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BECHTEL SAIC COMPANY LLC

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

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DISCLAIMER

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

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11. Remarks

For TSPA-LA

Kevin G. Mon is responsible for the technical content of all sections of this document.

Lisa A. Rottinghous is Responsible for procedural compliance for all sections of this document.

Change History

12. Revision/ICN No.	13. Total Pages	14. Description of Change
00	53	Initial Issue.
00/01	59	Interim Change Notice. Update for no backfill base case.
01	94	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	106	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

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TABLES

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ACRONYMS

BSC	Bechtel SAIC Company, LLC
BWR	boiling water reactor
CRWMS	Civilian Radioactive Waste Management System
CSNF	commercial spent nuclear fuel
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
HAZ	heat affected zone
HIC	hydrogen induced cracking
LA	License Application
LLNL	Lawrence Livermore National Laboratory
M	Molality, moles per kilogram of solvent
M	Molarity, mol/liter of solution
M&O	Management and Operating
MIC	microbially influenced corrosion
NRC	U.S. Nuclear Regulatory Commission
PRD	Project Requirements Document
PWR	pressurized water reactor
QA	quality assurance
RMEI	reasonably maximally exposed individual
SAW	simulated acidified water
SCC	stress corrosion cracking
SCW	simulated concentrated water
SDW	simulated dilute water
SNF	spent nuclear fuel
SR	Site Recommendation

ACRONYMS (Continued)

TSPA	Total System Performance Assessment
TWP	Technical Work Plan
YMP	Yucca Mountain Project

1. PURPOSE

As directed by a written development plan (BSC 2002 [DIRS 161132]), the primary purpose of this scientific analysis is to identify and document the analyses and resolution of the features, events, and processes (FEPs) affecting the waste package and drip shield performance in the repository. Thirty-three FEPs were identified that are associated with the waste package and drip shield performance. This scientific analysis has been prepared to document the screening methodology used in the process of FEP inclusion and exclusion.

The scope of this scientific analysis is to identify the treatment of the FEPs affecting postclosure waste package and drip shield performance. It should be noted that seismic effects are not treated within this report. A full discussion of seismic effects is contained in the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]). The FEPs that are deemed potentially important to repository postclosure performance are evaluated, either as components of the total system performance assessment (TSPA) or as a separate discussion in a scientific analysis report. The scope for this activity involves two tasks, namely:

Task 1: Identify which FEPs are to be considered explicitly in the TSPA (called included FEPs) and in which scientific analyses these FEPs are addressed.

Task 2: Identify FEPs not to be included in the TSPA (called excluded FEPs) and provide justification for why these FEPs do not need to be a part of the TSPA model.

The analyses documented in this scientific analysis are for the license application (LA) base case design (BSC 2004 [DIRS 167040]). In this design, a drip shield is placed over the waste package and no backfill is placed over the drip shield (BSC 2004 [DIRS 167040]). Each FEP may include one or more specific issues that are collectively described by a FEP name, a FEP description, and descriptor phrases. The FEP Description may encompass a single feature, process or event, or a few closely related or coupled processes provided that the entire FEP can be addressed by a single specific screening argument or TSPA disposition. Descriptor phrases provide additional detail about the subject and content of the FEP beyond the FEP name and description. The FEPs have been assigned to associated scientific analyses. The assignments were based on the nature of the FEPs so that the analysis and resolution for screening decisions reside with the subject-matter experts in the relevant disciplines. This scientific analysis addresses the screening decisions associated with the FEPs for the Performance Assessment Project Waste Package Degradation Modeling group.

1.1 OVERVIEW OF FEATURES, EVENTS PROCESSES ANALYSIS AND DEVELOPMENT

The overall FEPs identification and selection processes are documented in *The Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain* (BSC 2002 [DIRS 158966]) and supplemented by the Key Technical Issue Letter Report *Response to Additional Information Needs on TSPAI 2.05 and TSPAI 2.06* (Freeze 2003 [DIRS 165394]). The initial set of FEPs was created for the Yucca Mountain Project (YMP) TSPA by combining lists of FEPs identified as relevant to the YMP. This list consists of FEPs from the Nuclear Energy Agency working group, FEPs from YMP literature and site studies, and FEPs identified during YMP project staff workshops. The FEPs were identified by a variety of methods, including expert judgment, informal elicitation, event tree analysis, stakeholder review, and regulatory stipulation. All potentially relevant FEPs were evaluated, regardless of origin. This approach led to considerable redundancy in the FEP list because the same FEPs are frequently identified by multiple sources, but it also ensures that a comprehensive review of all of the FEPs will be performed.

To eliminate the redundancy in the FEP list and to create a more efficient aggregation of FEPs to carry forward into the TSPA-LA screening process, the FEPs were classified into a two-dimensional FEP matrix having a physical (subsystem elements and features) hierarchy along one axis (the rows) and a process/event hierarchy along the other axis (the columns) (Table 1-1). Each of the specific FEPs was assigned to one or more of the matrix "intersections," generally representative of a process or event acting upon a feature or element. In the case of coupled processes (e.g., thermal-hydrologic-chemical effects on the waste package), several of the intersections along a physical row (e.g., the Waste Package row) may include the same coupled-process FEP. Similarly, in the case of a high-level process or event (e.g., repository dry-out due to waste heat), several intersections along a process column may include the same FEP.

Table 1-1. Example FEP Matrix

		Repository Processes and Events						
		Flow	**Transport**	**Chemical**	**Mechanical**	**Thermal**	**Microbiological**	**Radiological**
Repository Subsystem Physical Elements and Features	Ground Support							
	Backfill							
	Drip Shield							
	Waste Package							
	Cladding							
	Waste Form							
	Invert							

Note: Only a few of the processes/events and subsystem elements/features are included in this example table.

The matrix classification process resulted in 367 unique FEPs (DTN: MO0307SEPFEPS4.000 [DIRS 164527]), each of which encompassed a single process or event, or a few closely related or coupled processes or events that could be addressed by a specific screening discussion. A few of the 33 FEPs are broad ranging and cover multiple technical areas in addition to waste package and drip shield. These FEPs are shared with other FEP reports, and this FEP report may provide only a partial technical basis for the screening of the FEP. The full technical basis for these shared FEPs is addressed collectively by all of the sharing FEP reports.

The approach used for the evaluation of the 33 waste package and drip shield FEPs is a combination of qualitative and quantitative analyses. The analyses are based on the U.S. Nuclear Regulatory Commission (NRC) regulatory requirements provided in 10 CFR Part 63 [DIRS 159535]) to determine whether or not each FEP should be included in the TSPA. For FEPs that are excluded from the TSPA based on the NRC regulatory requirements, the screening argument includes a summary of the basis and results that indicate either low probability of occurrence or low consequence to radiological exposures to the reasonably maximally exposed individual (RMEI) and radionuclide releases to the accessible environment. As appropriate, screening arguments may cite work done outside the YMP, such as in other scientific analyses. For FEPs that are included in the TSPA, the TSPA disposition includes a short summary of how the FEP has been incorporated in the process models and the TSPA models and, where necessary, a reference to the scientific analysis that describes the disposition in greater detail.

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2. QUALITY ASSURANCE

The quality assurance (QA) program applies to this analysis. The technical work plan entitled *Waste Package Materials Data Analyses and Modeling* (BSC 2002 [DIRS 161132]) determined that this activity is subject to the *Quality Assurance Requirements and Description* (DOE 2003 [DIRS 162903]) requirements. All waste package configurations have been determined to be important to waste isolation in accordance with AP-2.22Q, *Classification Analyses and Maintenance of the Q-List,* and therefore are classified as safety category (SC) on the *Q-List* (BSC 2003 [DIRS 165179], Appendix A; BSC 2003 [DIRS 164554], Section 6.4.2). The drip shields have been determined to be important to waste isolation in accordance with AP-2.22Q and therefore are classified as SC on the *Q-List* (BSC 2003 [DIRS 165179], Appendix A; BSC 2003 [DIRS 164554], Section 6.4.2).

This document was prepared in accordance with AP-SIII.9Q, *Scientific Analyses,* and reviewed in accordance with AP-2.14Q, *Document Review*.

The electronic management of data is not applicable to this analysis report since data were not developed as part of this analysis.

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3. USE OF SOFTWARE

This analysis report uses no computational software; therefore, this analysis is not subject to software controls. The analyses and arguments presented herein are based on guidance and regulatory requirements, results of analyses presented and documented in other analysis reports, or on other technical literature. Software and models used in the supporting documents are cited in this analysis report for traceability and transparency purposes, but were not used in its development. This includes the FEPS Database Software Program, V.2.PC, tracking number 10418-.2-00 (BSC 2002 [DIRS 159684]). This database program was utilized outside of this analysis to establish and organize the original list of TSPA-Site Recommendation (TSPA-SR) FEPs to be addressed for the repository (refer to Section 4.1.3).

This analysis report was developed using only commercial-off-the-shelf software, Microsoft® Word 2000, for word processing which is exempt from qualification requirements in accordance with AP-SI.1Q, *Software Management*. No additional applications (Routines or Macros) were developed using this commercial off-the-shelf software.

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4. INPUTS

4.1 DIRECT INPUTS

4.1.1 Data

The technical information used in this scientific analysis as input has been obtained, where possible, from controlled source documents and references using the appropriate document identifiers or records system accession numbers. These inputs were documented according to the procedure outlined in AP-3.15Q, *Managing Technical Product Inputs*. Inputs that are relied upon in screening arguments within Section 6.2 of this AR are summarized in Table 4-1.

Table 4-1. Direct Inputs

Reference [DIRs Identifier]	Description of Input	Input Status
BSC 2003 [DIRS 168489]	Displacement of DS based on 11.5 MT rockfall	Product Output
BSC 2003 [DIRS 168489]	Mechanical loading of waste package interfaces	Product Output
BSC 2004 [DIRS 167040]	Configuration of WP and DS design; Distance from top of invert to centerline of WP (5 DHLW)	Product Output
BSC 2003 [DIRS 165269], Tables 59 and 60	Radiation dose rates at waste package surface	Product Output
BSC 2003 [DIRS 16168489]	Analysis of multiple rockfalls	Product Output
LL021012712251.021 [DIRS 163112]	Results of long term corrosion testing	Data – Qualified
BSC 2003 [DIRS 168489], Table 1	Interior height of drip shield	Product Output
BSC 2004 [DIRS 167207], Table 1	Nominal diameter of 5 DHLW/DOE SNF short waste package	Product Output
BSC 2004 [DIRS 166694]	Gap between WP barriers	Product Output
BSC 2003 [DIRS 163855]	Mass of WP internals	Product Output

MT = metric ton, WP = waste package, DS = drip shield, DHLW = defense high-level radioactive waste, DOE SNF = U.S. Department of Energy spent nuclear fuel.

4.1.2 Parameters

The analyses and arguments presented herein are based on guidance and regulatory requirements, results of analyses presented and documented in other analysis reports, model reports, or other technical literature. There were no parameters used and this subsection is not applicable.

4.1.3 Other Model/Analysis Inputs and Technical Information

Table 4-2 summarizes inputs used in this analysis that originated from other models or analyses and from other sources of technical information.

Table 4-2. Other Model/Analysis Inputs and Technical Information

Reference [DIRS Identifier]	Description of Input	Input Status
ASM International 1987 [DIRS 103753]	Effects of radiation on Alloy 22; oxygen embrittlement of titanium; effects of hydrogen induced cracking on Alloy 22 and titanium	Technical Information
ASM International 1990 [DIRS 144385], p. 626	Deformation characteristics of titanium and nickel-based alloys	Technical Information
Boyer and Gall [DIRS 155318], Section 32	Creep temperatures for nickel-based alloys and titanium	Technical Information
Haynes International 1988 [DIRS 101995]	Melting temperature of Alloy 22	Technical Information
Revie, R.W. ed. 2000 [DIRS 159370], Chapter 47	Titanium alloy properties	Technical Information
Schutz, R.W. and Thomas, D.E. 1987. [DIRS 144302], p. 669-706	Hydrogen absorption conditions in titanium alloys	Technical Information

The TSPA-LA FEPs list for this study originated from the TSPA-SR FEP list (BSC 2002 [DIRS 159684]) that is described in BSC 2001 [DIRS 154365]. The development of a comprehensive list of FEPs potentially relevant to post-closure performance of the Yucca Mountain repository is an ongoing, iterative process based on site-specific information, design, and regulations. The approach described in Section 0 for developing an initial list of FEPs, in support of TSPA-SR (CRWMS M&O 2000 [DIRS 153246]), was documented in BSC 2001 [DIRS 154365]. The initial FEP list contained 328 FEPs, of which 176 were included in TSPA-SR models (CRWMS M&O 2000 [DIRS 153246], Appendix B, Tables B-9 through B-17). To support TSPA-LA, the FEP list was re-evaluated in accordance with *The Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain* (BSC 2002 [DIRS 158966]) and Key Technical Issue Letter Report *Response to Additional Information Needs on TSPAI 2.05 and TSPAI 2.06* (Freeze 2003 [DIRS 165394]). The resulting TSPA-LA list contains 367 FEPs (DTN: MO0307SEPFEPS4.000 [DIRS 164527]). Significant changes to the waste package and drip shield FEPs from TSPA-SR to TSPA-LA are summarized in a general fashion in Table 4-3. In addition, descriptions for five of the 33 waste package and drip shield FEPs addressed in this report differ from the descriptions in the TSPA-LA list (DTN: MO0307SEPFEPS4.000 [DIRS 164527]) due to recommended changes that were made during the review process. The five FEPs are FEP 2.1.03.07.0A, Mechanical Impact on Waste Package, FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield, FEP 2.1.07.01.0A, Rockfall, FEP 2.1.11.07.0A, Thermal Expansion/Stress of In-Drift EBS Components, and FEP 2.1.13.02.0A, Radiation Damage in EBS.

Table 4-3. Significant Changes to the Waste Package FEPs from TSPA-SR to TSPA-LA

TSPA–SR FEP Number and Name	TSPA-LA FEP Number and Name	Significant Changes from TSPA-SR to TSPA-LA
1.1.03.01.00 Error in waste or backfill emplacement	1.1.03.01.0A Error in waste emplacement	Discussion of waste package and drip shield emplacement error was added.
2.1.03.01.00 Corrosion of waste containers	2.1.03.01.0A General corrosion of waste packages	FEP was split meaning that the FEP only addresses general corrosion of waste packages for LA, but addressed general corrosion of waste packages and drip shields for SR. The FEP also discussed other forms of corrosion for SR, but for LA, these other forms of corrosion are completely addressed in other LA FEPs.
2.1.03.01.00 Corrosion of waste containers	2.1.03.01.0B General corrosion of drip shields	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.01.00.
2.1.03.02.00 SSC of waste containers and DSs	2.1.03.02.0A SCC of waste packages	FEP was split meaning that the FEP only addresses SCC of waste packages for LA but addressed SCC of waste packages and drip shields for SR.
2.1.03.02.00 SSC of waste containers and drip shields	2.1.03.02.0B SCC of drip shields	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.02.00.
2.1.03.03.00 Pitting of waste containers and DSs	2.1.03.03.0A Localized corrosion of waste packages	FEP was split meaning that the FEP only addresses localized corrosion of waste packages for LA but addressed localized corrosion of waste packages and drip shields for SR.
2.1.03.03.00 Pitting of waste containers and DSs	2.1.03.03.0B Localized corrosion of drip shields	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.03.00.
2.1.03.04.00 Hydride cracking of waste containers and DSs	2.1.03.04.0A Hydride cracking of waste packages	FEP was split meaning that the FEP only addresses hydride cracking of WPs for LA but addressed hydride cracking of WPs and DSs for SR.
2.1.03.04.00 Hydride cracking of waste containers and DSs	2.1.03.04.0B Hydride cracking of drip shields	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.04.00.
2.1.03.05.00 Microbially-mediated corrosion of waste container andDSs	2.1.03.05.0A MIC of waste packages	FEP was split meaning that the FEP only addresses MIC of WPs for LA but addressed MIC of WPs and DSs for SR.
2.1.03.05.00 Microbially-mediated corrosion of waste container and DS	2.1.03.05.0B MIC of DSs	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.05.00.
2.1.03.06.00 Internal corrosion of waste container	2.1.03.06.0A Internal corrosion of waste packages prior to breach	None
2.1.03.07.00 Mechanical impact on waste container and DS	2.1.03.07.0A Mechanical impact on WP	FEP was split meaning that the FEP only addresses mechanical impact on WPs for LA, but addressed mechanical impacts on WPs and DSs for SR. The FEP does not address static loading or seismic impacts for LA (they are addressed in other LA FEPs) but addressed all types of mechanical impacts for SR.

