441 BSC (Bechtel SAIC Company) 2001. *Committed Materials in Repository Drifts*. CAL-GCS-GE-000002 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010412.0157.

152248 BSC 2001. *Flow of Water and Pooling in a Waste Package*. ANL-EBS-MD-000055 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010604.0218.

156807 BSC 2001. *Plugging of Stress Corrosion Cracks by Precipitates*. CAL-EBS-MD-000017 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011010.0168.

154004 BSC 2001. *Waste Package Barrier Stresses Due to Thermal Expansion*. CAL-EBS-ME-000008 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011211.0149.

152655 BSC 2001. *Waste Package Outer Barrier Stress Due to Thermal Expansion with Various Barrier Gap Sizes*. CAL-EBS-ME-000011 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011212.0222.

161519 BSC 2003. *Interlocking Drip Shield*. 000-MW0-TED0-00101-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20030205.0001.

171924 BSC 2004. *Aging and Phase Stability of Waste Package Outer Barrier*. ANL-EBS-MD-000002 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041005.0003.

170024 BSC 2004. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure*. CAL-EBS-MD-000030 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040913.0006.

168030 BSC 2004. *Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada*. ANL-CRW-GS-000003 REV 00, with 1 errata. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20000510.0175; DOC.20040223.0007.

170019 BSC 2004. *Clad Degradation – FEPs Screening Arguments*. ANL-WIS-MD-000008, Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company.

169766 BSC 2004. *Commercial SNF Waste Package Design Report*. 000-00C-DSU0-02800-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040709.0001.

168489 BSC 2004. *D&E / PA/C IED Emplacement Drift Configuration and Environment*. 800-IED-MGR0-00201-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040326.0001.

169480 BSC 2004. *D&E / PA/C IED Typical Waste Package Components Assembly*. 800-IED-WIS0-00201-000-00E. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040517.0007.

169472 BSC 2004. *D&E/PA/C IED Typical Waste Package Components Assembly*. 800-IED-WIS0-00202-000-00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040517.0008.

170803 BSC 2004. *DOE and Commercial Waste Package System Description Document*. 000-3YD-DS00-00100-000-004. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040720.0009.

169593 BSC 2004. *Dose Rate Calculation for 21-PWR Waste Package*. 000-00C-DSU0-01800-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040607.0001.

166879 BSC 2004. *Drip Shield and Waste Package Emplacement Pallet Design Report*. 000-00C-SSE0-00100-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040225.0016.

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.

170017 BSC 2004. *Features, Events, and Processes: Disruptive Events*. ANL-WIS-MD-000005, Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company.

169845 BSC 2004. *General Corrosion and Localized Corrosion of the Drip Shield*. ANL-EBS-MD-000004 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040921.0002.

169984 BSC 2004. *General Corrosion and Localized Corrosion of Waste Package Outer Barrier*. ANL-EBS-MD-000003 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041004.0001.

169847 BSC 2004. *Hydrogen-Induced Cracking of the Drip Shield*. ANL-EBS-MD-000006 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040909.0004.

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

168556 BSC 2004. *Screening Analysis for Criticality Features, Events, and Processes for License Application*. ANL-EBS-NU-000008, Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company.

169985 BSC 2004. *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material*. ANL-EBS-MD-000005, Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company.

170265 BSC 2004. *Subsurface Facility Description Document*. 800-3YD-MGR0-00100-000-004. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040630.0006.

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.

169996 BSC 2004. *WAPDEG Analysis of Waste Package and Drip Shield Degradation*. ANL-EBS-PA-000001 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041004.0005.

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.

105212 Cowan, J.C. and Weintritt, D.J. 1976. *Water-Formed Scale Deposits*. Houston, Texas: Gulf Publishing Company. TIC: 245620.

150823 CRWMS M&O 2000. *Design Analysis for the Defense High-Level Waste Disposal Container*. ANL-DDC-ME-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000627.0254.

123881 CRWMS M&O 2000. *Engineering Files for Site Recommendation*. TDR-WHS-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000607.0232.

152016 CRWMS M&O 2001. *Water Distribution and Removal Model*. ANL-EBS-MD-000032 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010214.0031.

162213 Dunn, D.S. and Brossia, C.S. 2002. "Assessment of Passive and Localized Corrosion Processes for Alloy 22 as a High-Level Nuclear Waste Container Material." *Corrosion/2002, 57th Annual Conference & Exposition, April 7-11, 2002, Denver, Colorado*. Paper No. 02548. Houston, Texas: NACE International. TIC: 254579.

164789 Etherington, H., ed. 1958. *Nuclear Engineering Handbook*. 1st Edition. New York, New York: McGraw-Hill. TIC: 231758.

100859 Gdowski, G.E. 1991. *Survey of Degradation Modes of Four Nickel-Chromium-Molybdenum Alloys*. UCRL-ID-108330. Livermore, California: Lawrence Livermore National Laboratory. ACC: NNA.19910521.0010.

102789 Gdowski, G.E. 1997. *Degradation Mode Survey Candidate Titanium - Base Alloys for Yucca Mountain Project Waste Package Materials*. UCRL-ID-121191, Rev. 1. Livermore, California: Lawrence Livermore National Laboratory. ACC: MOL.19980120.0053.

