MOL.20010216.0006

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT ANALYSIS/MODEL COVER SHEET

1. QA: QA

Page: 1 of: 218

Complete Only Applicable Items

2. ☑ Analysis Check all that apply

Type of Analysis
☐ Engineering
☑ Performance Assessment
☐ Scientific

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

Describe use:

Select FEPS important for TSPA modeling of waste-form degradation

3. ☐ Model Check all that apply

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

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

Describe use:

4. Title:

Miscellaneous Waste-Form FEPs

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

ANL-WIS-MD-000009 REV 00 ICN 01

6. Total Attachments:

Five (5)

7. Attachment Numbers - No. of Pages in Each:

Attachment I – 4 pages
Attachment II – 4 pages
Attachment III – 2 pages
Attachment IV – 12 pages
Attachment V – 39 pages

	Printed Name	Signature	Date
8. Originator	Albert R. Schenker	Albert R Schenker	12/08/00
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12. Remarks:

On January 26, 2000 a design change was initiated to resolve certain thermal design issues. This design change will result in a greater ability of the waste packages to eject heat after closure of the potential repository, thereby maintaining the two thermal requirements. The first requirement is protective of the fuel cladding, and the second requires that sections of the rock pillars between drifts remain below the boiling temperature of water, providing a path for water drainage. This design change is described in Technical Change Request T2000-0133, dated January 26, 2000 (CRWMS M&O 2000ai). This AMR originally considered a design with backfill and is now being updated to address the design without backfill.

As a general comment, the primary effect of the elimination of backfill on the waste form is the decrease of peak temperatures inside the waste package, which is beneficial. For example, the temperature decrease (1) reduces the chance for creep rupture and stress corrosion cracking of CSNF cladding; (2) reduces the degradation rates of the CSNF and HLW matrices; (3) improves the applicability of the current data for in-package chemistry; and (4) decreases the solubility of uranium. However, elimination of backfill design change does not affect the Waste Form Degradation Model because temperature is explicitly included as a model variable. That is, the model is an explicit function of the surface temperature of the waste package; thus, any changes in the surface temperature of the waste package because of the design change are automatically included.

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ANALYSIS/MODEL REVISION RECORD
Complete Only Applicable Items

1. Page: 2 of: 218

2. Analysis or Model Title:

Miscellaneous Waste-Form FEPs

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

ANL-WIS-MD-000009 REV 00 ICN 01

4. Revision/Change No.	5. Description of Revision/Change
REV 00 REV 00 ICN 01	Initial Issue ICN changes: Ensured consistency with FEPs summary in Waste Form PMR. Made some editorial corrections and updated procedure references. Transferred the following three FEP discussions from this AMR (ANL-WIS-MD-000009) to the Clad Degradation FEPs AMR (ANL-WIS-MD-000008 REV 00 ICN 01): Volume Increase of Corrosion Products - YMP No. 2.1.09.03.00 Gap and Grain release of Cs, I - YMP No. 2.1.02.07.00 Rockfall (Large Block)- YMP No. 2.1.07.01.00 Transferred FEP discussion for mutations, YMP No. 2.1.13.03.00, from this AMR (ANL-WIS-MD-000009) to the Waste Form Colloid FEPs AMR (ANL-WIS-MD-000012). Added text to address legal team direction on regulatory wording. Added text to address no backfill design change. Assesses that there is no impact. Added new FEP 2.1.02.28.00 on various features of the 250 DSNF type and arranged by waste groups. Modified screening decisions based on legal team review. Added Attachment V to list all secondary FEPs associated with primary FEPs covered in this AMR.

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ACRONYMS AND ABBREVIATIONS

ACC	accession number of an internal document produced by CRWMS M&O
AMR	Analysis/Model Report
ANL	Argonne National Laboratory
ATM	approved testing material
BWR	boiling-water reactor
CFR	Code of Federal Regulations
CLST	Container Life and Source Term
CRWMS	Civilian Radioactive Waste Management System
CSNF	commercial spent nuclear fuel
DCCG	diffusion controlled cavity growth
DOE	U.S. Department of Energy
DSNF	DOE-owned spent nuclear fuel
EBS	engineered barrier system
EDTA	ethylenediamine tetraacetate
Eh	redox potential
ENFE	Evolution of Near-Field Environment
EPA	U.S. Environmental Protection Agency
FEP	features, events, and processes
HIC	high-integrity can
HLW	high-level waste
INEEL	Idaho National Engineering and Environmental Laboratory
IRSR	Issue Resolution Status Report
IT	isomeric transition
KTI	Key Technical Issue
LA	license application
LET	linear energy transfer
M&O	Management and Operating Contractor
MCO	multi-cask overpack
MIC	microbially influenced corrosion
MOL	Reference Information System Document Number Prefix
MOX	mixed oxide (fuel)
MTHM	metric tons heavy metal
NEA	Nuclear Energy Agency
NFE	near-field environment

ACRONYMS AND ABBREVIATIONS (Continued)

NRC	U.S. Nuclear Regulatory Commission
NSNFP	National Spent Nuclear Fuel Program
NWPA	Nuclear Waste Policy Act
OCRWM	Office of Civilian Radioactive Waste Management
PA	performance assessment
PE	pyrophoric event
pH	negative logarithm to the base 10 of the effective hydrogen-ion concentration
PMR	Process Model Report
PNNL	Pacific Northwest National Laboratory
PWR	pressurized-water reactor
QARD	Quality Assurance Requirements and Description
RIP	Repository Integration Program (model)
RMEI	reasonably maximally exposed individual
RN	radionuclide(s)
SCC	stress corrosion cracking
SEM	scanning electron microscope
SF	spontaneous fission
SNF	spent nuclear fuel
SR	site recommendation
SZ	saturated zone
TIC	Technical Information Center
TRIGA	research fuel reactor (U-ZrH$_x$)
TSPA	Total System Performance Assessment
TSPAI	Total System Performance Assessment and Integration
UZ	unsaturated zone
VA	viability assessment
WAPDEG	waste package degradation
WF	waste form
WIPP	Waste Isolation Pilot Plant
WP	waste package
WPE	waste-package equivalent
YM	Yucca Mountain
YMP	Yucca Mountain Project

1. PURPOSE

The U.S. DOE must provide a reasonable assurance that the performance objectives for the Yucca Mountain Project (YMP) potential radioactive-waste repository can be achieved for a 10,000-year post-closure period. The guidance that mandates this direction is under the provisions of 10 CFR Part 63 (Nuclear Regulatory commission proposed regulations governing Yucca Mountain) (64 FR 8640) and the U.S. Department of Energy's (DOE's) "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada" (Dyer 1999 and herein referred to as DOE's Interim Guidance). This assurance must be demonstrated in the form of a performance assessment that:

1. Identifies the features, events, and processes (FEPs) that might affect the performance of the potential geologic repository
2. Examines the effects of such FEPs on the performance of the potential geologic repository
3. Estimates the expected annual dose to a specified receptor group
4. Provides the technical basis for inclusion or exclusion of specific FEPs.