Table 4. Significant Changes to the WP FEPs from TSPA-SR to TSPA-LA (Continued)

TSPA–SR FEP Number and Name	TSPA-LA FEP Number and Name	Significant Changes from TSPA-SR to TSPA-LA
2.1.03.07.00 Mechanical impact on waste container and DS	2.1.03.07.0B Mechanical impact on DS	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.07.00.
2.1.03.08.00 Juvenile and early failure of waste containers and DSs	2.1.03.08.0A Early failure of DSs	FEP was split meaning that the FEP only addresses early failure of waste packages for LA but addressed early failure of waste packages and drip shields for SR.
2.1.03.08.00 Juvenile and early failure of waste containers and DSs	2.1.03.08.0B Early failure of DSs	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.08.00.
2.1.03.09.00 Copper corrosion	2.1.03.09.0A Copper corrosion in EBS	None
2.1.03.10.00 Container healing	2.1.03.10.0A Healing of WPs	FEP was split meaning that the FEP only addresses healing of WPs for LA but addressed healing of WPs and DSs for SR.
2.1.03.10.00 Container healing	2.1.03.10.0B Healing of DSs	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.03.10.00.
2.1.03.11.00 Container form	2.1.03.11.0A Physical form of WP and DS	None
2.1.06.06.00 Effects and degradation of DS	2.1.06.06.0B Oxygen embrittlement of DSs	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.06.06.00.
2.1.06.07.00 Effects of material interfaces	2.1.06.07.0B Mechanical effects at EBS component interfaces	FEP was split meaning that the FEP only addresses mechanical effects at EBS component interfaces for LA but addressed chemical and mechanical effects at EBS component interfaces for SR. Chemical effects are discussed under LA FEP 2.1.06.07.0A in the EBS FEPs AR.
2.1.07.01.00 Rockfall (Large Block)	2.1.07.01.0A Rockfall	FEP was modified for LA to only address rockfall under nominal conditions. Seismic-induced rockfall is addressed under LA FEP 1.2.03.02.0B in the EBS and DE FEPs ARs.
2.1.07.05.00 Creeping of metallic materials in the EBS	2.1.07.05.0A Creep of metallic materials in the WP	FEP was split meaning that the FEP only addresses creep of metallic materials in the WP for LA but addressed creep of metallic materials in the WP and DSs for SR.
2.1.07.05.00 Creeping of metallic materials in the EBS	2.1.07.05.0B Creep of metallic materials in the DS	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.07.05.00.
2.1.09.03.00 Volume increase of corrosion products	2.1.09.03.0B Volume increase of corrosion products impacts WP	FEP was split meaning that the FEP only addresses volume increase of corrosion products impacts WP for LA but addressed volume increase of corrosion products impacts on WP, cladding, and other EBS components for SR.

Table 5. Significant Changes to the WP FEPs from TSPA-SR to TSPA-LA (Continued)

TSPA–SR FEP Number and Name	TSPA-LA FEP Number and Name	Significant Changes from TSPA-SR to TSPA-LA
2.1.09.09.00 Electrochemical effects in waste and EBS	2.1.09.09.0A Electrochemical effects in EBS	Discussion of galvanic coupling added.
2.1.11.06.00 Thermal sensitization of waste containers and DS increases their fragility	2.1.11.06.0A Thermal sensitization of WPs	FEP was split meaning that the FEP only addresses thermal sensitization of WPs for LA but addressed thermal sensitization of WPs and DSs for SR.
2.1.11.06.00 Thermal sensitization of waste containers and DSs increases their fragility	2.1.11.06.0B Thermal sensitization of DSs	This is a new FEP for LA (given a new number) although the FEP was discussed in SR FEP 2.1.11.06.00.
2.1.11.07.00 Thermally-induced stress changes in waste and EBS 2.1.11.05.00 Differing Thermal Expansion of Repository Components	2.1.11.07.0A Thermal expansion/stress of in-drift EBS components	This FEP was re-scoped for LA. SR FEPs 2.1.11.05.00 and 2.1.11.07.00, which collectively addressed thermal effects in the repository, were was combined for LA and then split into 2 LA FEPs. This FEP addresses in-drift thermal effects.
2.1.12.03.00 Gas generation (H_2) from metal corrosion	2.1.12.03.0A Gas generation (H_2) from WP corrosion	None
2.1.13.01.00 Radiolysis	2.1.13.01.0A Radiolysis	None
2.1.13.02.00 Radiation damage in waste and EBS	2.1.13.02.0A Radiation damage in EBS	None

DS = drip shield, WP = waste page(s), LA = license application, SR = site recommendation, FEP = feature, event, and process, SSC = stress corrosion cracking, MIC = microbially influenced corrosion, EBS = engineered barrier system, DE = disruptive events

4.2 CRITERIA

4.2.1 NRC Regulatory Requirements

As discussed in *The Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain* (BSC 2002 [DIRS 158966]), FEPs will be screened and arguments documented in accordance with the NRC technical screening regulatory requirements provided in 10 CFR 63 [DIRS 158535].

These regulatory requirements are summarized as follows:

10 CFR 63.21(c)(1) – The Safety Analysis Report must include a description of the Yucca Mountain site, with appropriate attention to those features, events, and processes of the site that might affect design of the geologic repository operations area and performance of the geologic repository. The description of the site must include information regarding features, events, and processes outside of the site to the extent the information is relevant and material to safety or performance of the geologic repository. The information referred to in this paragraph must include:

(i) The location of the geologic repository operations area with respect to the boundary of the site

(ii) Information regarding the geology, hydrology, and geochemistry of the site, including geomechanical properties and conditions of the host rock

(iii) Information regarding surface water hydrology, climatology, and meteorology of the site

(iv) Information regarding the location of the reasonably maximally exposed individual, and regarding local human behaviors and characteristics, as needed to support selection of conceptual models and parameters used for the reference biosphere and reasonably maximally exposed individual.

10 CFR 63.21(c)(9) – The Safety Analysis Report must include an assessment to determine the degree to which those features, events, and processes of the site that are expected to materially affect compliance with 10 CFR 63.113 – whether beneficial or potentially adverse to the performance of the geologic repository – have been characterized, and the extent to which they affect waste isolation.

10 CFR 63.102(j) - . . . Those features, events, and processes expected to materially affect compliance with 10 CFR 63.113(b) or be potentially adverse to performance are included . . .

10 CFR 63.114(d) – Consider only events that have at least one chance in 10,000 of occurring over 10,000 years.

10 CFR 63.114(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.

10 CFR 63.114(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.

4.2.2 Yucca Mountain Review Plan

The Waste Package TWP (BSC 2002 [161132], Attachment C, Table C5) identified acceptance criteria based on the requirements mentioned in the *Project Requirements Document* (PRD)

(Canori and Leitner 2003 [DIRS 166275]) and the *Yucca Mountain Review Plan* (NRC 2003 [DIRS 163274]). The following acceptance criteria from the PRD and *Yucca Mountain Review Plan* are updates to those mentioned in the Waste Package TWP (BSC 2002 [DIRS 161132], Attachment C, Table C5):

System Description and Demonstration of Multiple Barriers (NRC 2003 [DIRS 163274], Section 2.2.1.1; Canori and Leitner 2003 [DIRS 166275], PRD-002/T-014, PRD-002/T-016) – Specific requirements involve identification of multiple barriers (natural and engineered), describing the capabilities of these barriers to isolate waste, and providing technical bases for capabilities descriptions consistent with the postclosure performance objectives as described below:

1. Identification of barriers relied on for postclosure performance; (including at least one barrier from the engineered system and one from the natural system)

2. Description of the capability of identified barriers to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment, or prevent the release or substantially reduce the release of radionuclides from the waste including the uncertainty associated with this capacity and the consistency with approaches used in the total system performance assessment

3. Discussion of the technical bases for assertions of barrier capability commensurate with the importance of a particular barrier in the performance assessment and with the associated uncertainties.

To comply with these requirements, the following acceptance criteria are identified in the Waste Package TWP for this report (BSC 2002 [DIRS 161132], Attachment C, Table C5):

AC1: Identification of Barriers is Adequate: Barriers relied on to achieve compliance with 10 CFR 63.113(b), as demonstrated in the TSPA, are adequately identified, and are clearly linked to their capability. The barriers identified include at least one from the engineered system and one from the natural system.

AC2: Description of the Capability to Isolate Waste is Acceptable: The capability of the identified barriers to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment, or prevent the release or substantially reduce the release rate of radionuclides from the waste is adequately identified and described:

1. The information on the time period over which each barrier performs its intended function, including any changes during the compliance period, is provided

2. The uncertainty associated with barrier capabilities is adequately described

3. The described capabilities are consistent with the results from the total system performance assessment

4. The described capabilities are consistent with the definition of a barrier at 10 CFR 63.2.

Scenario Analysis and Event Probability (NRC 2003 [DIRS 163274], Section 2.2.1.2; Canori and Leitner 2003 [DIRS 166275], PRD-002/T-015) – Specific requirements include providing technical bases for inclusion or exclusion of specific FEPs. In order to meet these requirements, the following acceptance criteria are identified in the Waste Package TWP for this report (BSC 2002 [DIRS 161132], Attachment C, Table C5):

AC1: The Identification and Initial List of Features, Events, and Processes is Adequate: The Safety Analysis Report contains a complete list of FEPs 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 FEPs 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.

AC2: Screening of the Initial List of Features, Events, and Processes is Appropriate:

1. The U.S. Department of Energy (DOE) has identified all FEPs 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;

2. The DOE has provided justification for those FEPs that have been excluded. An acceptable justification for excluding FEPs is that either the FEP is specifically excluded by regulation; probability of the FEP (generally an event) falls below the regulatory criterion; or omission of the FEP 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

3. The DOE has provided an adequate technical basis for each FEP, excluded from the performance assessment, to support the conclusion that either the FEP is specifically excluded by regulation; the probability of the FEP falls below the regulatory criterion; or omission of the FEP 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.

AC3: Formation of Scenario Classes Using the Reduced Set of Events is Adequate:

1. Scenario classes are mutually exclusive and complete, clearly documented, and technically acceptable.

AC4: Screening of Scenario Classes is Appropriate:

1. Screening of scenario classes is comprehensive, clearly documented, and technically acceptable;

2. The DOE has adequately considered coupling of processes in estimates of consequences used to screen scenario classes. Scenario classes were not prematurely excluded by a narrow definition;

3. Scenario classes that are screened from the performance assessment, on the basis that they are specifically ruled out by regulation or are contrary to stated regulatory assumptions are identified, and sufficient justifications are provided;

4. Scenario classes that are screened from the performance assessment, on the basis that their probabilities fall below the regulatory criterion, are identified, and sufficient justifications are provided

5. Scenario classes that are screened from the performance assessment, on the basis that their omission would not significantly change the magnitude and time of the resulting radiological exposure to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, are identified, and sufficient justifications are provided.

Degradation of Engineered Barriers (NRC 2003 [DIRS 163274], Section 2.2.1.3.1; Canori and Leitner 2003 [DIRS 166275]; PRD-002/T-015) – Specific requirements include describing deterioration or degradation of engineered barriers and modeling degradation processes using data for performance assessment, including TSPA. Consideration of uncertainties and variabilities in model parameters and alternative conceptual models are also required. Specific wording from the *Yucca Mountain Review Plan* (NRC 2003 [DIRS 163274], Section 2.2.1.3.1) is as follows:

To review this model abstraction, evaluate the adequacy of the DOE LA, relative to the degree to which degradation of engineered barriers affects the DOE LA. Review this model abstraction, considering the risk information evaluated in the "Multiple Barriers" Section 2.2.1.1. For example, if the DOE relies on the engineered barriers to provide significant delay in the transport of radionuclides to the reasonably maximally exposed individual, then perform a detailed review of this abstraction. If, on the other hand, the DOE demonstrates this abstraction to have a minor impact on the dose to the RMEI, then conduct a simplified review focusing on the bounding assumptions. The review methods and acceptance criteria provided here are for a detailed review. Some of the review methods and acceptance criteria may not be necessary in a simplified review for those abstractions that have a minor impact on performance. The demonstration of compliance with the performances objective is evaluated using Section 2.2.1.4 of the *Yucca Mountain Review Plan*.

To fulfill these requirements, the following acceptance criteria are identified in the TWP for this report (BSC 2002 [DIRS 161132], Attachment C, Table C5):

AC1: System Description and Model Integration are Adequate:

1. The TSPA adequately incorporates important design features, physical phenomena, and couplings, and uses consistent and appropriate assumptions throughout the degradation of engineered barriers abstraction process.

2. Assessment abstraction of the degradation of engineered barriers uses assumptions, technical bases, data, and models that are appropriate and consistent with other related DOE abstractions. For example, the assumptions used for degradation of engineered barriers should be consistent with the abstractions of the quantity and chemistry of water contacting waste packages and waste forms (Section 2.2.1.3.3); climate and infiltration (Section 2.2.1.3.5); and mechanical disruption of waste packages (Section 2.2.1.3.2). The descriptions and technical bases provide transparent and traceable support for the abstraction of the degradation of engineered barriers.

3. The descriptions of engineered barriers, design features, degradation processes, physical phenomena, and couplings that may affect the degradation of the engineered barriers are adequate. For example, materials and methods used to construct the engineered barriers are included, and degradation processes, such as uniform corrosion, pitting corrosion, crevice corrosion, stress corrosion cracking, intergranular corrosion, microbially influenced corrosion, dry-air oxidation, hydrogen embrittlement, and the effects of wet and dry cycles, material aging and phase instability, welding, and initial defects on the degradation modes for the engineered barriers are considered.

4. Boundary and initial conditions used in the TSPA abstractions are propagated consistently throughout the abstraction approaches. For example, the conditions and assumptions used in the degradation of engineered barriers abstraction are consistent with those used to model the quantity and chemistry of water contacting waste packages and waste forms (Section 2.2.1.3.3); climate and infiltration (Section 2.2.1.3.5); and mechanical disruption of waste packages (Section 2.2.1.3.2).

5. Sufficient technical bases for the inclusion of FEPs related to degradation of engineered barriers in the TSPA abstractions are provided.

6. Adequate technical bases are provided, for selecting the design criteria, that mitigate any potential impact of in-package criticality on repository performance, including considering all FEPs that may increase the reactivity of the system inside the waste package. For example, the technical bases for the abstraction of the degradation of engineered barriers include configuration classes and configurations that have potential for nuclear criticality, changes in radionuclide inventory, and changes in thermal conditions.

7. Guidance in NUREG–1297 (Altman et al. 1988 [103597]) and NUREG–1298 (Altman et al. 1988 [103750]), or other acceptable approaches, is followed.

AC2: Data are Sufficient for Model Justification:

1. Parameters used to evaluate the degradation of engineered barriers in the LA are adequately justified (e.g., laboratory corrosion tests, site-specific data such as data from drift-scale tests, in-service experience in pertinent industrial applications, and test results not specifically performed for the Yucca Mountain site, etc.). The DOE describes how the data were used, interpreted, and appropriately synthesized into the parameters

2. Sufficient data have been collected on the characteristics of the engineered components, design features, and the natural system to establish initial and boundary conditions for abstraction of degradation of engineered barriers

3. Data on the degradation of the engineered barriers (e.g., general and localized corrosion, microbially influenced corrosion, galvanic interactions, hydrogen embrittlement, and phase stability), used in the abstraction, are based on laboratory measurements, site-specific field measurements, industrial analog and/or natural analog research, and tests designed to replicate the range of conditions that may occur at the Yucca Mountain site. As appropriate, sensitivity or uncertainty analyses, used to support the DOE TSPA abstraction, are adequate to determine the possible need for additional data4. Degradation models for the processes that may be significant to the performance of the engineered barriers are adequate. For example, the DOE models consider the possible degradation of the engineered barriers, as a result of uniform and localized corrosion processes, stress-corrosion cracking, microbially influenced corrosion, hydrogen embrittlement, and incorporate the effects of fabrication processes, thermal aging, and phase stability.

Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms (NRC 2003 [DIRS 163274], Section 2.2.1.3.3; Canori and Leitner 2003 [DIRS 166275], PRD-002/T-015) – Specific requirements include quantifying the amount and chemistry of water contacting the waste package and the waste forms as described below:

1. Description of the geological, hydrological, and geochemical aspects of quantity and chemistry of water contacting engineered barriers and waste forms, and the technical bases the DOE provides to support model integration across the TSPA abstractions

2. Sufficiency of the data and parameters used to justify the model abstraction

3. Methods the DOE uses to characterize data uncertainty, and propagate the effects of this uncertainty through the TSPA model abstraction

4. Methods the DOE uses to characterize model uncertainty, and propagate the effects of this uncertainty through the TSPA model abstraction

5. Approaches the DOE uses to compare TSPA output to process-level model outputs and empirical studies

6. Use of expert elicitation.

To comply with these requirements, the following acceptance criteria are identified in the TWP for this report (BSC 2002 [161132], Attachment C, Table C5):

AC1: System Description and Model Integration are Adequate (This acceptance criterion was not specifically mentioned in the TWP (BSC 2002 [DIRS 161132]) for this analysis report, but is relevant to this analysis):

1. TSPA adequately incorporates important design features, physical phenomena, and couplings, and uses consistent and appropriate assumptions throughout the quantity and chemistry of water contacting engineered barriers and waste forms abstraction process;.

2. The abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms uses assumptions, technical bases, data, and models, that are appropriate and consistent with other related DOE abstractions. For example, the assumptions used for the quantity and chemistry of water contacting engineered barriers and waste forms are consistent with the abstractions of "Degradation of Engineered Barriers" (Section 2.2.1.3.1); "Mechanical Disruption of Engineered Barriers (Section 2.2.1.3.2); "Radionuclide Release Rates and Solubility Limits" (Section 2.2.1.3.4); "Climate and Infiltration" (Section 2.2.1.3.5); and "Flow Paths in the Unsaturated Zone" (Section 2.2.1.3.6). The descriptions and technical bases provide transparent and traceable support for the abstraction of quantity and chemistry of water contacting engineered barriers and waste forms.

3. Important design features, such as waste package design and material selection, backfill, drip shield, ground support, thermal loading strategy, and degradation processes, are adequate to determine the initial and boundary conditions for calculations of the quantity and chemistry of water contacting engineered barriers and waste forms.

4. Spatial and temporal abstractions appropriately address physical couplings (thermal-hydrologic-mechanical-chemical). For example, the DOE evaluates the potential for focusing of water flow into drifts, caused by coupled thermal-hydrologic-mechanical-chemical processes.

5. Sufficient technical bases and justification are provided for TSPA assumptions and approximations for modeling coupled thermal-hydrologic-mechanical-chemical effects on seepage and flow, the waste package chemical environment, and the chemical environment for radionuclide release. The effects of distribution of flow on the amount of water contacting the engineered barriers and waste forms are consistently addressed, in all relevant abstractions.

6. The expected ranges of environmental conditions within the waste package emplacement drifts, inside of breached waste packages, and contacting the waste forms and their evolution with time are identified. These ranges may be developed to include: (i) the effects of the drip shield and backfill on the quantity and chemistry of water (e.g., the potential for condensate formation and dripping from the underside of the shield); (ii) conditions that promote corrosion of engineered barriers and degradation of waste forms; (iii) irregular wet and dry cycles; (iv) gamma-radiolysis; and (v) size and distribution of penetrations of engineered barriers.

7. The model abstraction for quantity and chemistry of water contacting engineered barriers and waste forms is consistent with the detailed information on engineered barrier design and other engineered features. For example, consistency is demonstrated for: (i) dimensionality of the abstractions; (ii) various design features and site characteristics; and (iii) alternative conceptual approaches. Analyses are adequate to demonstrate that no deleterious effects are caused by design or site features that the DOE does not take into account in this abstraction.

8. Adequate technical bases are provided, including activities such as independent modeling, laboratory or field data, or sensitivity studies, for inclusion of any thermal-hydrologic-mechanical-chemical couplings and FEPs .

9. Performance-affecting processes that have been observed in thermal-hydrologic tests and experiments are included into the performance assessment. For example, the DOE either demonstrates that liquid water will not reflux into the underground facility or incorporates refluxing water into the performance assessment calculation, and bounds the potential adverse effects of alteration of the hydraulic pathway that result from refluxing water.

10. Likely modes for container corrosion (Section 2.2.1.3.1 of the Yucca Mountain Review Plan) are identified and considered in determining the quantity and chemistry of water entering the engineered barriers and contacting waste forms. For example, the model abstractions consistently address the role of parameters, such as pH, carbonate concentration, and the effect of corrosion on the quantity and chemistry of water contacting engineered barriers and waste forms.

11. The abstraction of in-package criticality or external-to-package criticality, within the emplacement drift, provides an adequate technical basis for screening these events. If either event is included in the assessment, then the DOE uses acceptable technical bases for selecting the design criteria that mitigate the potential impact of in-package criticality on repository performance; identifies the FEPs that may increase the reactivity of the system inside the waste package; identifies the configuration classes and configurations that have potential for nuclear criticality; and includes changes in thermal conditions and degradation of engineered barriers in the abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms.

12. Guidance in NUREG–1297 (Altman et al. 1988 [DIRS 103597]) and NUREG–1298 (Altman et al. 1988 [DIRS 103750]), or other acceptable approaches, is followed.

AC 2: Data are Sufficient for Model Justification:

1. Geological, hydrological, and geochemical values used in the LA are adequately justified. Adequate description of how the data were used, interpreted, and appropriately synthesized into the parameters is provided

2. Sufficient data were collected on the characteristics of the natural system and engineered materials to establish initial and boundary conditions for conceptual models of thermal-hydrologic-mechanical-chemical coupled processes, that affect seepage and flow and the engineered barrier chemical environment

3. Thermo-hydrologic tests were designed and conducted with the explicit objectives of observing thermal-hydrologic processes for the temperature ranges expected for repository conditions and making measurements for mathematical models. Data are sufficient to verify that thermal-hydrologic conceptual models address important thermal-hydrologic phenomena

4. Sufficient information to formulate the conceptual approach(es) for analyzing water contact with the drip shield, engineered barriers, and waste forms is provided

5. Sufficient data are provided to complete a nutrient- and energy-inventory calculation, if it has been used to justify the inclusion of the potential for microbial activity affecting the engineered barrier chemical environment and the chemical environment for radionuclide release. As necessary, data are adequate to support determination of the probability for microbially influenced corrosion and microbial effects, such as production of organic byproducts and microbially enhanced dissolution of the high-level radioactive waste glass form.

4.2.3 Screening Decisions

The NRC requires the consideration and evaluation of FEPs as part of the performance assessment activities. More specifically, the NRC regulations allow the exclusion of FEPs from the TSPA if they can be shown to be of low probability of occurrence or of low consequence. The specified criteria can be summarized in the form of two FEP screening statements as follows:

1. The event has at least one chance in 10,000 of occurring over 10,000 years (see 10 CFR 63.114(d) (66 FR 55732 [DIRS 156671])).

2. The magnitude and time of the resulting radiological exposure to the RMEI, or radionuclide release to the accessible environment, would be significantly changed by its omission (see 10 CFR 63.114 (e and f) (66 FR 55732 [DIRS 156671]).

Additionally, the Acceptance Criteria in the Yucca Mountain Review Plan (NRC 2003, Section 2.2.1.2.1.3 [DIRS 163274]) calls for evaluating the FEPs based on the regulations. This criterion can be summarized in the form of a third FEP screening statement.

3. The FEP is not excluded by regulation.

If there are affirmative conditions for all three screening criteria, the FEP is included in the TSPA-LA model. Any negating condition in the three screening criteria excludes the FEP from the TSPA-LA model. The approach used for this analysis is a combination of qualitative and quantitative screening of FEPs.

Criteria presented here are addressed in each of the screening arguments and disposition statements presented in Section 6.2 FEPS ANALYSES of this analysis report.

4.3 CODES AND STANDARDS

This document was prepared to comply with NRC regulatory requirements presented in 10 CFR 63 [DIRS 158535]. 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).

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5. ASSUMPTIONS

There are three assumptions made in screening of the waste package FEPs. These assumptions or combinations thereof are used throughout this report.

Assumption 1) Assume the evolution of the geologic setting is consistent with present knowledge of natural processes.

This assumption is justified based on the regulations stated in 10 CFR 63.305(c) [DIRS 158535] where 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 assumption affects waste package and drip shield FEPs concerned with geologic processes. The assumption implies that existing knowledge of natural processes is sufficient to adequately quantify future states of the system.

Assumption 2) Assume that the repository will be constructed, operated, and closed according to the regulatory requirements applicable to the construction, operation, and closure period, and that deviations from design will be detected and corrected.

This assumption is justified based on the conditions specified in 10 CFR 63.32 [DIRS 158535], which pertains to construction authorization and which requires

> "Periodic or special reports regarding:
>
> (1) Progress of construction;
>
> (2) Any data about the site, obtained during construction, that are not within the predicted limits on which the facility design was based;
>
> (3) Any deficiencies, in design and construction that, if uncorrected, could adversely affect safety at any future time;
>
> (4) Results of research and development programs being conducted to resolve safety questions."

In addition, 10 CFR 63 Subpart F [DIRS 158535] requires that a performance confirmation program be instituted. The focus of the program is confirmation of geotechnical and design parameters (Section 63.132), design testing (Section 63.133) and monitoring and testing waste packages (Section 63.134). In addition, under 10 CFR 63 Subpart G [DIRS 158535], quality assurance requirements are applied to "site characterization, acquisition, control, and analyses of samples and data; tests and experiments; scientific studies; facility and equipment design and construction; facility operation; performance confirmation; permanent closure; and decontamination and dismantling of surface facilities." The assumption impacts waste package and drip shield FEPs that are affected by events occurring during the construction, operation, or closure period.

Assumption 3) Assume that the design parameters for the waste package and drip shield can be used to justify an excluded decision.

This assumption is justified based on the conditions specified in 10 CFR 63.142 Subpart G [DIRS 158535] that pertain to quality assurance.

> "DOE shall establish measures to assure that applicable regulatory requirements and the design basis, as defined in Section 63.2 and as specified in the license application, for those structures, systems, and components to which this subpart applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must assure that appropriate quality standards are specified and included in the design documents and that deviations from such standards are controlled. Measures must also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are important to waste isolation and important to safety functions of the structures, systems and components."

The assumption allows exclusion of FEPs when the design process specifically addresses the issue described by that particular FEP (refer to Section 4.2.3 for possible screening decisions). Note that deviation from a design process despite a set of quality controls is allowed for in the TSPA. None of the assumptions presented herein require confirmation.

6. SCIENTIFIC ANALYSIS DISCUSSION

The primary concern in this scientific analysis is to address and document the screening decisions for the 33 waste package and drip shield degradation FEPs listed in Table 6-1. In some cases, where a FEP covers multiple technical areas and is shared with other FEP analysis reports, this analysis report may provide only a partial technical basis for the screening of the FEP. The shared FEPs are identified in column 3 of Table 6-1. The full technical basis for these shared FEPs is addressed collectively by all of the sharing FEP analysis reports.

Table 6-1. TSPA-LA FEP List

LA FEP Number	FEP Name	Shared With
1.1.03.01.0A	Error in Waste Emplacement	EBS
2.1.03.01.0A	General Corrosion of WPs	
2.1.03.01.0B	General Corrosion of DSs	
2.1.03.02.0A	SCC of WPs	
2.1.03.02.0B	SCC of DSs	
2.1.03.03.0A	Localized Corrosion of WPs	
2.1.03.03.0B	Localized Corrosion of DSs	
2.1.03.04.0A	Hydride Cracking of WPs	
2.1.03.04.0B	Hydride Cracking of DSs	
2.1.03.05.0A	MIC of WPs	
2.1.03.05.0B	MIC of DSs	
2.1.03.06.0A	Internal Corrosion of WPs Prior To Breach	WF
2.1.03.07.0A	Mechanical Impact on WP	
2.1.03.07.0B	Mechanical Impact on DS	
2.1.03.08.0A	Early Failure of WPs	
2.1.03.08.0B	Early Failure of DSs	
2.1.03.09.0A	Copper Corrosion in EBS	
2.1.03.10.0A	Healing of WPs	
2.1.03.10.0B	Healing of DSs	
2.1.03.11.0A	Physical Form of WP and DS	
2.1.06.06.0B	Oxygen Embrittlement of DSs	
2.1.06.07.0B	Mechanical Effects at EBS Component Interfaces	EBS
2.1.07.01.0A	Rockfall	EBS, Cladding
2.1.07.05.0A	Creep of Metallic Materials in the WP	
2.1.07.05.0B	Creep of Metallic Materials in the DS	
2.1.09.03.0B	Volume Increase of Corrosion Products Impacts WP	
2.1.09.09.0A	Electrochemical Effects in EBS	Cladding
2.1.11.06.0A	Thermal Sensitization of WPs	
2.1.11.06.0B	Thermal Sensitization of DSs	
2.1.11.07.0A	Thermal Expansion/Stress of In-Drift EBS Components	EBS
2.1.12.03.0A	Gas Generation (H_2) From WP Corrosion	EBS Cladding
2.1.13.01.0A	Radiolysis	EBS, WF
2.1.13.02.0A	Radiation Damage in EBS	EBS, WF

EBS = engineered barrier system, WPs = waste packages, DSs = drip shields, SSC = stress corrosion cracking, MIC = microbially influenced corrosion, WF – waste form

DTN: MO0307SEPFEPS4.000 [164527]

6.1 APPROACH

The approach used for this analysis is a combination of qualitative and quantitative screening of FEPs. The analyses are based on the regulatory requirements provided by the NRC in 10 CFR Part 63 [DIRS 158535] to determine whether or not each of the 33 waste package and drip shield FEPs should be included in the TSPA or excluded from further analysis. Each FEP is screened against the low-probability of occurrence and low-consequence criteria in Section 4.2 of this analysis report.

For FEPs that are excluded from the TSPA based on the NRC regulatory requirements, the screening argument includes a summary of the basis and results that indicate either low probability of occurrence or low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment. As appropriate, screening arguments cite work done outside the YMP, such as in other scientific analyses. The Screening Argument section also addresses each of the descriptor phrases, which provide additional detail about the subject and content of the FEP beyond the FEP name and description. The screening arguments are documented in sufficient enough detail to meet the criteria listed in Section 4.2 of this analysis report.

For FEPs that are included in the TSPA, the TSPA Disposition section of each FEP discussion includes a short summary of how the FEP has been incorporated in the process models or the TSPA models and a reference to the scientific analysis that describes the disposition in greater detail. The TSPA Disposition section also addresses each of the descriptor phrases, which provide additional detail about the subject and content of the FEP beyond the FEP name and description. The disposition statements are documented in sufficient enough detail to meet the criteria listed in Section 4.2 of this AR.

6.2 FEATURES, EVENTS, AND PROCESSES ANALYSES

This scientific analysis addresses the 33 FEPs that pertain to waste package and drip shield degradation.

6.2.1 Error in Waste Emplacement

6.2.1.1 Features, Events, and Process Number

FEP 1.1.03.01.0A

6.2.1.2 Features, Events, and Process 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 involves erroneously emplacing the waste packages in a saturated or wet zone of the repository. This would clearly impact the repository performance both by impacting waste package corrosion and radionuclide transport.

6.2.1.3 Descriptor Phrases

Waste emplacement error (mechanical impact)

Deviations from design

Quality control (inadequate)

Waste emplacement error (placement in wet zone)

Drip shield emplacement error

6.2.1.4 Screening Decision

Excluded – Low Consequence

6.2.1.5 Screening Argument

Analyses presented in *Analysis of Mechanisms for Early Waste Package Failure* (BSC 2003 [DIRS 164475], Section 6) indicate that the consequences of errors in waste package emplacement are very low because of the administrative and procedural control measures that will be instituted.

Accidental misloading of waste packages could result in thermal outputs not within the expected range. These misloadings could cause the waste package surface temperatures and relative humidities *to* be outside the expected ranges and impact the degradation characteristics of the waste package and drip shield. Analyses presented in the *Analysis of Mechanisms for Early Waste Package Failure* (BSC 2003 [DIRS 164475], Section 6.2.8), indicate that thermal misloads would not be expected to have any significant consequences for postclosure performance of the waste package/drip shield barriers. This is because the waste package has a large heat transfer area and therefore, the increased heat output from an overloaded waste package would be quickly dissipated into the emplacement drift and not alter the waste package/drip shield surface temperatures to an extent significant enough to affect postclosure performance. On this basis, the effects of thermal misloadings are excluded based on negligible consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

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. These effects are excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment as discussed above. 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 *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]).

The current engineered barrier design includes a titanium drip shield that is placed over the waste packages (BSC 2004 [DIRS 167040]). The drip shield will be continuous down the entire length of the drift, and will be fabricated and emplaced in segments. Each segment will slightly overlap the previously emplaced segment. The benefits of the drip shield could be diminished for a particular waste package if a drip shield segment fails to overlap with the previously emplaced segment, such that a large separation exists that would allow any dripping water above it to directly fall onto the waste package below. The *Analysis of Mechanisms for Early Waste Package Failure* (BSC 2003 [DIRS 164475], Section 6.4.7) indicates that drip shield emplacement errors large enough to allow dripping water to contact the underlying waste package would be detected. Thus, errors in drip shield emplacement can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment since drip shield emplacement errors large enough to allow dripping water to contact the underlying waste package would be detected.