101995 Haynes International 1988. *Hastelloy Alloy C-22*. Kokomo, Indiana: Haynes International. TIC: 239938.

163111 Hua, F. and Gordon, G. 2003. "On Apparent Bi-Linear Corrosion Rate Behavior of Ti Grade 7 in Basic Saturated Water (BSW-12) Below and Above 80°C." *Corrosion/2003, 58th Annual Conference & Exposition, March 16-20, 2003, San Diego, California*. Paper No. 03687. Houston, Texas: NACE International. TIC: 254248.

160670 Hua, F.; Sarver, J.; Jevec, J.; and Gordon, G. 2002. "General Corrosion Studies of Candidate Container Materials in Environments Relevant to Nuclear Waste Repository." *Corrosion/2002, 57th Annual Conference & Exposition, April 7-11, 2002, Denver, Colorado*. Paper No. 02530. Houston, Texas: NACE International. TIC: 252067.

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.

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.

159833 Lorenzo de Mele, M.F. and Cortizo, M.C. 2000. "Electrochemical Behaviour of Titanium in Fluoride-Containing Saliva." Journal of Applied Electrochemistry, 30, (1), 95-100. Dordrecht, The Netherlands: Kluwer Academic Publishers. TIC: 253126.

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

163274 NRC (U.S. Nuclear Regulatory Commission) 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.

165536 Pan, Y.-M.; Brossia, C.S.; Cragnolino, G.A.; Dunn, D.S.; Gute, G.D.; and Yang, L. 2002. *Stress Corrosion Cracking and Hydrogen Embrittlement of Container and Drip Shield Materials*. CNWRA 2003-02. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. TIC: 254055.

156800 Plinski, M.J. 2001. *Waste Package Operations Fabrication Process Report*. TDR-EBS-ND-000003 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011003.0025.

165537 Pulvirenti, A.L.; Needham, K.M.; Adel-Hadadi, M.A.; Marks, C.R.; Gorman, J.A.; and Barkatt, A. 2002. "Effects of Lead, Mercury, and Reduced Sulfur Species on the Corrosion of Alloy 22 in Concentrated Groundwaters as a Function of pH and Temperature." *Scientific Basis for Nuclear Waste Management XXV, Symposium held November 26-29, 2001, Boston, Massachusetts*. McGrail, B.P. and Cragnolino, G.A., eds. 713, 89-95. Warrendale, Pennsylvania: Materials Research Society. TIC: 248663.

159370 Revie, R.W., ed. 2000. *Uhlig's Corrosion Handbook*. 2nd Edition. New York, New York: John Wiley & Sons. TIC: 248360.

154465 Sakai, T.; Aoki, K.; Shigemitsu, T.; and Kishi, Y. 1992. "Effect of Lead Water Chemistry on Oxide Thin Film of Alloy 600." *Corrosion*, *48*, (9), 745-750. Houston, Texas: National Association of Corrosion Engineers. TIC: 249773.

144302 Schutz, R.W. and Thomas, D.E. 1987. "Corrosion of Titanium and Titanium Alloys." In *Corrosion*, Volume 13, Pages 669-706 of *Metals Handbook*. 9th Edition. Metals Park, Ohio: ASM International. TIC: 209807.

151179 Shoesmith, D.W. and Ikeda, B.M. 1997. *The Resistance of Titanium to Pitting, Microbially Induced Corrosion and Corrosion in Unsaturated Conditions*. AECL-11709. Pinawa, Manitoba, Canada: Whiteshell Laboratories. TIC: 236226.

112178 Shoesmith, D.W. and King, F. 1998. *The Effects of Gamma Radiation on the Corrosion of Candidate Materials for the Fabrication of Nuclear Waste Packages*. AECL-11999. Pinawa, Manitoba, Canada: Atomic Energy of Canada Limited. ACC: MOL.19990311.0212.

117892 Shoesmith, D.W.; Ikeda, B.M.; Bailey, M.G.; Quinn, M.J.; and LeNeveu, D.M. 1995. *A Model for Predicting the Lifetimes of Grade-2 Titanium Nuclear Waste Containers*. AECL-10973. Pinawa, Manitoba,Canada: Atomic Energy of Canada Limited. TIC: 226419.

120498 Thomas, J.G.N. 1994. “The Mechanism of Corrosion Prevention by Inhibitors.” In *Corrosion Control*, Volume 2, Chapter 17.3 of *Corrosion*. 3rd Edition. Reprinted 1998. Sheir, L.L.; Jarman, R.A.; and Burstein. Woburn, Massachusetts: Butterworth-Heinemann. TIC: 244694.

8.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

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-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.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.

141257 ASME (American Society of Mechanical Engineers) 1995. “Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components.” Section III of *1995 ASME Boiler and Pressure Vessel Code*. New York, New York: American Society of Mechanical Engineers. TIC: 245287.

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.

8.3 SOURCE DATA, LISTED BY DATA TRACKING

163112 LL021012712251.021. Chemical Analysis of Waters from Select Vessels in the Long-Term Corrosion Test Facility at Lawrence Livermore National Laboratory. Submittal date: 01/20/2003.

161253 LL021105312251.023. Stress Corrosion Crack Growth and Initiation Measurements for C-22 and Ti-7, General Electric Global Research Center (GEGRC) 121202. Submittal date: 01/08/2003.

152926 MO0003RIB00073.000. Physical and Chemical Characteristics of Ti Grades 7 and 16. Submittal date: 03/13/2000.

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

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