To implement these requirements, the DOE has adopted the scenario-development methodology that Cranwell et al. (1990) developed for the U.S. Nuclear Regulatory Commission (NRC) with one significant change. Yucca Mountain TSPA has expanded the definition of scenario so that it is not limited to a single, deterministic future of the system, but rather as a set of similar futures that share common FEPs (Swift et al. 1999). Focusing only on waste-form (WF) FEPs, this Analysis/Model Report (AMR) considers the first two steps of the scenario-development methodology.

To fulfill its oversight role for the Yucca Mountain Project (YMP), the staff of the NRC has developed a process for early resolution of technical issues. Here, the NRC staff releases Issue Resolution Status Reports (IRSRs) for the nine Key Technical Issues (KTIs) important to postclosure performance. Three such issues, identified as Total System Performance Assessment and Integration (TSPAI), Container Life and Source Term (CLST), and Evolution of Near-Field Environment (ENFE) relate to WF FEPs.

This AMR has a three-fold purpose:

1. As part of the FEP-identification step, it summarizes the screening decisions for 86 WF FEPs and relates them to the AMRs in which they are documented.
2. It shows the correspondence between WF FEPs and the sub-issues and acceptance criteria of three KTIs.
3. It documents the screening discussions and/or TSPA dispositions for the 54 miscellaneous WF FEPs of this AMR.

1.1 SCOPE

This AMR has been prepared to satisfy the FEP screening documentation requirements in the Technical Work Plan entitled *Technical Work Plan for Waste Form Degradation Process Model Report for SR,* TWP-EBS-MD-000006 (CRWMS M&O 2000b).

The FEPs list (CRWMS M&O 2000al) consists of 1797 entries classified as heading entries or primary and secondary FEPs. The primary and secondary FEPs have been assigned to various Process Model Reports (PMRs). The FEP assignments were based on the nature of the FEPs so that the analysis and disposition for each FEP reside with the subject-matter experts in the relevant disciplines. The disposition of FEPs other than Waste-Form FEPs is documented in AMRs prepared by the responsible PMR groups.

This AMR addresses the 54 FEPs that have been identified by the Waste-Form Group as Miscellaneous Waste-Form FEPs and assigned to the Miscellaneous Waste-Form FEP report (this document).

Additionally this ICN evaluated the FEPs discussed in Revision 00 with respect to any impact from the design change to a "no backfill option" (CRWMS M&O 2000ai). The design of the potential repository continues to evolve in preparation for the future license application. In general, these design changes can directly influence the screening arguments for FEPs and their inclusion in or exclusion from the Waste Form Degradation Model. The elimination of backfill is a performance-affecting design change that was made after the development of the current Waste Form Degradation Model. The primary effect of the elimination of backfill is the decrease of peak temperatures inside the waste package, which is beneficial. For example, the temperature decrease (1) reduces the chance for creep rupture and stress corrosion cracking of CSNF cladding, (2) reduces the degradation rates of the CSNF and HLW matrices, (3) improves the applicability of the current temperature data for in-package chemistry, and (4) decreases the solubility of uranium. However, the design change to eliminate backfill does not affect the Waste Form Degradation Model because temperature is explicitly included as a model variable. That is, the model incorporates an explicit function of the surface temperature of the waste, so any changes in the surface temperature of the waste package because of the design change are automatically included.

1.2 FEPs IDENTIFICATION AND ANALYSIS

To demonstrate that regulatory-specified performance objectives of proposed 10 CFR Part 63 (64 FR 8640) and the DOE's interim guidance (Dyer 1999) can be achieved for a 10,000-year post-closure period, the Yucca Mountain Project is implementing a stochastic scenario-development methodology based on the work of Cranwell et al. (1990). The methodology provides a systematic approach for considering, as completely as practicable, the possible future states of a potential repository system. It seeks to span the set of all possible future states using a finite set of scenarios. Here, each scenario represents the ensemble of possible futures corresponding to parameter and model uncertainties present in the group of FEPs comprising the scenario. The methodology begins with a comprehensive FEP identification step followed by a rigorous FEP screening step. With its focus on waste form FEPs, this AMR considers these first two steps of scenario development.

1.2.1 PA and FEPs Relationship

Generally, the process of assessing whether a radioactive-waste disposal system meets a set of performance criteria is a performance assessment (PA). The NRC specifically states (NRC 1998c, 10 CFR Section 63.2) that this process of "performance assessment means a probabilistic analysis that includes (1) identification of features, events, and processes that might affect the performance of the potential geologic repository; (2) examines the effects of such features, events, and processes on the performance of the geologic repository; and (3) estimates the expected annual dose to the average member of the critical group as a result of releases from the geologic repository." The EPA standard (40 CFR 191) has similar wording but the performance measure was cumulative releases of radioisotopes and associated uncertainties expressed as a complementary cumulative probability function. Thus, features, events, and processes (FEPs) are a fundamental aspect of a PA, where a feature is some aspect of the disposal system, an event is a phenomenon that occurs on a short time frame relative to the life of the disposal system, and a process is phenomenon relevant to the functioning of the disposal system that occurs on a long time frame.

As described below, the identification of FEPs that are potentially relevant to the functioning of the disposal system conceptually produces the initial domain or parameter space of the model of the disposal system. The screening process omits those portions of the domain that are not pertinent. The formal and defensible selection of the pertinent domain of FEPs when developing the conceptual model of the disposal system is one aspect that sets PA apart from typical scientific or engineering analysis. The criteria employed to omit FEPs are based on either regulatory guidance for the modeling style (e.g., regulatory period, omission of purposeful intrusion, or omission of inadvertent intrusion other than exploratory drilling) or the prediction of negligible influence on the performance measure (here the probabilistic estimate of the annual dose). Because of the nature of FEP screening and model development, several iterations of the PA process are potentially necessary to eliminate those FEPs of negligible influence and improve the modeling of those FEPs retained.

1.2.2 Regulatory Periods

Similar to 40 CFR 191 (1993, Federal Register Vol. 58, no. 27), the NRC proposed period for conducting the PA in 10 CFR Part 63 (64 FR 8640) is 10^4 years because this period (Dyer 1999) (1) is consistent with other regulations (e.g., 40 CFR 191 1993 and guidance on no-migration petitions for hazardous, non-radioactive materials), (2) is sufficiently long to challenge the natural engineered barriers and natural barriers (especially when including human intrusion), and (3) includes the period when the waste is most hazardous.