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 due to the fact that the drip shields must fail in order 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 failure mode of any consequence to repository performance (note that seismic effects are not considered in this section). As shown in the *General Corrosion and Localized Corrosion of Drip Shield* report (BSC 2003 [DIRS 161236], Tables 13 and 14) the maximum general corrosion rate which can be applied to the underside of drip shield is about 1.13×10^{-4} mm/yr and the maximum general corrosion rate which 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 plates can fail by general corrosion is about 35 thousand years. Therefore, emplacement of waste packages in a wet zone can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

6.2.1.6 Total System Performance Assessment Disposition

N/A

6.2.1.7 Supporting Analysis Reports

Analysis of Mechanisms for Early Waste Package Failure (BSC 2003 [DIRS 164475])

General Corrosion and Localized Corrosion of Drip Shield (BSC 2003 [DIRS 161236])

6.2.2 General Corrosion of Waste Packages

6.2.2.1 Features, Events, and Process Number

FEP 2.1.03.01.0A

6.2.2.2 Features, Events, and Process Description

General corrosion may contribute to waste package failure.

6.2.2.3 Descriptor Phrases

General (uniform) corrosion of waste packages

Aqueous corrosion

Dry-air oxidation

Humid-air corrosion

Effects of trace metals on general corrosion of waste packages

Effects of corrosive gases on corrosion of waste packages

6.2.2.4 Screening Decision

Included

6.2.2.5 Screening Argument

N/A

6.2.2.6 TSPA Disposition

General corrosion is included in waste package degradation analysis. This includes the effects of corrosive gases. Because general corrosion is likely to be operative for most of the repository operation period, it is one of the key corrosion processes that could lead to degradation and failure of waste packages in the repository. General corrosion due to dry-air oxidation, aqueous corrosion, microbially influenced corrosion and aging and phase instability of the waste package outer barrier are discussed in the *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2003 [DIRS 161235]) model report.

It was concluded in the *General Corrosion and Localized Corrosion of the Waste Package Outer Barrier* model report (BSC 2003 [DIRS 161235], Section 6.4.2) 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 does not need to be considered in TSPA analyses.

Penetration rates for general corrosion are provided in Section 6.4.3 of the *General Corrosion and Localized Corrosion of the Waste Package Outer Barrier* model report (BSC 2003 [DIRS 161235]). General corrosion rates of the waste package outer barrier were estimated using the weight-loss of Alloy 22 crevice geometry specimens after 5-year exposure in the Long Term Corrosion Test Facility (BSC 2003 [DIRS 161235], Section 6.4.3). General corrosion progresses uniformly at a time independent constant rate and the depth of penetration or thinning of the waste package outer barrier by general corrosion is equal to the general corrosion rate multiplied by the time the waste package is exposed to an environment under which general corrosion occurs.

Details of the general corrosion rate distributions used for the Alloy 22 Waste Package Outer Barrier (WPOB) are given in *General Corrosion and Localized Corrosion of the Waste Package Outer Barrier* model report (BSC 2003 [DIRS 161235]). The Alloy 22 general corrosion rate is considered to be a function of exposure temperature. The temperature dependence follows an Arrhenius relationship, i.e.,

$$\ln(R_T) = \left[C_o + \frac{C_1}{T} \right] \qquad \text{(Eq. 1)}$$

where

R_T	=	temperature-dependent general corrosion rate
T	=	temperature (Kelvin)
C_o	=	intercept term
C_1	=	slope term (Kelvin)

as discussed in the report entitled *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2003 [DIRS 161235], Section 6.4.3). The slope term, C_1, is determined from short-term polarization resistance data for Alloy 22 specimens tested for a range of sample configurations, metallurgical conditions, and exposure conditions (BSC 2003 [DIRS 161235], Section 6.4.3). The intercept term, C_o, is determined from the general corrosion rate distribution derived from the weight loss of the 5-year crevice geometry samples exposed in the Long Term Corrosion Test Facility (BSC 2003 [DIRS 161235], Section 6.4.3) and the value of the slope term, C_1. The general corrosion rate distribution derived from the weight loss of the 5-year crevice geometry samples exposed in the Long Term Corrosion Test Facility are considered to represent the distribution of long-term general corrosion rates of the waste package outer barrier at 60°C (333.15 K) (BSC 2003 [DIRS 161235], Section 6.4.3). Therefore,

$$\ln(R_o) = C_o + \frac{C_1}{T_o(=333.15K)} \qquad \text{(Eq. 2)}$$

or

$$C_o = \ln(R_o) - \frac{C_1}{333.15} \qquad \text{(Eq. 3)}$$

where R_o is the general corrosion rate distribution from the 5-year crevice geometry samples. Substituting for C_o in Equation 1,

$$\ln(R_T) = \left[\ln(R_o) + C_1 \left(\frac{1}{T} - \frac{1}{333.15} \right) \right] \qquad \text{(Eq. 4)}$$

The *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* report (BSC 2003 [DIRS 161235], Section 6.4.3) states that R_o is given by a Weibull distribution. The patch size used to model the waste packages is four times the area of the crevice geometry specimen size used to evaluate R_o. Therefore, the general corrosion rates are adjusted to account for the effects of this change of scale. Conceptually, the method employed corresponds to using the highest of four sampled corrosion rates (from the R_o Weibull distribution) to model general corrosion of 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 both commercial spent nuclear fuel and co-disposed 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 Lawrence Livermore National Laboratory (LLNL) Long Term Corrosion Test Facility . Samples of Alloy 22 were immersed in simulated acidified water for five years (DTN: LL021012712251.021 [DIRS 163112]) in the LLNL LTCTF. After five years, a sample of the acidified water, including any corrosion products, was analyzed for trace metals using inductively coupled plasma/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 (BSC 2003 [DIRS 161235]) 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 although the effects are not explicitly and separately modeled.

Additional aqueous corrosion processes are addressed in other FEPs: Localized Corrosion of Waste Packages (FEP 2.1.03.03.0A), Stress Corrosion Cracking (SCC) of Waste Packages (FEP 2.1.03.02.0A), Hydrogen Induced Cracking (HIC) of Waste Packages (FEP 2.1.03.04.0A), and Microbially Influenced Corrosion (MIC) of Waste Packages (FEP 2.1.03.05.0A). These processes are also described in BSC 2003 [DIRS 161235] and BSC 2003 [DIRS 161234].

The effect of microbial activity on the general corrosion process of the waste package outer barrier is represented in TSPA analyses with a general corrosion rate enhancement factor (BSC 2003 [DIRS 161235], Table 6-1). 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 welds and thermal aging of the waste package outer barrier on the general corrosion rates (BSC 2003 [DIRS 161235], Table 6-1). It was also concluded that the aging of both the base metal and welds of the waste package outer barrier under the thermal conditions expected in the repository is not significant for the regulatory time period (BSC 2003 [DIRS 161235], Table 6-1) and will not be specifically modeled in the TSPA.

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 Model (BSC 2003 [DIRS 161317]). The output from the Integrated Waste Package Degradation Model 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 Model is used directly in the TSPA-LA analysis.

The TSPA-LA waste package degradation analysis simulates the behavior of a few hundred waste packages (BSC 2003 [DIRS 161317], 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 that were considered to vary over the repository are relative humidity and temperature at the waste package surface. In addition, potentially variable corrosion processes within a single waste package are represented by dividing the waste package surface into subareas called “patches” and stochastically sampling the degradation model parameter values for each patch. The use of patches explicitly represents the variability in degradation processes within 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 Model. For each realization, values are sampled for the uncertain degradation parameters and passed to the Integrated Waste Package Degradation Model. Each realization is a complete Integrated Waste Package Degradation Model simulation of a given number of waste packages, explicitly considering variability in the degradation processes. Accordingly, each of the Integrated Waste Package Degradation Model 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) which represent the potential range of the output parameters. For example, the waste-package failure time profiles are reported with a group of “curves” representing the cumulative probability of waste package failures as a function of time. The outputs of the Integrated Waste Package Degradation Model are used as input for waste form degradation analysis and radionuclide release analysis from failed waste packages conducted within the TSPA-LA model.

6.2.2.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.3 General Corrosion of Drip Shields

6.2.3.1 Features, Events, and Process Number

FEP 2.1.03.01.0B

6.2.3.2 FEP Description

General corrosion may contribute to drip shield failure.

6.2.3.3 Descriptor Phrases

General (uniform) corrosion of drip shields

Aqueous corrosion

Dry-air oxidation

Humid-air corrosion

Effects of trace metals on general corrosion of drip shields

Effects of corrosive gases on corrosion of drip shields

6.2.3.4 Screening Decision

Included

6.2.3.5 Screening Argument

N/A

6.2.3.6 Total System Performance Analysis 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 the *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2003 [DIRS 161236]) model report.

It was concluded in the *General Corrosion and Localized Corrosion of the Drip Shield* model report (BSC 2003 [DIRS 161236], Section 8.2) that although dry air oxidation occurs, it results in a negligible amount of barrier thinning over repository time scales (only ~2,129 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 analyses.

Penetration rates for general corrosion are provided in the *General Corrosion and Localized Corrosion of the Drip Shield* model report (BSC 2003 [DIRS 161236]) and are used in TSPA analyses. Both humid-air and aqueous corrosion processes are considered part of general corrosion (BSC 2003 [DIRS 161236], Sections 6.1 and 6.3). General corrosion rates of the drip shield were estimated with weight-loss data of Titanium Grade 16 samples after 1 and 5-year exposure in the Long Term Corrosion Test Facility (BSC 2003 [161236], Sections 6.1 and 6.3). 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 LLNL Long Term Corrosion Test Facility. Samples of Titanium Grade 16, an excellent analogue for the more corrosion resistant Titanium Grade 7, were immersed in simulated acidified water for five years (DTN: LL021012712251.021 [DIRS 163112]) in the LLNL Long Term Corrosion Test Facility. After five years, a sample of the acidified water, including any corrosion products, was analyzed for trace metals using inductively coupled plasma/mass spectrometry methods. Low concentrations of potentially deleterious metals, including arsenic, calcium, and magnesium, were detected. Given that 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 2003 [DIRS 161236], Section 6.5). On this basis, the effects of trace metals on general corrosion of the drip shield are included in TSPA although the effects are not explicitly and separately modeled.

The drip shield outer surface may be exposed to a more complicated chemistry and geometry than the drip shield inner surface since dust and/or mineral films (from evaporation of dripping water) 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 nor significant dust film formation (BSC 2003 [DIRS 161236], Sections 6.1 and 6.3). Therefore, the general corrosion of the inner surface and the outer surface of the drip shield are modeled by using different sets of corrosion data (BSC 2003 [161236], Sections 6.1 and 6.3). Based on these test results, the corrosion (or oxidation) rates range from 0 to 113 nm/year with the 50th percentile being 16 nm/year for the underside of the drip shield (BSC 2003 [DIRS 161236], Section 6.5.3). For the outer surface of the drip shield, the corrosion rates range from 0 to 320 nm/year with the 50th percentile being about 25 nm/year (BSC 2003 [DIRS 161236], Section 6.5.3).

The variations in the drip shield general corrosion rate distributions are considered to be entirely due to uncertainty (BSC 2003 [DIRS 161236], Section 6.3.4). For each realization of the Integrated Waste Package Degradation Model (BSC 2003 [DIRS 161317]), 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 which is applicable to the drip shield underside and one general corrosion rate value is sampled from the general corrosion rate distribution which is applicable to the drip shield outer surface). The two sampled values are then applied one to the drip shield outer surface and one to the Drip shield inner surface 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/year 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/year (BSC 2003 [DIRS 161236], Section 6.3.5); therefore, the earliest possible drip shield failure by general corrosion is about 35,000 years.

The general corrosion rate distributions for the drip shields are incorporated into the Integrated Waste Package Degradation Model (BSC 2003 [DIRS 161317]). The output from the Integrated Waste Package Degradation Model 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 Model is used directly in the TSPA-LA analysis.

Additional aqueous corrosion processes are addressed in other FEPs: Localized Corrosion of Drip Shields (FEP 2.1.03.03.0B), Stress Corrosion Cracking (SCC) of Drip Shields (FEP 2.1.03.02.0B), Hydride Cracking of Drip Shields (FEP 2.1.03.04.0B), and Microbially

Influenced Corrosion (MIC) of Drip Shields (FEP 2.1.03.05.0B). These processes are described in BSC 2003 [DIRS 161236], BSC 2003 [DIRS 161234], and BSC 2003 [DIRS 161759]. As discussed in FEP 2.1.03.05.0B, Microbially Influenced Corrosion of Drip Shields, the drip shield is considered to be immune to Microbially Influenced Corrosion. Also, as discussed in FEP 2.1.11.06.0B, Thermal Sensitization of Drip Shields, aging and phase instability has no effect on drip shield degradation processes.

6.2.3.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236])

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.4 Stress Corrosion Cracking of Waste Packages

6.2.4.1 Feature, Event, and Process Number

FEP 2.1.03.02.0A

6.2.4.2 Feature, Event, and Process Description

At specific locations where waste packages become wet and are stressed, stress corrosion cracking ensues. The possibility of SCC under dry conditions or due to thermal stresses are also addressed as part of this FEP.

6.2.4.3 Descriptor Phrases

Waste package closure welds (stress corrosion cracking)

Waste package fabrication welds (stress corrosion cracking)

Threshold relative humidity (stress corrosion cracking)

6.2.4.4 Screening Decision

Included

6.2.4.5 Screening Argument

N/A

6.2.4.6 Total System Performance Assessment Disposition

SCC of the waste package outer barrier closure weld regions is included in TSPA as part of waste package degradation analyses. The *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model report (BSC 2003 [DIRS 161234], Section 8) provides input to the TSPA for waste package degradation.

As discussed in the *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model report (BSC 2003 [DIRS 161234], Section 6), the slip dissolution/film rupture model was used to assess the failure (or lack of it) of the waste package due to the SCC crack propagation for given manufacturing cracks and/or cracks initiated by the combined effects of stress and environment. The threshold stress intensity factor is based on the theory that there exists a threshold value for the stress intensity factor at the crack tip below which a pre-existing crack or flaw does not grow. The stress intensity factor provides a criterion for determining if a SCC crack will reach an arrest state or enter the propagation phase.

The application of the SCC models to the waste package and drip shield also requires input of weld residual stress profiles and stress intensity factor profiles along with uncertainty and variability (BSC 2003 [DIRS 161234]). These input data were developed for the 25-mm outer lid (subjected to laser peening) and the as-welded 10-mm middle lid. This process model report (BSC 2003 [DIRS 161234]) also provides other needed 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, size, density and orientation distributions for manufacturing flaws or defects, and an estimate of crack opening size. This SCC treatment applies to both CSNF and co-disposed waste packages.

Because, among other exposure condition parameters, tensile stress is required to initiate SCC and the waste package closure welds are the only places under such tensile stresses, only the waste package closure welds are considered subject to SCC (BSC 2003 [DIRS 161234], Section 6.4.2). Welds are the most susceptible to SCC because (1) welding can produce high tensile residual stress in the weld; (2) pre-existing 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 non-equilibrium brittle phases, which could enhance material susceptibility to SCC (BSC 2003 [DIRS 161234], Section 6.4.2). SCC of the fabrication welds of the waste package outer barrier will not occur due to the resistance of Alloy 22 (UNS N06022) to SCC when under tensile stress and because the fabrication welds will be fully annealed before waste is loaded into the waste containers (Plinski 2001 [DIRS 156800], Section 8.1.7). It is recognized that 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 the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]).

The *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model report (BSC 2003 [DIRS 161234], Section 6.2.1) describes the results of SCC crack initiation measurements under constant load conditions while immersed in basic saturated water. Alloy 22 (UNS N06022) exhibits excellent SCC resistance since failure 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 and 90°C in the Long Term Corrosion Test Facility (BSC 2003 [DIRS 161234]). No evidence of SCC initiation has been observed in these U-bend specimens after five years of exposure.

The presence of stable "liquid" water is required to initiate corrosion processes (including SCC) that are supported by electrochemical corrosion reactions. Threshold relative humidity is used in the waste package degradation analysis to simulate such a corrosion initiation condition. Under conditions with the relative humidity below the threshold value, SCC will not occur.

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 Model (BSC 2003 [DIRS 161317]). The output from the Integrated Waste Package Degradation Model 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 Model is used directly in the TSPA-LA analysis.