Use of a waste package that lasts beyond this 10^4-yr regulatory period does not necessarily justify the elimination of most FEPs except those that potentially influence the waste-package life for several reasons. First, the NRC regulation also calls for an environmental impact statement (EIS) (NRC 1998c, 10 CFR 63.21) to comply with the Nuclear Waste Policy Act (NWPA) (NWPA 1982; 1987). This EIS modeling would require the inclusion of FEPs beyond those influencing the life of the waste package since the modeling may need to evaluate behavior up to 10^6 yr. Furthermore, the NRC regulations require multiple barriers (NRC 1998c, 10 CFR 63.111(h); 63.113(a)), and a simulation of the disposal system after failure of the waste package provides some evaluation of the resiliency of other barriers in the disposal system. Specifically, in the preamble to the proposed regulation, NRC states, "Irrespective of the projected lifetime of the waste-package design, the capability of the natural barriers to limit exposures would need to be evaluated in the context of the multiple barrier requirement."

Figure 1 provides the development and screening process for FEPs related to waste-form degradation. Although shown as sequential steps, iteration occurs as new FEPs are identified. The methodology begins with a comprehensive FEP identification step followed by a rigorous FEP screening step. The development of the YMP FEP list is given in the YMP FEPs Database (CRWMS M&O 2000al).

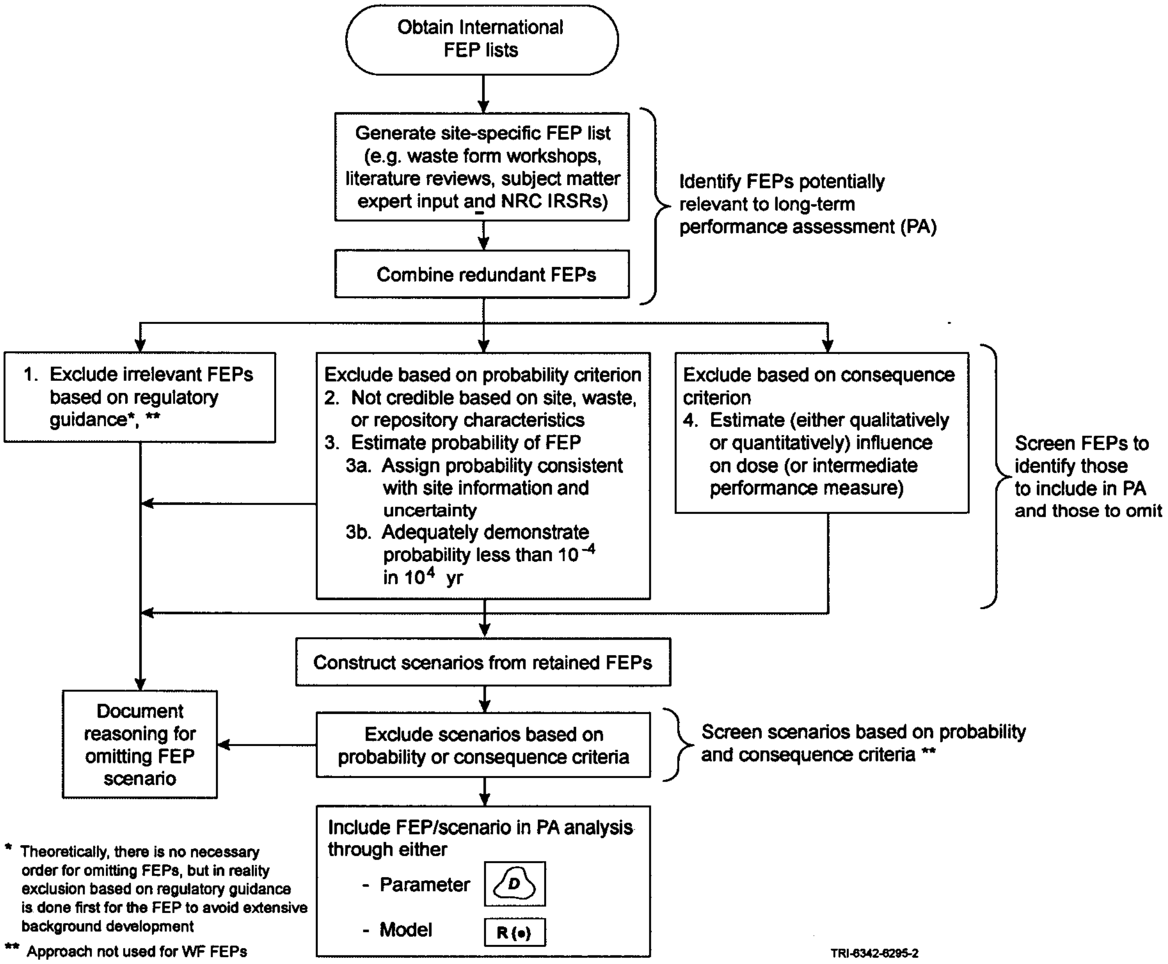


Figure 1. Development and Screening Process for FEPs Related to Waste-form Degradation

NOTE: Procedure adopted by NRC to select FEPs (i.e., the model domain of the analysis) for either inclusion or exclusion in the assessment of the performance of the radioactive waste disposal system (NRC 2000, pp. 22-23) (Swift et al. 1999, Figure 2). Although shown as sequential steps, iteration occurs as new FEPs are identified.

1.2.3 FEP Identification

The first step of the scenario-development methodology is the identification of FEPs potentially relevant to the performance of the Yucca Mountain potential repository. The initial set of FEPs was created for the Yucca Mountain TSPA by Swift et al. (1999). A universal FEP list was generated from a combined lists of FEPs previously identified as relevant to the YMP (e.g., lists by Wilson et al. 1994 and by CRWMS M&O 1995) with a draft FEP list compiled by a Nuclear Energy Agency (NEA) working group. The NEA list is the most comprehensive list available internationally, and it currently contains 1261 entries from Canadian, Swiss, and Swedish spent-fuel programs, from intermediate and low-level waste programs of the United Kingdom, and from the United States' Waste Isolation Pilot Plant (WIPP) program. The NEA grouped these 1261 FEPs into a hierarchical structure consisting of 150 layers, categories, and headings that were also entered into the database. In addition, one more heading was added (far-field criticality), thus, the total number of NEA entries is 1412. An additional 292 FEPs have been identified from YMP literature and site studies, and 82 FEPs have been identified during YMP staff workshops. Three primary sources were used to identify waste-form FEPs. An additional 11 non-waste-form FEPs were identified by the NRC and SME review during preparation of CRWMS M&O (2000al) for a total of 1797 FEPs.

- During the initial performance assessment of the disposal of SNF and HLW stored at the Idaho National Engineering Laboratory (Rechard 1993)
- DSNF/HLW Workshop held January 19, 1999 (Eide 2000)
- Waste-form workshop held February 2-4,1999; results documented in this AMR (CRWMS M&O 1999o).

Consistent with the diverse backgrounds of the programs contributing, FEPs have been identified by a variety of methods including expert judgment, informal elicitation, event-tree analysis, stakeholder review, and regulatory stipulation. Table 1 displays the results of the FEPs identification process developed from the core Nuclear Energy Agency (NEA) FEPs (CRWMS M&O 2000al), the addition of YMP specific FEPs developed during the earlier period of the YM project, those developed as a result of the waste-form workshops, analysis supporting the development of the YM performance assessment models, and the AMR process.

Table 1. Results of the FEP Selection Process for Waste Form

NEA General List		YMP Extension	
Started with 81 Primary FEPs with remote association to waste form (281 NEA primary and secondary FEPs)		**Identified During Early YMP/YSCP Process**	**Identified during AMR Preparation Includes WF Workshops**
Inventory	5	0	0
In- Package Chemistry	20	3	2
CSNF	6	1	0
Cladding	3	1	16
DSNF	2	0	4
HLW	4	0	0
Solubility	2	1	1
Colloids	8	5	3
Total	50	11	26

All potentially relevant FEPs have been included, regardless of origin, giving a FEP list consisting of 1797 entries. This approach has led to considerable redundancy, because the same FEPs are frequently identified by multiple sources, but it also ensures that a comprehensive review of narrowly defined FEPs will be performed. The FEPs list is considered open and will continue to grow as additional FEPs are identified.

Each FEP has been identified as either type of heading entry or a primary or a secondary FEP. Primary FEPs are those FEPs for which the project proposes to develop detailed screening arguments. The classification and description of primary FEPs strives to capture the technical essence of all the secondary FEPs that are subsidiary to the primary FEPs. For example, the primary FEP, Meteorite Impact, can be used appropriately to resolve multiple and redundant secondary FEPs that address size and effects of meteorite impacts. By working to the primary FEP description, the subject-matter experts assigned to the primary FEP will address all relevant secondary FEPs, and arguments for secondary FEPs will be included in the primary FEP analysis. Secondary FEPs either are FEPs that are completely redundant or that can be aggregated into a single primary FEP. If a secondary FEP is deemed to be singularly important with regard to a technical issue, it may be elevated and assigned as a primary FEP.

Each primary FEP and its corresponding secondary FEPs are assigned to various Process Model Reports (PMRs). Table 2 shows the 87 primary FEPs assigned to the Waste Form PMR. The

FEPs discussed in this AMR have Document Identifier ANL-WIS-MD-000009. Gray shading on the tables indicates FEPs discussed in this AMR. .

1.3 FEPs SCREENING PROCESS

The second step in the scenario-development methodology includes the screening of each primary FEP. Each primary FEP has been screened for inclusion or exclusion in the TSPA against the screening criteria listed in Figure 1. The criteria are stated in proposed 10 CFR Part 63 (64 FR 8640), DOE's Interim Guidance (Dyer 1999), and in Environmental Protection Agency's (EPA's) proposed rule 40 CFR Part 197 (64 FR 46976). FEPs are excluded from the TSPA for one of the following reasons:

1. They are specifically ruled out by regulation, are contrary to the stated regulatory assumptions, or are in conflict with statements made in background information regarding intent or directions of the regulations;

2. They can be shown to have a probability of occurrence less than 10^{-4} in 10^{4} years; or

3. Their occurrence can be shown to have no significant effect on the overall performance of the system.

The decision to include or exclude a FEP related to the disposal system is based on three requirements for PA described in the NRC's proposed rule, 10 CFR 63 (NRC 1998c, §63.114):

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

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

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

In accordance with its proposed rule (10 CFR Part 63, 64 FR 8640; Dyer 1999), the NRC provides additional guidance in the TSPAI IRSR (NRC 2000, Section 4.2.3) on the screening process to follow to exclude FEPs. The guidance states four criteria that will be used; the first three criteria deal with evaluating the probability of the FEP (Fig. 1). The fourth deals with evaluating the consequence:

- "Criterion T1: Categories of processes and events that are not credible for the YM potential repository because of waste characteristics are identified and sufficient justification is provided for DOE's conclusions."

- "Criterion T2: The probability assigned to each category of processes and events not screened based on criterion T1 or criterion T2 (sic) is consistent with site information, well documented, and appropriately considers uncertainty."

- "Criterion T3: DOE has demonstrated that processes and events screened from the PA on the basis of their probability of occurrence, have a probability of less that one chance in 10,000 of occurring over 10,000 years."

- "Criterion T4: DOE has demonstrated that categories of processes and events omitted from the PA on the basis that their omission would not significantly change the calculated expected annual dose, do not significantly change the calculated expected annual dose."

In summary, the first criterion permits an exclusion argument for FEPs that are not credible because of waste characteristics, repository design, or site characteristics provided NRC staff finds sufficient justification. The second and third criteria deal with the assignment of probabilities to FEPs and demonstrating that the probabilities of excluded FEPs are below the quantitative probability limit of 10^{-4} over 10^4 yr. The fourth criterion permits exclusion of FEPs that do not significantly change the calculated expected annual dose provided the NRC staff finds sufficient support in accompanying discussions or calculations, including the use of either bounding or representative estimates. As regards the fourth criterion, the NRC report further states (NRC 2000, Section 4.2.3), "Detailed calculations of the consequences are not required for screening purposes...The amount of information required to support excluding categories of processes and events from the performance assessment may vary from one category to another, based on the processes and events involved."

None of the waste form FEPs were excluded by regulation. Therefore, based on the four criteria stated above, a statement on whether to Include or Exclude the FEP is provided in this report.

1.3.1 Screening Decisions

Most of the FEPs related to waste form that have been excluded from further consideration were excluded based on the fourth criterion, "no significant change in the expected annual dose." However, since bounding estimates are often used in the TSPA-SR, the argument for exclusion is often that the exclusion of the FEP provides a bound on the expected annual dose (i.e., inclusion of the FEP would only decrease the expected annual dose). For this case, the succinct description of the FEP reasoning used in the summary tables is "Excluded based on low consequence (conservative bound)." All FEPs excluded based on "no significant change in the expected annual dose" are succinctly described as "Excluded based on low consequence." Several waste-form FEPs have been excluded because the FEP is not credible for the characteristics of the waste proposed for disposal at Yucca Mountain. The succinct description for this reasoning is "Excluded based on low probability (not credible)." Probability estimates used in the FEP screening process may be based on technical analysis of the past frequency of similar events (such as seismic events) or, in some cases, on expert elicitation. Probability arguments, in general, require including some information about the magnitude of the event in its definition. Probability arguments are also sensitive to the spatial and temporal scales at which

FEPs are defined. For example, definition of the probability of meteorite impact depends on the size of the meteorite of interest and must consider that meteorite impacts are less likely in shorter time intervals and at smaller locations. Probability arguments, therefore, are made at reasonably coarse scales.

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

Some FEPs discussed in this AMR have consequences associated with their occurrence that would tend to improve overall performance, rather than to degrade it. Where these potentially beneficial consequences are significant, the process has been included in the TSPA. For those cases where the potential beneficial consequences are not significant, FEPs have been excluded from the TSPA, consistent with the proposed regulatory guidance (10 CFR Part 63, 64 FR 8640; or Dyer 1999, 114e, f) that allows exclusion of FEPs that have no significant impact on overall performance. Text within this AMR identifies these FEPs as having been excluded on the basis of low (beneficial) consequence. This term indicates that the only plausible consequences associated with these FEPs have been shown to improve, rather than degrade, overall performance, and that these consequences have not been included in the TSPA. To the extent that these FEPs might have any affect on the estimate of overall performance, their exclusion is conservative.