Lead 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]). One should note that the results of 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 investigating 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 the *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model report (BSC 2003 [DIRS 161234], Section 6.3.4). SCC initiation test results were obtained using Slow Strain Rate Tests, at 76-95°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]). In this latter experiment a Basic Saturated Water solution with a pH ~ 12 at 110°C was used. After 8670 hours of exposure in the lead-free solution,1000 ppm Pb (as $PbNO_3$) was added to the test solution and SCC crack growth rates were monitored for ~1800 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 ground waters is expected to have little consequence on Alloy 22 (UNS N06022) SCC processes.

6.2.4.7 Supporting Analysis Reports

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2003 [161234])

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [161317])

6.2.5 Stress Corrosion Cracking of Drip Shields

6.2.5.1 Features, Events, and Process Number

FEP 2.1.03.02.0B

6.2.5.2 Features, Events, and Process Description

At specific locations where drip shields become wet and are stressed, stress corrosion cracking ensues. The possibility of SCC under dry conditions or due to thermal stresses is also addressed as part of this FEP.

6.2.5.3 Descriptor Phrases

Drip shield fabrication welds (stress corrosion cracking)

Drip shield cracks (stress corrosion cracking)

Drip shield crack plugging

6.2.5.4 Screening Decision

Excluded – Low Consequence

6.2.5.5 Screening Argument

For the drip shields, all the 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 2003 [DIRS 161234], Section 6.3.7). However, the drip shields are subject to SCC under the action of seismic-induced loading and rockfalls. Seismic effects on drip shield degradation are discussed in FEP 2.1.07.01.0A, Rockfall and FEP 1.2.03.02.0A, Seismic Ground Motion Damages EBS Components (BSC 2004 [DIRS 167253]). In the nominal case (in the absence of seismic-induced loading and rockfalls) even if SCC of the drip shield were to occur, cracks in passive alloys, such as Titanium Grade 7, tend to be tight (i.e., small crack opening displacement) (BSC 2003 [DIRS 161234], Section 6.3.7). The opposing sides of through-wall cracks will continue to corrode at very low passive corrosion rates until the gap region of the tight crack opening is "plugged" by corrosion products and precipitates such as carbonate minerals. As discussed in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model document (BSC 2003 [DIRS 161234], Section 6.3.7), SCC cracks are sealed in a few hundred years at most when water is allowed to flow through the cracks at the expected low flow rate. When the cracks are bridged by water, the sealing process may take thousands of years, but no flow occurs since the water is held by capillary forces. Following plugging of the crack, any solution flow through the crack would be dominated by an efficiency factor determined by the ratio of solution run-off on the drip shield surface compared to through crack flow which in turn is determined by scale porosity/permeability. Because of the expected high density of the calcite deposits and lack of pressure gradient to drive water through the crack, the probability of solution flow through the crack would approach zero (BSC 2003 [DIRS 161234], Section 6.3.7). 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 its intended design purpose. 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.

6.2.5.6 Total System Performance Assessment Disposition

N/A

6.2.5.7 Supporting Analysis Reports

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2003 [DIRS 161234])

6.2.6 Localized Corrosion of Waste Packages

6.2.6.1 Features, Events, and Process Number

FEP 2.1.03.03.0A

6.2.6.2 Features, Events, and Process Description

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

6.2.6.3 Descriptor Phrases

Pitting corrosion of waste packages

Crevice corrosion of waste packages

Effects of trace metals on localized corrosion of waste packages

6.2.6.4 Screening Decision

Included

6.2.6.5 Screening Argument

N/A

6.2.6.6 Total System Performance Assessment Disposition

Localized corrosion (pitting and crevice) is a corrosion mode that could lead to eventual compromise of waste packages in the repository. As discussed in the *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* model report (BSC 2003 [DIRS 161235]), localized corrosion of the waste package outer barrier (Alloy 22) (UNS N06022) is not likely to occur under repository-relevant exposure conditions particularly during the time period when the drip shield performs its design function and prevents the seepage water from directly contacting the underlying waste package. Since the probability of occurrence of localized corrosion has not been quantified, localized corrosion initiation and propagation models are included in TSPA to be conservative.

A possible scenario for the waste package outer barrier to be subjected to localized corrosion in the repository is the concurrent occurrence of the drip shield failure and direct contact of the waste package with seepage water during the first few hundred years of active thermal distribution after closure. The seepage water with a characteristic chemistry could evolve to highly chloride-containing brines by evaporative concentration. After the active thermal perturbation period, the waste package temperature slowly decreases with time. Once it cools to the temperature that is lower than the minimum temperature required for localized corrosion initiation in highly concentrated chloride-containing brines, the waste packages are completely immune to localized corrosion.

Localized corrosion of the waste package outer barrier is modeled with two model components: initiation model and propagation model (BSC 2003 [161235], Section 6.4.4; BSC 2003 [DIRS 161317], Section 6.3.6). The initiation model assumes that localized corrosion of the waste package outer barrier occurs when the open circuit potential, or corrosion potential (E_{corr}), is equal to or greater than a certain critical 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 (e.g., E_{corr} and $E_{critical}$) could be affected by the sample configuration (crevice, disk, or rod), metallurgical conditions (mill or annealed or welded), and exposure condition (temperature, pH, chloride ion concentration, 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. The crevice repassivation model is expressed as follows.

$$E_{rcrev} = E^{o}_{rcrev} + \Delta E^{NO_3^-}_{rcrev} \quad \text{(Eq. 5)}$$

where E^{o}_{rcrev} is the crevice repassivation potential in the absence of nitrate ions (which tend to inhibit localized corrosion (LC) initiation), and $\Delta E^{NO_3^-}_{rcrev}$ is the crevice repassivation potential changes resulted from the inhibiting effect of nitrate ion in solution (BSC 2003 [DIRS 161235], Section 6.4.4).

Further, the functional form for $\Delta E_{rp}^{NO_3^-}$ was evaluated by fitting the crevice repassivation potential differences between the E_{rcrev} data with and without nitrate ions for the same chloride and temperature conditions. The effect of nitrate ions on the crevice repassivation potential is represented as follows.

$$\Delta E_{rp}^{NO_3^-} = b_o + b_1[NO_3^-] + b_2 \frac{[NO_3^-]}{[Cl^-]} \qquad \text{(Eq. 6)}$$

The effect of the interaction of the competing aggressive ion (chloride ion) and inhibiting ion (nitrate ion) on the crevice repassivation potential is represented with the ratio of the concentrations of the competing ions (BSC 2003 [DIRS 161235], Section 6.4.4).

Nitrate ions have a strong inhibitive effect on localized corrosion of Alloy 22 in chloride containing solutions (BSC 2003 [DIRS 161235], Section 6.4.4; Dunn and Brossia 2002 [DIRS 162213]). It is known that anions containing nitrogen, phosphorus and sulfur that are abundant in the repository groundwater exhibit varying degrees of inhibitive effects (Thomas 1994 [DIRS 120498]). An important function of inhibitive anions is to 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], page 17:54).

A model for the effect of inhibitive anions on the crevice repassivation potentials of Alloy 22 (UNS N06022) (i.e. increase of the crevice repassivation potentials) was developed using the crevice repassivation potential data measured for the test solutions containing varying concentrations of chloride and nitrate ions. The environments tested cover temperatures from 60°C to 130°C, chloride concentrations from 2.6 *m* to 20.8 *m*, and nitrate concentrations from 0.03 *m* to 2.1 *m* (BSC 2003 [DIRS 161235], Section 6.4.4). In the model the interaction between the competing aggressive anion (chloride ion) and inhibitive anion (nitrate ion) on the crevice repassivation potential is represented with the ratio of the concentrations of the two competing ions and the concentration of nitrate ion. 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 2003 [DIRS 161235], Section 8.3).

The analysis of the crevice corrosion initiation model for the Alloy 22 (UNS N06022) waste package outer barrier has shown that for a condition of neutral chloride-containing brines at 95°C (10 *m* chloride and pH 7), the conservative lower bound of the nitrate to chloride concentration ratio for the crevice corrosion immunity is 0.03, and the conservative lower bound ratio for a condition of highly corrosive chloride-containing brines (10 *m* chloride and pH 3) at 95°C is 0.18 (BSC 2003 [DIRS 161235], Section 6.4.4, Figures 6-53 and 6-54).

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 and 90°C in the Long Term Corrosion Test Facility. The nitrate to chloride concentration ratio of these solutions is about 0.5. None of the crevice samples have 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 2003 [DIRS 161235], Section 7.3, Table 7-2).

A recent 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 has 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]).

All the experimental data and model analyses that are available to date indicate that the lower bound of the nitrate to chloride concentration ratio for the immunity of Alloy 22 (UNS N06022) to localized corrosion in chloride-containing brines is about 0.2. Therefore, a nitrate to chloride concentration ratio of 1.0 is selected as the conservative bounding measure for total immunity of the Alloy 22 waste package outer barrier to crevice corrosion for the environmental conditions expected in the repository.

Implementation of the localized corrosion model will be carried outside of the main TSPA-LA model. The output of this analysis will be one or more environmentally independent uncertainty distributions for the number (or fraction) of packages that experience localized corrosion. These uncertainty distributions for the number of localized corrosion packages will then be incorporated and sampled in the main TSPA-LA model. The main premise is that localized corrosion is a rare occurrence and need only be applied to a small number of waste packages in each TSPA realization. This localized corrosion treatment applies to both CSNF and Co-disposed waste packages.

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 analyses. These same solutions were considered in developing the localized corrosion initiation criterion (BSC 2003 [DIRS 161235], Section 6.4.4; DTN: LL021012712251.021 [DIRS 163112]). The small amounts of trace metals identified in the acidified waters were appropriately considered in analyzing localized corrosion of waste packages.

6.2.6.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.7 Localized Corrosion of Drip Shields

6.2.7.1 Feature, Event, and Process Number

FEP 2.1.03.03.0B

6.2.7.2 Feature, Event, and Process Description

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

6.2.7.3 Descriptor Phrases

Pitting corrosion of drip shields

Crevice corrosion of drip shields

Effects of trace metals on localized corrosion of drip shields

6.2.7.4 Screening Decision

Excluded – Low Consequence

6.2.7.5 Screening Argument

As discussed in the *General Corrosion and Localized Corrosion of the Drip Shield* model report (BSC 2003 [DIRS 161236], Section 6.4), localized corrosion of the drip shield will not occur under the exposure conditions in the repository. A relationship between exposure parameters (temperature, chloride ion concentration, and pH) and the difference between the critical potential and the corrosion potential was developed. The localized corrosion model for the titanium drip shield assumes that localized attack occurs if the open circuit corrosion potential exceeds the threshold potential for breakdown of the passive film (BSC 2003 [DIRS 161236]). The critical potential versus temperature and composition model of various test media indicates that localized corrosion of Titanium Grade 7 would not initiate in repository-relevant environments even at pH values as high as 14 (BSC 2003 [DIRS 161236]).

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* model report (BSC 2003 [DIRS 161236], Section 6.3.8, 6.4.4, and 6.4.5). The general corrosion rates for Titanium Grade 7 are low when in the presence of these species or when crevices are present on the drip shield.

As discussed in FEP 2.1.03.01.0B, General Corrosion of Drip Shields, acidified waters that contain trace metals such as arsenic, calcium, and magnesium were used to develop the Titanium Grade 7 general corrosion rates used in the TSPA analyses. These same solutions were used in developing the localized corrosion initiation criterion (BSC 2003 [DIRS 161236], Section 6.4; 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.

Since localized corrosion of Titanium Grade 7 will not initiate under the expected repository conditions and general corrosion rates are low when crevices are present, this FEP can be excluded based on low consequence because its omission will not have a significant effect on radiological exposures to the RMEI and radionuclide releases to the accessible environment.

6.2.7.6 Total System Performance Assessment Disposition

N/A

6.2.7.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236])

6.2.8 Hydride Cracking of Waste Packages

6.2.8.1 Feature, Event, and Process Number

FEP 2.1.03.04.0A

6.2.8.2 Feature, Event, and Process Description

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

6.2.8.3 Descriptor Phrases

Hydrogen induced cracking (waste packages)

Hydride embrittlement (waste packages)

Galvanic coupling (drip shield-ground support)

Hydride cracking (waste package)

6.2.8.4 Screening Decision

Excluded – Low Consequence

6.2.8.5 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 (BSC 2003 [DIRS 161759], Section 6.1.1). It may be more appropriate to use a more general term, HIC, 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.

HIC of the waste package outer barrier (Alloy 22) [UNS N06022])is not considered to be a credible degradation mechanism under repository-relevant exposure conditions. Handbook data (ASM International 1987 [DIRS 103753], p. 650-652) indicate that fully annealed nickel-base alloys, such as Alloy 22, may be immune to HIC. Similar to the drip shield material, the extremely low corrosion rates exhibited by nickel alloys are not sufficient to generate enough hydrogen to cause HIC. HIC of these alloys is generally not observed unless their yield strengths are increased through heavy cold working or certain heat treatments. Several Ni-Cr-Mo alloys with compositions and properties similar to Alloy 22 (Alloy C-276, Alloy C-4, and Alloy 625) maintain their resistance to HIC even when heavily cold worked to yield strengths in excess of 1240 MPa (180 ksi) (ASM International 1987 [DIRS 103753], p. 169). Cold worked Ni-Cr-Mo alloys are not susceptible to HIC 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 and/or grain-boundary segregation of deleterious elements such as phosphorous and sulfur which can increase susceptibility to HIC (ASM International 1987 [DIRS 103753], p. 169). However, since the waste package temperature never exceeds 190°C (CRWMS M&O 2001 [DIRS 154594], Section 6.3.1) 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 about HIC of the waste package are applicable to both CSNF and Co-disposed waste packages.

As discussed in FEP 2.1.09.09.0A, Electrochemical Effects in EBS, electrical contact between the Alloy 22 waste package outer barrier and Titanium Grade 7 drip shield is not expected and would not be more detrimental to corrosion due to the similarity of these materials' corrosion potential.

In summary, HIC of Alloy 22 is not expected to occur under the anticipated repository conditions. Even in the unlikely case of HIC resulting from galvanic coupling, the corrosion behavior of Alloy 22 would not be affected. Therefore, this FEP can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

6.2.8.6 Total System Performance Assessment Disposition

N/A

6.2.8.7 Supporting Analysis Reports

Hydrogen Induced Cracking of Drip Shield (BSC 2003 [DIRS 161759])

6.2.9 Hydride Cracking of Drip Shields

6.2.9.1 Feature, Event, and Process Number

FEP 2.1.03.04.0B

6.2.9.2 Feature, Event, and Process Description

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

6.2.9.3 Descriptor Phrases

Hydrogen induced cracking (drip shield)

Hydride embrittlement (drip shield)

Galvanic coupling (drip shield-ground support)

Drip shield crack plugging

Hydride cracking (drip shield)

6.2.9.4 Screening Decision

Excluded – Low Consequence

6.2.9.5 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. It may be more appropriate to use a more general term, HIC, to refer to the impact of hydrogen on waste package and drip shield materials. HIC 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 2003 [DIRS 161759], Section 6.1.2):

1. A mechanism for generating nascent (atomic) hydrogen on the surface.

2. Metal temperature above approximately 80^{o}C (175^{o}F) where the diffusion rate of hydrogen into α-titanium is significant.

3. Solution pH less than 3 or greater than 12, or impressed potentials more negative than -0.7 V (Saturated Calomel Electrode).

As discussed in *General Corrosion and Localized Corrosion of Drip Shield* (BSC 2003 [DIRS 161236], Section 6.4) and 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 current repository design (BSC 2004 [DIRS 167040]), 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 high dose rate (> 10^2 Gy/h) and high temperature (> 150^oC) (BSC 2003 [DIRS 161759], Section 6.1.2). These conditions are unattainable in the repository (BSC 2003 [DIRS 161759], Section 6.1.2). At certain repository locations, where temperatures are high (≥ 80°C) and concentrated groundwater is present, conditions two and three may also be satisfied. When all three conditions are present simultaneously, hydrogen absorption can be anticipated.