For the 87 waste-form FEPs, Table 2 presents the primary FEPs screened in the waste-form group. Although the primary FEP are coarse aggregates of FEPs, suitable for analysis, situations may arise in which a primary FEP contains some secondary FEPs that are *Included* and some that are *Excluded.* In these situations the screening decision specifies which elements are *Included* and which are *Excluded.* For FEPs assigned to this AMR, the disposition of the individual FEP is included in Section 6. Each discussion provides documentation of both the screening arguments and the TSPA disposition. Two other waste-form AMRs provide documentation for the remaining waste-form FEPs. Table 3 lists all the secondary FEPs that have been considered in the development of the waste-form primary FEPs.

Table 2. Primary Waste-Form FEPs
(Gray shading indicates primary miscellaneous waste-form FEPs in this AMR)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
1.2.04.04.00	Magma Interacts with Waste	ANL-WIS-MD-000009
2.1.01.01.00	Waste Inventory	ANL-WIS-MD-000009 ANL-WIS-MD-000006
2.1.01.02.00	Codisposal/Collocation of Waste	ANL-WIS-MD-000009
2.1.01.03.00	Heterogeneity of Waste Forms	ANL-WIS-MD-000009
2.1.01.04.00	Spatial Heterogeneity of Emplaced Waste	ANL-WIS-MD-000009
2.1.02.01.00	DSNF Degradation, Alteration, and Dissolution	ANL-WIS-MD-000009
2.1.02.02.00	CSNF Alteration, Dissolution, and Radionuclide Release	ANL-WIS-MD-000009
2.1.02.03.00	Glass Degradation, Alteration, and Dissolution	ANL-WIS-MD-000009
2.1.02.04.00	Alpha Recoil Enhances Dissolution	ANL-WIS-MD-000009
2.1.02.05.00	Glass Cracking and Surface Area	ANL-WIS-MD-000009
2.1.02.06.00	Glass recrystallization	ANL-WIS-MD-000009
2.1.02.07.00	Gap and Grain Release of Radionuclides after Cladding Perforation	ANL-WIS-MD-000008
2.1.02.08.00	Pyrophoricity	ANL-WIS-MD-000009
2.1.02.09.00	Void Space (in waste package)	ANL-WIS-MD-000009
2.1.02.10.00	Cellulosic Degradation	ANL-WIS-MD-000009
2.1.02.11.00	Waterlogged Rods	ANL-WIS-MD-000008
2.1.02.12.00	Cladding Degradation before YMP Receives It	ANL-WIS-MD-000008 ANL-WIS-MID-000048
2.1.02.13.00	General Corrosion of Cladding	ANL-WIS-MD-000008
2.1.02.14.00	Microbial Corrosion (MIC) of Cladding	ANL-WIS-MD-000008
2.1.02.15.00	Acid Corrosion of Cladding from Radiolysis	ANL-WIS-MD-000008
2.1.02.16.00	Localized Corrosion of Cladding through Pitting	ANL-WIS-MD-000008
2.1.02.17.00	Localized Corrosion (crevice corrosion) of Cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000012
2.1.02.18.00	High Dissolved Silica Content of Waters Enhances Corrosion of Cladding	ANL-WIS-MD-000008
2.1.02.19.00	Creep Rupture of Cladding	ANL-WIS-MD-000008
2.1.02.20.00	Pressurization from He Production Causes Cladding Failure	ANL-WIS-MD-000008
2.1.02.21.00	Stress Corrosion Cracking (SCC) of Cladding	ANL-WIS-MD-000008
2.1.02.22.00	Hydride Embrittlement of Cladding	ANL-WIS-MD-000008 ANL-EBS-MD-000011
2.1.02.23.00	Cladding Unzipping	ANL-WIS-MD-000008 ANL-EBS-MD-000013 ANL-EBS-MD-000014
2.1.02.24.00	Mechanical Failure of Cladding	ANL-WIS-MD-000008
2.1.02.25.00	DSNF Cladding Degradation	ANL-WIS-MD-000008
2.1.02.26.00	Diffusion Controlled Cavity Growth (DCCG) Concerns	ANL-WIS-MD-000008
2.1.02.27.00	Localized Corrosion Perforation of a Cladding by Fluoride	ANL-WIS-MD-000008
2.1.02.28.00	Various Features of the Approximately 250 DSNF Fuel Types and Grouping into Waste Categories	ANL-WIS-MD-000009
2.1.02.29.00	Flammable Gas Generation from DSNF	ANL-WIS-MD-000009
2.1.03.06.00	Internal Corrosion of Waste Container	ANL-WIS-MD-000009
2.1.07.01.00	Rockfall (large block)	ANL-WIS-MD-000008
2.1.08.07.00	Pathways for Unsaturated Flow and Transport in the Easte and EBS	ANL-WIS-MD-000009
2.1.08.10.00	Desaturation/Dewatering of the Repository	ANL-WIS-MD-000009
2.1.08.08.00	Induced Hydrological Changes in the Waste and EBS	ANL-WIS-MD-000009
2.1.08.15.00	Waste Form and Backfill Consolidation	ANL-WIS-MD-000009
2.1.09.01.00	Properties of the Potential Carrier Plume in the Waste and EBS	ANL-WIS-MD-000009
2.1.09.02.00	Interaction with Corrosion Products	ANL-WIS-MD-000009
2.1.09.03.00	Volume Increase of Corrosion Products	ANL-WIS-MD-000008
2.1.09.04.00	Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and EBS	ANL-WIS-MD-000009
2.1.09.05.00	In-Package Sorption	ANL-WIS-MD-000009
2.1.09.06.00	Reduction-Oxidation Potential in Waste and EBS	ANL-WIS-MD-000009

Table 2. Primary Waste-Form FEPs (Gray shading indicates primary miscellaneous waste-form FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.09.07.00	Reaction Kinetics in Waste and EBS	ANL-WIS-MD-000009
2.1.09.08.00	Chemical Gradients/Enhanced Diffusion in Waste and EBS	ANL-WIS-MD-000009
2.1.09.09.00	Electrochemical Effects (electrophoresis and galvanic coupling) in Waste and EBS	ANL-WIS-MD-000009 ANL-EBS-PA-000002
2.1.09.10.00	Secondary Phase Effects on Dissolved Radionuclide Concentrations at the Waste Form	ANL-WIS-MD-000009
2.1.09.11.00	Waste Rock Contact	ANL-WIS-MD-000009
2.1.09.12.00	Rind (altered zone) Formation in Waste, EBS, and Adjacent Rock	ANL-WIS-MD-000009
2.1.09.13.00	Complexation by Organics in Waste and EBS	ANL-WIS-MD-000009
2.1.09.14.00	Colloid Formation in Waste and EBS	ANL-WIS-MD-000012
2.1.09.15.00	Formation of True (real) Colloids in Waste and EBS	ANL-WIS-MD-000012
2.1.09.16.00	Formation of Pseudo-Colloids (natural) in Waste and EBS	ANL-WIS-MD-000012
2.1.09.16.01	Colloidal Phases Produced by Coprecipitation (in the waste package or EBS)	ANL-WIS-MD-000012
2.1.09.17.00	Formation of Pseudo-Colloids (corrosion products) in Waste and EBS	ANL-WIS-MD-000012
2.1.09.18.00	Microbial Colloid Transport in the Waste and EBS.	ANL-WIS-MD-000012
2.1.09.19.00	Colloid Transport and Sorption in the Waste and EBS.	ANL-WIS-MD-000012
2.1.09.20.00	Colloid Filtration in the Waste and EBS	ANL-WIS-MD-000012
2.1.09.21.00	Suspensions of Particles Larger than Colloids	ANL-WIS-MD-000012
2.1.09.22.00	Colloidal Sorption at the Groundwater Interface	ANL-WIS-MD-000012
2.1.09.23.00	Colloidal Stability and Concentration Dependence on Aqueous Chemistry	ANL-WIS-MD-000012
2.1.09.24.00	Colloid Diffusion	ANL-WIS-MD-000012
2.1.09.25.00	Colloidal Phases Produced by Coprecipitation (in the waste package or EBS)	ANL-WIS-MD-000012
2.1.09.26.00	Colloid Gravitational Settling	ANL-WIS-MD-000012
2.1.10.01.00	Biological Activity in Waste and EBS	ANL-WIS-MD-000012
2.1.11.01.00	Heat Output/Temperature in Waste and EBS	ANL-WIS-MD-000009
2.1.11.03.00	Exothermic and other Thermal Reactions in Waste and EBS	ANL-WIS-MD-000009
2.1.11.04.00	Temperature Effects/Coupled Processes in Waste and EBS	ANL-WIS-MD-000009
2.1.11.05.00	Differing Thermal Expansion of Repository Components	ANL-WIS-MD-000009 ANL-EBS-MD-000015
2.1.11.07.00	Thermally Induced Stress Changes in Waste and EBS	ANL-WIS-MD-000009 ANL-WIS-MD-000008
2.1.11.08.00	Thermal Effects: Chemical and Microbiological Changes in the Waste and EBS	ANL-WIS-MD-000009
2.1.11.09.00	Thermal Effects on Liquid or Two-Phase Fluid Flow in the Waste and EBS	ANL-WIS-MD-000009
2.1.11.10.00	Thermal Effects on Diffusion (Soret effect) in Waste and EBS	ANL-WIS-MD-000009
2.1.12.01.00	Gas Generation	ANL-WIS-MD-000009
2.1.12.02.00	Gas Generation (He) from Fuel Decay	ANL-WIS-MD-000009
2.1.12.03.00	Gas Generation (H_2) from Metal Corrosion	ANL-WIS-MD-000009
2.1.12.04.00	Gas Generation (CO_2, CH_4, H_2S) from Microbial Degradation	ANL-WIS-MD-000009
2.1.12.06.00	Gas Transport in Waste and EBS	ANL-WIS-MD-000009
2.1.12.07.00	Radioactive Gases in Waste and EBS	ANL-WIS-MD-000009
2.1.12.08.00	Gas Explosions	ANL-WIS-MD-000009
2.1.13.01.00	Radiolysis	ANL-WIS-MD-000009
2.1.13.02.00	Radiation Damage in Waste and EBS	ANL-EBS-MD-000015 ANL-WIS-MD-000009 ANL-WIS-MD-000010
2.1.13.03.00	Mutation	ANL-WIS-MD-000012
2.2.08.12.00	Use of J-13 Well Water as a Surrogate for Water Flowing into the EBS and Waste	ANL-WIS-MD-000009
3.1.01.01.00	Radioactive Decay and Ingrowth	ANL-WIS-MD-000009
3.2.07.01.00	Isotopic dilution	ANL-WIS-MD-000009