As discussed in the *Hydrogen Induced Cracking of Drip Shield* model report (BSC 2003 [DIRS 161759], Section 8), a simple and conservative model was developed to evaluate the effects of HIC 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, noticeable hydrogen induced cracking of Titanium Grade 7 is not expected. This is mainly due to the existence of the oxide film and the high critical value, H_c, due to palladium (Pd) addition. This passive film acts as a good transport barrier to hydrogen absorption; the impermeability of this film to hydrogen absorption will be improved during a period of dry thermal oxidation expected under repository conditions (BSC 2003 [DIRS 161759], Section 6.1.5). 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 1000 $\mu g.g^{-1}$ (BSC 2003 [DIRS 161759], 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 1000 $\mu g.g^{-1}$ and could be much higher (BSC 2003 [DIRS 161759], Section 6.1.3). The hydrogen concentration in the drip shield from passive corrosion 10,000 years after emplacement is 755 $\mu g.g^{-1}$ resulting from a conservative estimate (BSC 2003 [DIRS 161759], Section 8). This is below the threshold concentration, and would not result in hydrogen induced cracking or any degradation of fracture toughness.

In the current repository design, hydrogen generation may be caused by the galvanic couple formed between the titanium 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 high (≥ 80°C) and concentrated ground waters are present, then the formation of locally hydrided "hot spots" are possible. However, the effect of these locally hydrided regions is negligible because (BSC 2003 [DIRS 161759], Section 6.3.2):

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

2. The intermittent nature of seepage at high 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.

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

4. Both the titanium drip shield and the steel component surfaces will experience a considerable period of dry high temperature ($\geq 85°C$). This will leave both 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 Pd addition reducing the tendency for hydrogen induced cracking (BSC 2003 [DIRS 161759], Sections 6.1.3 and 6.1.5).

In summary, HIC of 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 would be negligible. Therefore, this FEP can be excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

6.2.9.6 Total System Performance Assessment Disposition

N/A

6.2.9.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Drip Shield (BSC 2003 [DIRS 161236])

Hydrogen Induced Cracking of Drip Shield (BSC 2003 [DIRS 161759])

6.2.10 Microbially Influenced Corrosion of Waste Packages

6.2.10.1 Feature, Event, and Process Number

FEP 2.1.03.05.0A

6.2.10.2 Feature, Event, and Process 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.

6.2.10.3 Descriptor Phrases

Microbially influenced corrosion of waste packages

6.2.10.4 Screening Decision

Included

6.2.10.5 Screening Argument

N/A

6.2.10.6 Total System Performance Assessment Disposition

MIC is included in TSPA as part of the waste package degradation analysis (BSC 2003 [DIRS 161235], Table 6-1). Waste package MIC is discussed in the *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* model report (BSC 2003 [DIRS 161235]). The potential effect of MIC on waste package corrosion is analyzed with an enhancement factor approach, i.e. MIC increases the general corrosion penetration rate. In this approach, the abiotic corrosion rate is multiplied by the enhancement factor when the exposure conditions in the emplacement drift warrant significant microbial activity (BSC 2003 [DIRS 161235], Section 6.4.3.5 and Table 6-1). The MIC enhancement factor is taken to be uniformly distributed between 1 and 2 (BSC 2003 [DIRS 161235], Section 6.4.5). The MIC factor 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 2003 [DIRS 161235], Section 6.4.5). The general corrosion rate enhancement factor is applied to the entire waste package surface (BSC 2003 [DIRS 161235], Section 6.4.5) when the relative humidity threshold is satisfied. This treatment of MIC applies to both CSNF and co-disposed waste packages.

In the Integrated Waste Package Degradation Model (BSC 2003 [DIRS 161317]), the general corrosion rate enhancement factor is sampled once per realization, i.e. the variation in the general corrosion rate MIC enhancement factor is entirely due to uncertainty (BSC 2003 [DIRS 161235], Section 6.4.5), and applied to the entire waste package surface. The output from the Integrated Waste Package Degradation Model 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 Model is used directly in the TSPA-LA analysis.

6.2.10.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.11 Microbially Influenced Corrosion of Drip Shields

6.2.11.1 Feature, Event, and Process Number

FEP 2.1.03.05.0B

6.2.11.2 Feature, Event, and Process 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.

6.2.11.3 Descriptor Phrases

Microbially influenced corrosion of drip shields

6.2.11.4 Screening Decision

Excluded – Low Consequence

6.2.11.5 Screening Argument

MIC of titanium is discussed in the *General Corrosion and Localized Corrosion of the Drip Shield* report (BSC 2003 [DIRS 161236], Section 6.5.2). Corrosion handbooks and literature reviews generally state that titanium alloys are immune to MIC (Revie 2000 [DIRS 159370], Chapter 47; Little and Wagner 1996 [DIRS 131533]; Brossia et al. 2001 [DIRS 159836], Section 4.1.3). It is the remarkable stability of the TiO_2 passive film formed on titanium alloys which 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). It is clear from Figure 20 and Figure 21 of the *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2003 [DIRS 161236], Section 6.4.3) that 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 density of defects (Shoesmith and Ikeda 1997 [DIRS 151179], Section 3 and 6). According to Shoesmith et al. (1995 [DIRS 117892]), the initiation of crevice corrosion under biofilms has never been observed for titanium. Lastly, microbial growth in the repository will likely be limited by the availability of nutrients (BSC 2003 [DIRS 161235], Section 6.4.5).

MIC is expected to have 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.

6.2.11.5 Total System Performance Assessment Disposition

N/A

6.2.11.6 Supporting Analysis Reports

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236])

6.2.12 Internal Corrosion of Waste Packages Prior to Breach

6.2.12.1 Feature, Event, and Process Number

FEP 2.1.03.06.0A

6.2.12.2 Feature, Event, and Process 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.

6.2.12.3 Descriptor Phrases

Internal corrosion of commercial spent nuclear fuel

Internal corrosion of defense spent nuclear fuel

Chemical effects from radiolysis in-package

6.2.12.4 Screening Decision

Excluded – Low Consequence

6.2.12.5 Screening Argument

The *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 failure in the repository. This FEP is related 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 filled with an inert gas (Helium) prior to closure, displacing water and oxygen (necessary components for corrosion) from inside the package (CRWMS M&O 2000 [DIRS 123881], p. II-30). The inert gas environment within the package will result in a negligible amount of corrosion degradation prior to the breach of the waste packages.

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 source of this residual moisture is primarily 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 2003 [DIRS 163855]) which will corrode in preference to the Alloy 22 (UNS N06022) waste package outer barrier and the 316 stainless steel inner vessel (e.g., ASM International [DIRS 103753], p. 557). Thus, no significant corrosion damage to the waste package outer barrier and 316 stainless steel 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 interim storage. 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 interim storage. Damaged DSNF will be placed in high integrity 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 316 stainless steel inner vessel. This FEP is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

6.2.12.6 Total System Performance Assessment Disposition

N/A

6.2.12.7 Supporting Analysis Reports

N/A

6.2.13 Mechanical Impact on Waste Package

6.2.13.1 Feature, Event, and Process Number

FEP 2.1.03.07.0A

6.2.13.2 Feature, Event, and Process Description

Mechanical impact (dynamic loading) on the waste package is 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 impacts are addressed in a separate FEP.

6.2.13.3 Descriptor Phrases

Rockfall (waste package failure)

Rockfall (waste package damage)

Internal gas pressure (waste package damage)

Volume increase of corrosion products (waste package damage)

Rockfall (drip shield contacts waste package)

6.2.13.4 Screening Decision

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

Excluded – Low Probability (Rockfall)

6.2.13.5 Screening Argument

The conclusions reached in this Section are equally applicable to CSNF and co-disposed 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 the ASME code requirements for the allowable tensile strength of the waste package material (CRWMS M&O 2000 [DIRS 144128], Section 6.2.2.8). 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 2003 [DIRS 161235], Section 6.4.2), indicate that, even under very conservative assumptions, the growth of the corrosion product (Cr_2O_3) oxide layer is not thick enough to produce enough pressure to cause mechanical damage to the Type 316 stainless steel inner vessel or the Alloy 22 (UNS N06022) outer barrier. Therefore, waste package damage from swelling corrosion products is excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

Rockfall: Mechanical damage of the waste package by rockfall is discussed in greater detail under FEP 2.1.07.01.0A, Rockfall. The Emplacement Drift System design criteria require that the drip shield protect the waste package from rockfalls during postclosure (BSC 2002 [DIRS 159292], Section 2.1.3.2). Because the drip shield provides adequate protection to the waste packages from rockfall (refer to 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 the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]).

6.2.13.6 Total System Performance Assessment Disposition

N/A

6.2.13.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

6.2.14 Mechanical Impact on Drip Shield

6.2.14.1 Feature, Event, and Process Number

FEP 2.1.03.07.0B

6.2.14.2 Feature, Event, and Process Description

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

6.2.14.3 Descriptor Phrases

Rockfall (drip shield separation)

Rockfall (drip shield damage)

Rockfall (drip shield contacts waste package)

6.2.14.4 Screening Decision

Excluded – Low Consequence

6.2.14.5 Screening Argument

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 requires that the drip shield protect the waste package from rockfalls during postclosure (BSC 2002 [DIRS 159292], Section 2.1.3.2). According to the *Drip Shield Structural Response to Rock Fall* calculation (BSC 2003 [DIRS 162598], Section 6), LS-DYNA analysis shows that the deflection of the drip shield due to a 11.5 MT 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 MT rockfall event is 254 mm (BSC 2003 [DIRS 168489]). The minimum gap between the drip shield and waste package outer barrier was calculated to be about 367 mm based on the following equation:

$$Gap = \mathrm{h}_{\mathrm{int\ ds}} - \mathrm{dist}_{\mathrm{inv}} - \frac{\mathrm{d}_{\mathrm{wp}}}{2} \quad \text{(Eq. 7)}$$

where

$h_{int\ ds}$ = interior height of drip shield (BSC 2003 [168489], Table 1) (2716. mm)
$dist_{inv}$ = distance from top of invert to centerline of waste package for 5 DHLW (BSC 2004 [DIRS 167040]) (1286.1 mm)
d_{wp} = diameter of waste package (5 DHLW/DOE SNF – Short) (BSC 2004 [DIRS 167207], Table 1) (2126 mm)

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 the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]).

6.2.14.6 Total System Performance Assessment Disposition

N/A

6.2.14.7 Supporting Analysis Reports

N/A

6.2.15 Early Failure of Waste Packages

6.2.15.1 Feature, Event, and Process Number

FEP 2.1.03.08.0A

6.2.15.2 Feature, Event, and Process Description

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

6.2.15.3 Descriptor Phrases

Waste package closure welds (stress corrosion cracking)

Waste package fabrication welds (stress corrosion cracking)

Waste package emplacement error

6.2.15.4 Screening Decision

Included

6.2.15.5 Screening Argument

N/A

6.2.15.6 Total System Performance Assessment Disposition

The Analysis of Mechanisms for Early Waste Package/Drip Shield Failure (BSC 2003 [DIRS 164475]) evaluates several mechanisms for early failure of the waste package. Of these mechanisms, weld flaws, improper heat treatment, improper laser peening, and improper handling of waste packages were determined to be necessary for inclusion in TSPA models (BSC 2003 [DIRS 164475], Section 8).

As discussed in FEP2.1.03.02.0A, Stress Corrosion Cracking of Waste Packages, manufacturing defects (weld flaws) on waste packages act as sites for initiation of SCC (BSC 2003 [DIRS 161234]; BSC 2003 [DIRS 161317]). Manufacturing defects are included in TSPA analysis through the SCC analysis of waste packages.

Early failure (due to improper heat treatment, improper laser peening, and improper handling of waste packages) is included in the waste package performance analysis. Improper heat treatment results primarily from improper stress relief annealing and the consequence of improper heat treatment is assumed to be immediate failure upon initiation of degradation processes (BSC 2003 [DIRS 164475], Section 6.4.3). 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 (BSC 2003 [DIRS 164475], Section 6.4.4). Improper handling of the waste packages may lead to gouges in the waste package outer surface and provide sites for stress corrosion cracks (BSC 2003 [DIRS 164475], Section 6.4.6). Early failure (due to improper heat treatment, improper laser peening, and improper handling of waste packages) is included in TSPA analysis.

The number of early failed waste packages per realization is given 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 2003 [DIRS 164475], Section 7, Table 20). Since an improperly heat treated waste package might be susceptible to aging and phase stability, it is not possible to identify a single and specific mechanism of degradation. For these reasons, the following recommendations are made in the *Analysis of Mechanisms for Early Waste*

Package/Drip Shield Failure (BSC 2003 [DIRS 164475], Section 6.4.8) for evaluating waste package early failure

- 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.

This early failure treatment applies to both CSNF and Co-disposed 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 the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]).

6.2.15.7 Supporting Analysis Reports

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2003 [DIRS 161234]) (This AR did not explicitly list this FEP as an Included FEP (BSC 2003 [DIRS 161234], Section 1.1). However, this Analysis Report does provide relevant supporting information describing the implementation of this FEP in TSPA-LA).

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.16 Early Failure of Drip Shields

6.2.16.1 Feature, Event, and Process Number

FEP 2.1.03.08.0B

6.2.16.2 Feature, Event, and Process Description:

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

6.2.16.3 Descriptor Phrases

Drip shield fabrication welds (stress corrosion cracking)

Drip shield emplacement error

6.2.16.4 Screening Decision

Excluded – Low Consequence

6.2.16.5 Screening Argument

The *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (BSC 2003 [DIRS 164475]) evaluates 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. All of these mechanisms have a low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment (BSC 2003 [DIRS 164475], Section 6.3).

The consequence of weld flaws, base metal flaws, improper heat treatment, and damage by mishandling is SCC (BSC 2003 [DIRS 164475], Section 6.4). Among other exposure condition parameters, the surface stress at the tip of a SCC crack must exceed the critical threshold stress in order to initiate SCC (BSC 2003 [DIRS 161234], Section 6.2.1).

For the drip shields, all the 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 2003 [DIRS 161234], Section 6.3.7). However, the drip shields are subject to SCC under the action 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 167253]). In the nominal case (in the absence of seismic-induced loading and rockfalls) even if SCC of the drip shield were to occur, cracks in passive alloys, such as Titanium Grade 7, tend to be tight (i.e., small crack opening displacement) (BSC 2003 [DIRS 161234], Section 6.3.7). As the crack grows through-wall, the tensile stresses normal to the crack walls are relieved, and the resulting crack faces continue to corrode at very low passive corrosion rates until the gap region of the tight crack opening is "plugged" by corrosion products and precipitates such as carbonate minerals. As discussed in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model report (BSC 2003, [DIRS 161234], Section 6.3.7), SCC cracks are sealed in a few hundred years at most when water is allowed to flow through the cracks at the expected low flow rate. When the cracks are bridged by water, the sealing process may take thousands of years, but no flow occurs since the water is held by capillary forces. Following plugging of the crack, any solution flow through the crack would be dominated by an efficiency factor determined by the ratio of solution run-off on the drip shield surface compared to through crack flow which in turn is determined by scale porosity/permeability. Because of the expected high density of the calcite deposits and lack of pressure gradient to drive water through the crack, the probability of solution flow through the crack would approach zero (BSC 2003 [DIRS 161234], Section 6.3.7). 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 its intended design purpose.

The use of improper weld or base metal material is possible in the repository (BSC 2003 [DIRS 164475], Section 6.3.3), however, due to the strict controls that will govern the fabrication of the DS, it is expected that the material composition of the improper weld or base metal material will differ only slightly from the intended composition (BSC 2003 [DIRS 164475], 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 2003 [DIRS 164475], Section 6.4.2).

The probability of drip shield surface contamination is also evaluated in the *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (BSC 2003 [DIRS 164475], Section 6.3.5). It is found that the consequence of drip shield surface contamination is not significant from a corrosion standpoint (BSC 2003 [DIRS 164475], Section 6.4.5). On this basis, drip shield surface contamination is not significant.

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.

Thus, manufacturing defects in the drip shield and early failure mechanisms (not including drip shield emplacement error, see 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.

6.2.16.6 Total System Performance Assessment Disposition

N/A

6.2.15.7 Supporting Analysis Reports

Analysis of Mechanisms for Early Waste Package/Drip Shield Failure (BSC 2003 [DIRS 164475])

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2003 [DIRS 161234])

6.2.17 Copper Corrosion in Engineered Barrier System

6.2.17.1 Feature, Event, and Process Number

FEP 2.1.03.09.0A

6.2.17.2 Feature, Event, and Process Description

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

6.2.17.3 Descriptor Phrases

Corrosion of gantry rail system

6.2.17.4 Screening Decision

Excluded – Low Consequence

6.2.17.5 Screening Argument

The repository does not use copper containers. However, a small amount of copper may be present as part of the gantry rail system (BSC 2003 [DIRS 164101], Table 3). This will 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 2003 [DIRS 168489], Table 6) and is designed to keep the waste package from contacting other dissimilar metals. 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 2003 [DIRS 168489], Table 5). 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 with titanium and hydrogen absorption. The potential for hydrogen absorption, or hydrogen induced cracking, is considered low since Titanium Grade 7 contains Pd and since a protective oxide film forms on the surface of the drip shield, which hinders hydrogen absorption (BSC 2003 [DIRS 161759], Section 6.1.3). When Titanium Grade 7 is cathodically polarized at -1.2 V (vs. 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 2003 [DIRS 161759], 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.