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
1.2.04.04.00	Magma Interacts with Waste	ANL-WIS-MD-000009
1.2.04.04.01	Magmatic volatiles attack waste	SECONDARY FEP
1.2.04.04.02	Dissolution of spent fuel in magma	SECONDARY FEP
1.2.04.04.03	Dissolution of other waste in magma	SECONDARY FEP
1.2.04.04.04	Heating of waste container by magma (without contact)	SECONDARY FEP
1.2.04.04.05	Failure of waste container by direct contact w/magma	SECONDARY FEP
1.2.04.04.06	Fragmentation	SECONDARY FEP
2.1.01.01.00	Waste Inventory	ANL-WIS-MD-000009 ANL-WIS-MD-000006
2.1.01.01.01	Inventory	SECONDARY FEP
2.1.01.01.02	Inventory	SECONDARY FEP
2.1.01.01.03	Changes in radionuclide inventory (in waste form)	SECONDARY FEP
2.1.01.01.04	Waste product (glass)	SECONDARY FEP
2.1.01.01.05	Exotic Fuels	SECONDARY FEP
2.1.01.01.06	DOE SNF gap radionuclide inventory	SECONDARY FEP
2.1.01.01.07	DOE SNF initial radionuclide inventory	SECONDARY FEP
2.1.01.01.08	DOE SNF structure	SECONDARY FEP
2.1.01.01.09	DOE SNF initial radionuclide inventory	SECONDARY FEP
2.1.01.01.10	DOE SNF hazardous chemical inventory	SECONDARY FEP
2.1.01.02.00	Codisposal/Collocation of Waste	ANL-WIS-MD-000009
2.1.01.02.01	Other waste	SECONDARY FEP
2.1.01.02.02	Co-disposal of reactive wastes	SECONDARY FEP
2.1.01.02.03	Near storage of other waste	SECONDARY FEP
2.1.01.02.04	DOE SNF/HLW Glass Interactions	SECONDARY FEP
2.1.01.02.05	DOE SNF waste package placement	SECONDARY FEP
2.1.01.02.06	DOE SNF canister arrangement within waste package	SECONDARY FEP
2.1.01.02.07	DOE SNF collocation with HLW	SECONDARY FEP
2.1.01.02.08	DOE SNF geometry	SECONDARY FEP
2.1.01.02.09	DOE SNF waste package placement	SECONDARY FEP
2.1.01.02.10	DOE SNF collocation with HLW (waste-form degradation impact)	SECONDARY FEP
2.1.01.02.11	DOE SNF collocation with HLW (radionuclide mobilization impact)	SECONDARY FEP
2.1.01.02.12	DOE SNF collocation with HLW (cladding degradation impact)	SECONDARY FEP
2.1.01.03.00	Heterogeneity of Waste Forms	ANL-WIS-MD-000009
2.1.01.03.01	Damaged or deviating fuel	SECONDARY FEP
2.1.01.03.02	Heterogeneity of waste form	SECONDARY FEP
2.1.01.03.03	Deviant inventory flask	SECONDARY FEP
2.1.01.03.04	DOE SNF canister atmosphere	SECONDARY FEP
2.1.01.04.00	Spatial Heterogeneity of Emplaced Waste	ANL-WIS-MD-000009
2.1.02.01.00	DSNF Degradation, Alteration, and Dissolution	ANL-WIS-MD-000009
2.1.02.01.01	DOE SNF Dissolution	SECONDARY FEP
2.1.02.01.02	Alteration/Dissolution of DOE SNF	SECONDARY FEP
2.1.02.01.03	Oxidation of DOE SNF	SECONDARY FEP
2.1.02.01.04	Alteration/Dissolution of Pu Ceramic Waste	SECONDARY FEP
2.1.02.01.05	High Integrity Canisters for DOE SNF	SECONDARY FEP
2.1.02.02.00	CSNF Alteration, Dissolution, and Radionuclide Release	ANL-WIS-MD-000009
2.1.02.02.01	Source terms (expected)	SECONDARY FEP
2.1.02.02.02	Source terms (other) (in waste form)	SECONDARY FEP
2.1.02.02.03	Stability of U02 (in waste form)	SECONDARY FEP
2.1.02.02.04	Degradation of fuel elements	SECONDARY FEP
2.1.02.02.05	Corrosion of metal parts (in waste form)	SECONDARY FEP
2.1.02.02.06	Corrosion prior to wetting	SECONDARY FEP
2.1.02.02.07	Radionuclide Release (Diffusion) Through Failed Cladding	SECONDARY FEP
2.1.02.02.08	Water turnover, steel vessel	SECONDARY FEP
2.1.02.02.09	Dissolution chemistry (in waste and EBS)	SECONDARY FEP
2.1.02.02.10	Release from fuel matrix (release/migration factors)	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.02.02.11	Release from metal parts	SECONDARY FEP
2.1.02.02.12	Total release from fuel elements	SECONDARY FEP
2.1.02.02.13	Dissolution of waste (release/migration factors)	SECONDARY FEP
2.1.02.02.14	Release of radionuclides from the failed canister	SECONDARY FEP
2.1.02.02.15	Transport and release of nuclide, failed canister	SECONDARY FEP
2.1.02.03.00	Glass Degradation, Alteration, and Dissolution	ANL-WIS-MD-000009
2.1.02.03.01	Degradation and alteration of glass waste form	SECONDARY FEP
2.1.02.03.02	Phase separation (in waste form)	SECONDARY FEP
2.1.02.03.03	Congruent dissolution (in waste form)	SECONDARY FEP
2.1.02.03.04	Rate of glass dissolution	SECONDARY FEP
2.1.02.03.05	Selective leaching (in waste form)	SECONDARY FEP
2.1.02.03.06	Coprecipitates/solid solutions (in waste form)	SECONDARY FEP
2.1.02.03.07	Precipitation of silicates/silica gel (in waste form)	SECONDARY FEP
2.1.02.03.08	Iron corrosion products	SECONDARY FEP
2.1.02.03.09	Radionuclide release from glass	SECONDARY FEP
2.1.02.03.10	Composition of DHLW Glass	SECONDARY FEP
2.1.02.04.00	Alpha Recoil Enhances Dissolution	ANL-WIS-MD-000009
2.1.02.04.01	Recoil of alpha-decay	SECONDARY FEP
2.1.02.05.00	Glass Cracking and Surface Area	ANL-WIS-MD-000009
2.1.02.05.01	Solute transport resistance (in waste form)	SECONDARY FEP
2.1.02.06.00	Glass Recrystallization	ANL-WIS-MD-000009
2.1.02.07.00	Gap and Grain Release of Radionuclides after Cladding Perforation	ANL-WIS-MD-000008
2.1.02.07.01	Gap and grain release	SECONDARY FEP
2.1.02.07.02	Pb-I reactions (in waste form)	SECONDARY FEP
2.1.02.07.03	I, Cs-migration to fuel surface	SECONDARY FEP
2.1.02.08.00	Pyrophoricity	ANL-WIS-MD-000009
2.1.02.08.01	DOE SNF pyrophoricity	SECONDARY FEP
2.1.02.08.02	DOE SNF pyrophoric event (waste heat impact)	SECONDARY FEP
2.1.02.08.03	DOE SNF pyrophoric event (waste package degradation impact)	SECONDARY FEP
2.1.02.08.04	Acetylene generation from DSNF WFMisc—Flammable Gases Generation from DSNF- YMP	SECONDARY FEP
2.1.02.08.05	DOE SNF pyrophoric event (waste-form degradation impact)	SECONDARY FEP
2.1.02.08.06	DOE SNF pyrophoric event (cladding degradation impact)	SECONDARY FEP
2.1.02.09.00	Void Space (in waste package)	ANL-WIS-MD-000009
2.1.02.10.00	Cellulosic Degradation	ANL-WIS-MD-000009
2.1.02.11.00	Waterlogged Rods	ANL-WIS-MD-000008
2.1.02.12.00	Cladding Degradation before YMP Receives It	ANL-WIS-MD-000008 ANL-WIS-MID-000048
2.1.02.12.01	Pin Degradation During Reactor Operation	SECONDARY FEP
2.1.02.12.02	Pin Degradation During Spent Fuel Pool Storage	SECONDARY FEP
2.1.02.12.03	Pin Degradation During Dry Storage	SECONDARY FEP
2.1.02.12.04	Pin Degradation During Fuel Shipment and Handling	SECONDARY FEP
2.1.02.13.00	General Corrosion of Cladding	ANL-WIS-MD-000008
2.1.02.13.01	Cladding Degradation Mechanisms at YMP, Pre-Pin Failure	SECONDARY FEP