6.2.17.6 Total System Performance Assessment Disposition

N/A

6.2.17.7 Supporting Analysis Reports

Hydrogen Induced Cracking of Drip Shield (BSC 2003 [DIRS 161759])

6.2.18 Healing of Waste Packages

6.2.18.1 Feature, Event, and Process Number

FEP 2.1.03.10.0A

6.2.18.2 Feature, Event, and Process Description

Pits, cracks, and holes in waste packages could be partially or fully plugged by chemical or physical reactions during or after their formation, affecting corrosion processes and water flow and radionuclide transport through the breached waste package. Passivation by corrosion products is a potential mechanism for waste package healing.

6.2.18.3 Descriptor Phrases

Waste package crack plugging (precipitates, corrosion products)

Passivation by corrosion products

6.2.18.4 Screening Decision

Excluded – Low Consequence

6.2.18.5 Screening Argument

Plugging (or healing) of cracks and corrosion holes or pits in waste package by corrosion products and mineral precipitates is a possible process in the repository. However, the effect of plugging could only be to retard the release rate of radionuclides from breached waste packages through retardation of water flow through openings. On this basis, it is conservative to assume that cracks and corrosion holes or pits in the waste package do not heal. Because of this, potential performance credit from the plugging (or healing) of the corrosion penetration openings is not taken into account in TSPA analysis. This treatment of healing of waste packages applies to both CSNF and Co-disposed waste packages. This FEP is excluded on the basis of low consequence because it has no adverse effects on performance.

6.2.18.6 Total System Performance Assessment Disposition

N/A

6.2.18.7 Supporting Analysis Reports

N/A

6.2.19 Healing of Drip Shields

6.2.19.1 Feature, Event, and Process Number

FEP 2.1.03.10.0B

6.2.19.2 Feature, Event, and Process Description

Pits, cracks, and holes in drip shields could be partially or fully plugged by chemical or physical reactions during or after their formation, affecting corrosion processes and water flow through the drip shield.

6.2.19.3 Descriptor Phrases

Drip shield crack plugging (precipitates, corrosion products)

6.2.19.4 Screening Decision

Included

6.2.19.5 Screening Argument

N/A

6.2.19.6 Total System Performance Assessment Disposition

Plugging (or healing) of cracks, holes or pits in drip shields by corrosion products and mineral precipitates could occur in the repository. 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, pits (assumed to be equivalent to holes) will not form on the drip shield material. As discussed in FEP 2.1.03.02.0A, Stress Corrosion Cracking (SCC) of Drip Shields, plugging of any cracks developed by SCC processes is expected. Drip shields are subject to SCC under the action of seismic-induced loading and rockfalls. In the nominal case (in the absence of seismic-induced loading and rockfalls) even if SCC of the drip shield were to occur, cracks in passive alloys, such as Titanium Grade 7, tend to be tight (i.e., small crack opening displacement) (BSC 2003 [DIRS 161234], Section 6.3.7; BSC 2003 [DIRS 161317], Section 6.3.7). As the crack grows through-wall, the tensile stresses normal to the crack walls are relieved, and the resulting crack faces continue to corrode at very low passive corrosion rates until the gap region of the tight crack opening is "plugged" by corrosion products and precipitates such as carbonate minerals. As discussed in *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* model report (BSC 2003 [DIRS 161234], Section 6.3.7), SCC cracks are sealed in a few hundred years at most when water is allowed to flow through the cracks at the expected low flow rate. When the cracks are bridged by water, the sealing process may take thousands of years, but no flow occurs since the water is held by capillary forces. Following plugging of the crack, any solution flow through the crack would be dominated by an efficiency factor determined by the ratio of solution run-off on the drip shield surface compared to through crack flow which in turn is determined by scale porosity/permeability. Because of the expected high density of the calcite deposits and lack of pressure gradient to drive water through the crack, the probability of solution flow through the crack would approach zero (BSC 2003 [DIRS 161234], Section 6.3.7). 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. The development of pits or holes in the drip shield from localized corrosion is not expected.

Since the formation of corrosion products and precipitates precludes water flow through the drip shield, performance credit is taken for the ability of the drip shield to prevent water flow and protect the waste package. Healing of drip shields is included in TSPA as part of waste package degradation analyses in that stress corrosion crack openings are not considered to compromise the water diversion design function of the drip shield.

6.2.19.7 Supporting Analysis Reports

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2003 [DIRS 161234]) (This AR did not explicitly list this FEP as an Included FEP (BSC 2003 [DIRS 161234], Section 1.1). However, this analysis report does provide relevant supporting information describing the implementation of this FEP in TSPA LA).

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.20 Physical Form of Waste Package and Drip Shield

6.2.20.1 Feature, Event, and Process Number

FEP 2.1.03.11.0A

6.2.20.2 Feature, Event, and Process 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 container and affect heat dissipation through container 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.

6.2.20.3 Descriptor Phrases

Effects of commercial spent nuclear fuel waste package form on heat distribution

Effects of co-disposed waste package form on heat distribution

Effects of degraded waste package on flow

Effects of degraded waste package on transport

Effects of degraded drip shield on flow

6.2.20.4 Screening Decision

Included

6.2.20.5 Screening Argument

N/A

6.2.20.6 Total System Performance Analysis Disposition

The waste package, drip shield, and repository design are standardized for the YMP (BSC 2004 [DIRS 167040]). While there is more than one waste package configuration expected to 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. Effects of different waste forms (CSNF, DSNF, and DHLW) on heat dissipation and physical and chemical conditions in the vicinity of the waste packages are indirectly included in the TSPA analysis through different thermal-hydrologic-geochemical responses and their impacts on corrosion processes. Waste package and drip shield degradation modes are modeled in the *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2003 [DIRS 161235]), the *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2003 [DIRS 161236]), and the *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (BSC 2003 [DIRS 161317]) model reports.

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 (refer to FEP 2.1.08.07.0A, Unsaturated Flow in the EBS) (BSC 2004 [DIRS 167253]). 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 the *Engineered Barrier System Features, Events and Processes* report (BSC 2004 [DIRS 167253]).

6.2.20.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235]) (This analysis report did not explicitly list this FEP as an Included FEP (BSC 2003 [DIRS 161235], Table 6-1). However, this AR does provide relevant supporting information describing the implementation of this FEP in TSPA-LA).

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236]) (This AR did not explicitly list this FEP as an Included FEP (BSC 2003 [DIRS 161236], Table I-1). However, this analysis report does provide relevant supporting information describing the implementation of this FEP in TSPA-LA).

WAPDEG Analysis of Waste Package and Drip Shield Degradation (BSC 2003 [DIRS 161317])

6.2.21 Oxygen Embrittlement of Drip Shields

6.2.21.1 Feature, Event, and Process Number

FEP 2.1.06.06.0B

6.2.21.2 Feature, Event, and Process Description

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

6.2.21.2 Descriptor Phrases

Oxygen embrittlement (drip shield)

6.2.21.3 Screening Decision

Excluded – Low Probability

6.2.21.4 Screening Argument

Oxygen embrittlement of titanium results from diffusion of interstitial oxygen into the metal at higher temperatures (> 340°C) (ASM International 1987 [DIRS 103753], p. 681). The time to failure depends on the alloy composition, material thickness, and stress state. For the thermal hydrologic time history files used in the TSPA analyses, the drip shield surface temperatures never exceed 157°C , which is less than the threshold temperature for oxygen embrittlement of 340°C (CRWMS M&O 2001 [DIRS 154594], Section 6.3.5). Therefore, oxygen embrittlement of the titanium drip shields is excluded on the basis of low probability of occurrence under the exposure conditions in the repository.

6.2.21.5 Total System Performance Assessment Disposition

N/A

6.2.21.6 Supporting Analysis Reports

N/A

6.2.22 Mechanical Effects at Engineered Barrier System Component Interfaces

6.2.22.1 Feature, Event, and Process Number

FEP 2.1.06.07.0B

6.2.22.2 Feature, Event, and Process 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.

6.2.22.3 Descriptor Phrases

Rockfall - (drip shield contacts waste package)

Drip shield – pallet contact (mechanical effects)

Waste package – pallet contact (mechanical effects)

6.2.21.4 Screening Decision

Excluded - Low Consequence

6.2.22.5 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 167040]). The pallet is also designed to keep the waste package supported in a horizontal position, and away from the invert and ground support under non-seismic scenarios. 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.

There is some potential for the drip shield to contact the waste package due to mechanical damage caused by rockfall. This is, however, excluded as discussed in FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield.

Mechanical loading at the waste package/(degraded) pallet interfaces has been analyzed (BSC 2003 [DIRS 168489]). The contact stresses are shown to be much less (maximum stress intensity ~150 MPa) than the stress threshold for initiation of stress corrosion cracking (~286 MPa; (BSC 2003 [DIRS 161234], Section 8.3). On this basis, no enhanced degradation due to mechanical loading at the waste package/pallet interfaces is expected. One should note that 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 waste package, drip shield and backfill (if used), which are described in the *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2003 [DIRS 161235]) and *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2003 [DIRS 161236]) model reports.

This treatment of mechanical effects at EBS component interfaces applies to both CSNF and Co-disposed waste packages.

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

6.2.22.6 Total System Performance Assessment Disposition

N/A

6.2.22.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236])

6.2.23 Rockfall

6.2.23.1 Feature, Event, and Process Number

FEP 2.1.07.01.0A

6.2.23.2 Feature, Event, and Process 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.

6.2.23.3 Descriptor Phrases

Rockfall (geometry, key blocks, time dependence)

Rockfall (drip shield damage)

Rockfall (drip shield stress corrosion cracking)

Rockfall (waste package damage)

Rockfall (drip shield contacts waste package)

Drip shield crack plugging

6.2.23.4 Screening Decision

Excluded – Low Consequence (drip shield)

Excluded – Low Probability (waste package)

6.2.23.5 Screening Argument

It should be noted that seismic induced rockfalls/drift degradation are not treated within this FEP. A full discussion of seismic effects is contained in FEPs 1.2.03.02.0A, Seismic Ground Motion Damages EBS Components; 1.2.03.02.0B, Seismic Induced Rockfall Damages EBS Components; and 1.2.03.02.0C, Seismic Induced Drift Collapse Damages EBS Components, treated in the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]).

According to the *Drip Shield Structural Response to Rock Fall* calculation (BSC 2003 [DIRS 162598], Section 6), LS-DYNA analysis 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 a 11.5 MT rockfall (see Section 6.2.14) without contacting the waste package. The maximum displacement from the 11.5 MT rockfall event is 254 mm (BSC 2003 [DIRS 168489]) and the minimum gap between the drip shield and waste package outer barrier is 367 mm (refer to

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 the *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material* report (BSC 2003 [DIRS 161234], Section 6.3.7). It is concluded that a crack will take a minimum of 40 years to grow through the 15-mm drip shield wall (BSC 2003 [DIRS 161234], Section 6.3.7). These cracks are extremely tight and with time, become plugged with corrosion products and other mineral precipitates (BSC 2003 [DIRS 161234], 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.

Analyses related to multiple rockfalls are beyond the scope of this document. Bounding effects of multiple rockfalls and drift degradation are addressed as part of the seismic consequences in the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]).

It is feasible that 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, it is reasonable to predict that the drip shield continues to divert water from falling on the waste package until the drip shield fails after 10,000 years. Therefore, increased inflow of water related to rockfall is excluded from the TSPA analysis based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

The Emplacement Drift System design criteria requires that the drip shield protect the waste package from rockfalls during postclosure (BSC 2002 [DIRS 159292], Section 2.1.3.2). Furthermore, as discussed in FEP 2.1.03.07.0B, Mechanical Impact on Drip Shield, the maximum displacement from a 11.5 metric ton rockfall (which produces the maximum vertical displacement in the drip shield components) is 254 mm. This is less than the minimum gap between the drip shield and waste package outer barrier, which was calculated to be about 367 mm. Therefore, 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 of emplacement).

6.2.23.6 Total System Performance Assessment Disposition

N/A

6.2.23.7 Supporting Analysis Reports

Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material (BSC 2003 [DIRS 161234])

6.2.24 Creep of Metallic Materials in the Waste Package

6.2.24.1 Feature, Event, and Process Number

FEP 2.1.07.05.0A

6.2.24.2 Feature, Event, and Process Description:

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

6.2.24.3 Descriptor Phrases

Thermally-induced waste package creep

Waste package creep from deviatoric stress (rockfall, internal gas pressure)

Waste package creep from internal void space

6.2.24.4 Screening Decision

Excluded – Low Probability

6.2.24.5 Screening Argument

The maximum temperature at the waste package surface will be about 190°C (CRWMS M&O 2001 [DIRS 154594], Section 6.3.1). 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 [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 1370°C (1643 K) (Haynes International 1988 [DIRS 101995]) and the maximum surface temperature (about 190°C or 463 K) is only about 28% of the melting temperature. This treatment of creep of metallic materials in the waste package applies to both CSNF and Co-disposed waste packages. Creep of Alloy 22 at such low temperatures is not expected. Therefore, high temperature creep has a low probability of occurrence.

External stress, by rock displacements or ground motion for example, may lead to plastic deformations and mechanical damage of the container and subsequent leakage of radionuclides. The drip shield is designed to be protective of the waste package during rockfall and ground motion events (refer to 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 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.

6.2.24.5 Total System Performance Assessment Disposition

N/A

6.2.24.6 Supporting Analysis Reports

N/A

6.2.25 Creep of Metallic Materials in the Drip Shield

6.2.25.1 Feature, Event, and Process Number

FEP 2.1.07.05.0B

6.2.25.2 Feature, Event, and Process Description

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

6.2.25.3 Descriptor Phrases

Thermally-induced drip shield creep

Drip shield creep from deviatoric stress (rockfall)

6.2.25.4 Screening Decision

Excluded – Low Probability

6.2.25.5 Screening Argument

Based on the current analyses, the maximum surface temperatures at the drip shield will be about 157°C (CRWMS M&O 2001 [DIRS 154594], Section 6.3.5). Literature indicates that between 200 and 315°C (400 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 postclosure (BSC 2002 [DIRS 159292], Section 2.1.3.2). 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 the *Engineered Barrier System Features, Events and Processes* report (BSC 2004 [DIRS 167253]).

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

6.2.25.6 Total System Performance Assessment Disposition

N/A

6.2.25.7 Supporting Analysis Reports

N/A

6.2.26 Volume Increase of Corrosion Products Impacts Waste Package

6.2.26.1 Feature, Event, and Process Number

FEP 2.1.09.03.0B

6.2.26.2 Feature, Event, and Process 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, leading to waste package damage.

6.2.26.2 Descriptor Phrases

Volume increase of corrosion products (waste package damage)

6.2.26.3 Screening Decision

Excluded – Low Consequence

6.2.26.4 Screening Argument

In general, corrosion products have greater volume than the metal. When the corrosion products form in a tightly confined space, the volume increase by the corrosion products 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 due to the pressure exerted by the corrosion product (Cr_2O_3) of the inner vessel (Type 316 stainless steel) will not occur. Analyses cited in *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2003 [DIRS 161235], Section 6.4.2), indicate that for chromia (Cr_2O_3) scale-forming alloys (e.g., Alloy 22 and 316 stainless steel), 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 Alloy 22 waste package. In the current design of waste package and engineered barrier system in the emplacement drift (BSC 2004 [DIRS 167040]), 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 consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

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 166926]).

6.2.26.5 Total System Performance Assessment Disposition

N/A

6.2.26.6 Supporting Analysis Reports

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

6.2.27 Electrochemical Effects in Engineered Barrier System

6.2.27.1 Feature, Event, and Process Number

FEP 2.1.09.09.0A

6.2.27.2 Feature, Event, and Process Description

Electrochemical effects may establish an electric potential within the drift or between materials in the drift and more distant metallic materials in the natural system. 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.

6.2.27.3 Descriptor Phrases

Electrophoresis/electro-osmosis

Galvanic coupling (waste package)

Galvanic coupling (drip shield)

6.2.27.4 Screening Decision

Excluded – Low Consequence

6.2.27.5 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 electrochemical effects 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, which is constructed of Alloy 22 (UNS N06022) and is designed to keep the waste package from contacting other dissimilar metals (BSC 2004 [DIRS 167040]). The pallet is also designed to keep the waste package supported in a horizontal position, and away from the invert and ground support under non-seismic scenarios. 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 current waste package design (BSC 2004 [DIRS 167040]) includes an outer barrier of Alloy 22 (UNS N06022) over a 316 stainless steel inner vessel. In addition, a titanium drip shield is added to this design to provide defense in depth. Although the stainless steel 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 316 stainless steel are very close to each other under typical exposure conditions (e.g., ASM International [DIRS 103753], p. 557) with Alloy 22 (UNS N06022) slightly more noble than 316 stainless steel. After failure of the Alloy 22 (UNS N06022) waste package outer barrier, electrochemical coupling of the Alloy 22 (UNS N06022) waste package outer barrier with the 316 stainless steel waste package inner vessel could occur. Due to the similarity in corrosion potential of Alloy 22 (UNS N06022) and 316 stainless steel, any enhanced degradation of either material would be negligible. Therefore, electrochemical coupling of the Alloy 22 (UNS N06022) waste package outer barrier and the 316 stainless steel waste package inner vessel is of low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

There is some potential for the drip shield to contact the waste package due to mechanical damage caused by rockfall. This is, however, unlikely 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 (e.g., ASM International [DIRS 103753], p. 557), which indicates that there is very low probability for galvanic interaction between the two materials. 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.

6.2.27.6 Total System Performance Assessment Disposition

N/A

6.2.27.7 Supporting Analysis Reports

N/A

6.2.28 Thermal Sensitization of Waste Packages

6.2.28.1 Feature, Event, and Process Number

FEP 2.1.11.06.0A

6.2.28.2 Feature, Event, and Process Description:

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

6.2.28.3 Descriptor Phrases

Aging-phase instability (waste package)

Decreased resistance to waste package corrosion

6.2.28.4 Screening Decision

Excluded – Low Probability

6.2.28.5 Screening Argument

Alloy 22 (UNS N06022) could to be subject to “aging” and phase instability when exposed to elevated temperatures. 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 2003 [DIRS 162199]).

Before waste loading, the waste containers (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 container (Plinski 2001 [DIRS 156800], Section 8.1.8). Analyses presented in the *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* report (BSC 2003 [DIRS 161235], Section 6.4.6) show that the corrosion performance of Alloy 22 (UNS N06022) base metal is not affected by the aging and phase instability as long as the waste package surface temperature is kept below 200°C under the exposure conditions expected in the repository. The maximum temperature at the waste package surface will never exceed 190^{o}C (CRWMS M&O 2001 [DIRS 154594], Section 6.3.1).

The closure lid welds cannot be solution annealed without risking damage to the waste form. Therefore, the closure welds of the waste package out barrier could be more prone to thermal aging and phase instability than the base metal under long-term thermal exposure in the repository (BSC 2003 [DIRS 161235], Section 6.4.6). Analyses conducted in the report entitled *General Corrosion and Localized Corrosion of Waste Package Outer Barrier* (BSC 2003 [DIRS 161235], 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 the test temperatures varying from 45 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 2003 [DIRS 161235], Section 6.4.6) and the corrosion performance of the waste package outer barrier is not expected to be affected by the aging and phase stability in the repository. This treatment of thermal sensitization of waste packages applies to both CSNF and co-disposed waste packages. Thermal sensitization of waste packages is excluded on the basis of low probability of occurrence.

6.2.28.5 Total System Performance Assessment Disposition

N/A

6.2.28.6 Supporting Analysis Reports

Aging and Phase Stability of Waste Package Outer Barrier (BSC 2003 [DIRS 162199])

General Corrosion and Localized Corrosion of Waste Package Outer Barrier (BSC 2003 [DIRS 161235])

Note that the BSC 2003 [DIRS 161235] lists this FEP in a Table entitled "List of Included FEPs Relevant to This Model Report" (BSC 2003 [DIRS 161235], Table 1.1) and also in a Table entitled "Included FEPs for This Model Report and Their Disposition in TSPA-LA" (BSC 2003 [DIRS 161235], Table 6-1). Regardless of the titles of the Tables, the associated text in fact supports the exclusion of the FEP on the basis of low probability of occurrence (BSC 2003 [DIRS 161235], Table 6-1).

6.2.29 Thermal Sensitization of Drip Shields

6.2.29.1 Feature, Event, and Process Number

FEP 2.1.11.06.0B

6.2.29.2 Feature, Event, and Process Description

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

6.2.29.3 Descriptor Phrases

Aging-phase instability (drip shield)

Decreased resistance to drip shield corrosion

6.2.29.4 Screening Decision

Excluded – Low Probability

6.2.29.5 Screening Argument

Aging and Phase stability of the drip shield is considered in Section 6.5.3 of the report entitled *General Corrosion and Localized Corrosion of the Drip Shield* (BSC 2003 [DIRS 161236]). In the report, it is observed that Titanium Grade 7 is a stabilized alpha (α) phase alloy and possesses outstanding phase stability. While Titanium Grade 7 does contain small amounts of alloying elements (DTN: MO0003RIB00073.000 [DIRS 152926]), most notably Pd, it is essentially a pure titanium alloy which has no capability of forming intermetallic compounds under the thermal exposure conditions in the repository.

The solubility of Pd in Titanium Grade 7 is about 1 weight percent at 400°C. The nominal concentration of Pd in Titanium Grade 7 is well below the solubility limit at this temperature (Gdowski 1997 [DIRS 102789], pp. 1-8). Titanium-palladium intermetallic compounds capable of being formed in this 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, 70, 80, 90, 100 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 (HAZ) and between the HAZ 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.

6.2.29.6 Total System Performance Assessment Disposition

N/A

6.2.29.6 Supporting Analysis Reports

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236])

6.2.30 Thermal Expansion/Stress of In-Drift Engineered Barrier System Components

6.2.30.1 Feature, Event, and Process Description

FEP 2.1.11.07.0A

6.2.30.2 Feature, Event, and Process 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.

6.2.30.3 Descriptor Phrases

Thermal-mechanical effects on waste packages

Thermal expansion of drip shield

6.2.30.4 Screening Decision

Excluded - Low Consequence

6.2.30.5 Screening Argument

The coefficient of thermal expansion for Type 316L stainless steel (an analogue for the 316 stainless steel used for the waste package inner vessel) 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 the calculation entitled *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 barrier gap size to be at least 1 mm (BSC 2004 [DIRS 166694]).

The *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 (316 stainless steel) to accommodate the differing thermal expansion coefficients. Typical waste package designs also require large longitudinal barrier gaps (~30 mm) (BSC 2001 [DIRS 157816], page 3). Therefore, although thermal expansion of waste package components does occur, 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 current drip shield design (BSC 2003 [DIRS 161519]), the drip shield connectors are designed in such a way that allows 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 (367 mm) (refer to Equation 7 in Section 6.2.14.5). The space needed 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.

6.2.30.6 Total System Performance Assessment Disposition

N/A

6.2.30.7 Supporting Analysis Reports

N/A

6.2.31 Gas Generation (H_2) from Waste Package Corrosion

6.2.31.1 Feature, Event, and Process Number

FEP 2.1.12.03.0A

6.2.31.2 Feature, Event, and Process 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 containers, 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.

6.2.31.3 Descriptor Phrases

Internal gas pressure from H_2 (waste package damage)

Hydride cracking (drip shield)

Hydride cracking (waste package)

Repository pressurization

Chemical effects from H_2 generation

6.2.31.4 Screening Decision

Excluded – Low Consequence

6.2.31.5 Screening Argument

The materials selected for waste package outer barrier and the drip shield are highly corrosion resistant materials. 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 Yucca Mountain 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 (HIC). HIC results 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. Hydrogen generation and absorption in the drip shield due to corrosion and galvanic interaction with ground support materials has been addressed in FEP 2.1.03.04.0B, Hydride Cracking of Drip Shields. This treatment of gas generation from waste package corrosion applies to both CSNF and co-disposed 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.

6.2.31.6 Total System Performance Assessment Disposition

N/A

6.2.31.7 Supporting Analysis Reports

N/A

6.2.32 Radiolysis

6.2.32.1 Feature, Event, and Process Number

FEP 2.1.13.01.0A

6.2.32.2 Feature, Event, and Process 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).

6.2.32.3 Descriptor Phrases

Gas generation from radiolysis

Chemical effects from radiolysis in-drift

6.2.32.4 Screening Decision

Excluded – Low Consequence

6.2.32.5 Screening Argument

This section addresses only the effects of radiolysis on the waste package outer barrier. Radiolysis effects on waste form are addressed in *Miscellaneous Waste-Form FEPs* (BSC 2004 [DIRS 167252]). Chemical effects from radiolysis (cement degradation) are addressed in the *Engineered Barrier System Features, Events, and Processes* report (BSC 2004 [DIRS 167253]). Gamma radiation is the dominant contributor to dose rate at the waste package surface (BSC 2003 [DIRS 165269], Tables 59 and 60) because alpha and beta radiation will not penetrate the container and the neutron dose rate is relatively small. The effects of radiation on waste package materials corrosion 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 that may contain 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. These shifts in potential have been shown to be due to the formation of hydrogen peroxide (BSC 2003 [DIRS 161235], Section 6.5.1).

Calculations of the expected radiation levels at the surface of the waste package have been performed (BSC 2003 [DIRS 165269], Table 60). For a bounding case waste package containing 21 PWR spent fuel assemblies (75 GWd/MTU burnup, and 5 year decay), the maximum surface radiation level was calculated to be about 1100 rad/hr at the outer surface of the waste package barrier and 1550 rad/hr at the bottom lid of the barrier (BSC 2003 [DIRS 165269], Table 60). During the ventilation period of 50 years, no aqueous or humid air environment, and therefore no radiolysis, is expected. After 50 years, the maximum surface radiation level decreases to less than 100 rad/hr for the outer surface and 70 rad/hr for the bottom lid (BSC 2003 [DIRS 165269], Table 60). These values correspond to reduction factors of 0.09 for the outer surface and 0.045 for the bottom lid (BSC 2003 [DIRS 165269], Table 60). One hundred years after emplacement, the maximum calculated levels reduce to about 30 rad/hr for the outer surface and 20 rad/hr for the bottom lid region (BSC 2003 [DIRS 165269], Table 60). These values correspond to reduction factors of 0.03 for the outer surface and 0.015 for the bottom lid (BSC 2003 [DIRS 165269], Table 60). It is to be noted that there are bounding radiation levels for the highest burnup spent fuel and these are well below the levels at which some 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 that 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 [DIRS 112178], p. 30). Based on this data it is concluded that, even in aggressive $MgCl_2$ brines, the 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, the effects of radiolysis are excluded based on low consequence to radiological exposures to the RMEI and radionuclide releases to the accessible environment.

6.2.32.6 Total System Performance Assessment Disposition

N/A

6.2.32.7 Supporting Analysis Reports

General Corrosion and Localized Corrosion of the Drip Shield (BSC 2003 [DIRS 161236])

6.2.33 Radiation Damage in Engineered Barrier System

6.2.33.1 Feature, Event, and Process Number

FEP 2.1.13.02.0A

6.2.33.2 Feature, Event, and Process Description

Radiolysis due to the alpha, beta, gamma-ray, and neutron irradiation of water could result in the enhancement for the movement of the radionuclides from the surface of a degraded waste form into groundwater flow. When radionuclides decay, the emitted high-energy particle could result in the production of radicals in the water or air surrounding the SNF. 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 meat) into the groundwater flow. Strong radiation fields could lead to radiation damage to the waste forms and containers (CSNF, DSNF, DHLW), drip shield, seals and surrounding rock.

6.2.33.3 Descriptor Phrases

Radiation damage in-drift (waste package)

Radiation damage in-drift (drip shield, pallet, invert, seals, EDZ)

6.2.33.4 Screening Decision

Excluded – Low Consequence

6.2.33.5 Screening Argument

The enhanced degradation/corrosion of the fuel is covered in the *Miscellaneous Waste-Form FEPs* (BSC 2004 [DIRS 167252]). The Waste Package Materials Performance Peer Review Panel addressed the possibility of radiation damage in the repository in their Final Report (Beavers, et al. [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. These fluxes could cause the following types of damage: 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. [DIRS 158781], Section 3.10). The peak neutron flux has been calculated to be about $5x10^4$ n/cm^2/sec) in the repository environment (Beavers et al. [DIRS 158781], Section 3.10). The total neutron fluence, taking the most conservative estimate with no nuclear decay of the waste, will be about $1.5x10^{16}$ n/cm^2 in 10,000 years in the repository environment (Beavers et al. [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. [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], p. 971-974), 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.

6.2.33.6 Total System Performance Assessment Disposition

N/A

6.2.33.7 Supporting Analysis Reports

N/A

7. CONCLUSIONS

The analyses documented in this scientific analysis are for the current LA design where a drip shield is placed over the waste package (BSC 2004 [DIRS 167040]). Repository designs that are lower in temperature than the current LA design will not affect the screening decision of any of the waste package FEPs. Most of the waste package degradation mechanisms have no temperature dependence and those that do, have a positive correlation with rising temperatures and increased degradation rates (see screening arguments and TSPA Disposition discussions listed in this document). This is true for designs that only affect the repository temperature and not any other aspects of the design, such as waste package spacing in the repository, drift orientation, etc.

Thirty-three FEPs relevant to waste package and drip shield degradation processes were screened against criteria presented in Section 4.2 of this report. The results of this screening process are documented in the screening arguments and the TSPA disposition statements in Section 6.2 of this report. The screening basis for each FEP is summarized below in Table 7-1. This table shows the FEP number, FEP name, screening decision (included/excluded in TSPA) 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	Screening Basis
1.1.03.01.0A	Error in Waste Emplacement	Excluded	Low consequence
2.1.03.01.0A	General Corrosion of WPs	Included	N/A
2.1.03.01.0B	General Corrosion of DSs	Included	N/A
2.1.03.02.0A	SCC of W Ps	Included	N/A
2.1.03.02.0B	SCC of DSs	Excluded	Low consequence
2.1.03.03.0A	Localized Corrosion of WPs	Included	N/A
2.1.03.03.0B	Localized Corrosion of DSs	Excluded	Low consequence
2.1.03.04.0A	Hydride Cracking of WPs	Excluded	Low consequence
2.1.03.04.0B	Hydride Cracking of Drip Shields	Excluded	Low consequence
2.1.03.05.0A	MIC of WPs	Included	N/A
2.1.03.05.0B	MIC of DSs	Excluded	Low consequence
2.1.03.06.0A	Internal Corrosion of WPs Prior To Breach	Excluded	Low consequence
2.1.03.07.0A	Mechanical Impact on WP	Excluded	Low consequence / Low probability
2.1.03.07.0B	Mechanical Impact on DS	Excluded	Low consequence
2.1.03.08.0A	Early Failure of WPs	Included	N/A
2.1.03.08.0B	Early Failure of DSs	Excluded	Low consequence
2.1.03.09.0A	Copper Corrosion in EBS	Excluded	Low consequence
2.1.03.10.0A	Healing of WPs	Excluded	Low consequence
2.1.03.10.0B	Healing of DSs	Included	N/A
2.1.03.11.0A	Physical Form of WP and DS	Included	N/A
2.1.06.06.0B	Oxygen Embrittlement of DSs	Excluded	Low probability
2.1.06.07.0B	Mechanical Effects at EBS Component Interfaces	Excluded	Low consequence

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

FEP Number	FEP Name	Screening Decision	Screening Basis
2.1.07.01.0A	Rockfall	Excluded	Low consequence / Low probability
2.1.07.05.0A	Creep of Metallic Materials in the WP	Excluded	Low probability
2.1.07.05.0B	Creep of Metallic Materials in the DS	Excluded	Low probability
2.1.09.03.0B	Volume Increase of Corrosion Products Impacts WP	Excluded	Low consequence
2.1.09.09.0A	Electrochemical Effects in EBS	Excluded	Low consequence
2.1.11.06.0A	Thermal Sensitization of WPs	Excluded	Low probability
2.1.11.06.0B	Thermal Sensitization of DSs	Excluded	Low probability
2.1.11.07.0A	Thermal Expansion/Stress of In-Drift EBS Components	Excluded	Low consequence
2.1.12.03.0A	Gas Generation (H_2) From WP Corrosion	Excluded	Low consequence
2.1.13.01.0A	Radiolysis	Excluded	Low consequence
2.1.13.02.0A	Radiation Damage in EBS	Excluded	Low consequence

DS = drip shield, WP = waste package, SCC = stress corrosion cracking, MIC = microbially influences corrosion

It is to be noted that the screening decisions documented herein do not consider the consequences of combinations of FEPs. This is beyond the scope of this document at this time. If a decision is made to address combined FEPs, they will be considered on a case-by-case basis and any impact on screening decisions will be appropriately documented.

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