2.1.02.13.02	Corrosion (of cladding)	SECONDARY FEP
2.1.02.14.00	Microbial Corrosion (MIC) of Cladding	ANL-WIS-MD-000008
2.1.02.15.00	Acid Corrosion of Cladding from Radiolysis	ANL-WIS-MD-000008
2.1.02.16.00	Localized Corrosion of Cladding through Pitting	ANL-WIS-MD-000008
2.1.02.17.00	Localized Corrosion (crevice corrosion) of Cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000012
2.1.02.18.00	High Dissolved Silica Content of Waters Enhances Corrosion of Cladding	ANL-WIS-MD-000008
2.1.02.19.00	Creep Rupture of Cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000007
2.1.02.19.01	Thermal Cracking (in waste and EBS)	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.02.20.00	Pressurization from He Production Causes Cladding Failure	ANL-WIS-MD-000008
2.1.02.21.00	Stress Corrosion Cracking (SCC) of Cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000007
2.1.02.21.01	Inside Out from Fission Products (iodine) (failure of cladding)	SECONDARY FEP
2.1.02.21.02	Outside In from Salts or waste package Chemicals (failure of cladding)	SECONDARY FEP
2.1.02.21.03	Stress-corrosion cracking of Zircaloy cladding	SECONDARY FEP
2.1.02.22.00	Hydride Embrittlement of Cladding	ANL-WIS-MD-000008 ANL-EBS-MD-000011
2.1.02.22.01	Hydride Embrittlement From Zirconium Corrosion (of cladding)	SECONDARY FEP
2.1.02.22.02	Hydride Embrittlement From WP Corrosion & H2 Absorption (of cladding)	SECONDARY FEP
2.1.02.22.03	Hydride Embrittlement From Galvanic Corrosion of WP contacting Cladding	SECONDARY FEP
2.1.02.22.04	Delayed Hydride Cracking (of cladding) WFClad—Delayed Hydride Cracking (DHC) of Cladding	SECONDARY FEP
2.1.02.22.05	Hydride Reorientation (of cladding)	SECONDARY FEP
2.1.02.22.06	Hydrogen Axial Migration (of cladding)	SECONDARY FEP
2.1.02.22.07	Hydride Embrittlement from Fuel Reaction (causes failure of cladding) WFClad—Hydride Embrittlement from Fuel Reaction	SECONDARY FEP
2.1.02.23.00	Cladding Unzipping	ANL-WIS-MD-000008 ANL-EBS-MD-000013 ANL-EBS-MD-000014
2.1.02.23.01	Cladding Degradation after Initial Cladding Perforation	SECONDARY FEP
2.1.02.23.02	Dry Oxidation of Fuel (causes failure of cladding) WFClad—Dry Oxidation of Fuel	SECONDARY FEP
2.1.02.23.03	Wet Oxidation of Fuel (causes failure of cladding) WFClad—Wet Oxidation of Fuel	SECONDARY FEP
2.1.02.24.00	Mechanical Failure of Cladding	ANL-WIS-MD-000008
2.1.02.25.00	DSNF cladding degradation	ANL-WIS-MD-000009
2.1.02.25.01	DOE SNF cladding material	SECONDARY FEP
2.1.02.25.02	DOE SNF cladding condition	SECONDARY FEP
2.1.02.25.03	Internal Canister/Cladding Corrosion due to DOE SNF	SECONDARY FEP
2.1.09.26.00	Diffusion Controlled Cavity Growth (DCCG) Concerns	ANL-WIS-MD-000008
2.1.02.27.00	Localized Corrosion Perforation of a Cladding by Fluoride	ANL-WIS-MD-000008
2.1.02.28.00	Various Features of the Approximately 250 DSNF Types and Grouping into Waste Categories	ANL-WIS-MD-000009
2.1.02.29.00	Flammable Gas Generation from DSNF	ANL-WIS-MD-000009
2.1.03.06.00	Internal Corrosion of Waste Container	ANL-WIS-MD-000009
2.1.03.06.01	DOE SNF waste package internal corrosion	SECONDARY FEP
2.1.07.01.00	Rockfall (large block)	ANL-WIS-MD-000007 ANL-WIS-MD-000008
2.1.07.01.01	Rockbursts in container holes	SECONDARY FEP
2.1.07.01.02	Cave ins	SECONDARY FEP
2.1.07.01.03	Cave in (in waste and EBS)	SECONDARY FEP
2.1.07.01.04	Roof falls	SECONDARY FEP
2.1.08.07.00	Pathways for Unsaturated Flow and Transport in the Waste and EBS	ANL-WIS-MD-000009
2.1.08.07.01	Residual canister (crack/holes effects)	SECONDARY FEP
2.1.08.07.02	Properties of failed canister	SECONDARY FEP
2.1.08.07.03	Container-partial corrosion	SECONDARY FEP
2.1.08.07.04	Hydraulic conductivity (in waste and EBS)	SECONDARY FEP
2.1.08.07.05	Consolidation of waste	SECONDARY FEP
2.1.08.07.06	Channeling within the waste	SECONDARY FEP
2.1.08.07.07	Unsaturated transport (water transport)	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.08.07.08	Radionuclide transport (water transport)	SECONDARY FEP
2.1.08.08.00	Induced Hydrological Changes in the Waste and EBS	ANL-WIS-MD-000009
2.1.08.10.00	Desaturation/Dewatering of the Repository	ANL-WIS-MD-000009
2.1.08.10.01	Dewatering of host rock (in waste and EBS)	SECONDARY FEP
2.1.08.10.02	Dewatering	SECONDARY FEP
2.1.08.15.00	Waste Form and Backfill Consolidation	ANL-WIS-MD-000009
2.1.09.01.00	Properties of the Potential Carrier Plume in the Waste and EBS	ANL-WIS-MD-000009
2.1.09.01.01	Reactions with cement pore water	SECONDARY FEP
2.1.09.01.02	Reactions with cement pore water	SECONDARY FEP
2.1.09.01.03	Induced chemical changes (in waste and EBS)	SECONDARY FEP
2.1.09.01.04	Interactions of host materials and ground water with repository material	SECONDARY FEP
2.1.09.01.05	TRU silos cementitious plume	SECONDARY FEP
2.1.09.01.06	Water chemistry, canister	SECONDARY FEP
2.1.09.01.07	Transport of chemically-active substances into the near-field	SECONDARY FEP
2.1.09.01.08	Incomplete near-field chemical conditioning	SECONDARY FEP
2.1.09.01.09	Chemical processes (in waste and EBS)	SECONDARY FEP
2.1.09.01.10	Hyperalkaline carrier plume forms	SECONDARY FEP
2.1.09.01.11	Chemical interactions (in waste and EBS)	SECONDARY FEP
2.1.09.01.12	TRU alkaline or organic plume	SECONDARY FEP
2.1.09.01.13	Interactions of waste and repository materials with host materials	SECONDARY FEP
2.1.09.01.14	TRU alkaline or organic plume	SECONDARY FEP
2.1.09.02.00	Interaction with Corrosion Products	ANL-WIS-MD-000009
2.1.09.02.01	Interactions with corrosion products and waste	SECONDARY FEP
2.1.09.02.02	Effects of metal corrosion (in waste and EBS)	SECONDARY FEP
2.1.09.02.03	Container corrosion products	SECONDARY FEP
2.1.09.02.04	Chemical buffering (canister corrosion products)	SECONDARY FEP
2.1.09.02.05	Radionuclide sorption and co-precipitation (in EBS)	SECONDARY FEP
2.1.09.03.00	Volume Increase of Corrosion Products	ANL-WIS-MD-000008
2.1.09.03.01	Swelling of corrosion products (in waste and EBS)	SECONDARY FEP
2.1.09.04.00	Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and EBS	ANL-WIS-MD-000009
2.1.09.04.01	Elemental Solubility (in waste and EBS)	SECONDARY FEP
2.1.09.04.02	Speciation (in waste and EBS)	SECONDARY FEP
2.1.09.04.03	Geochemical Pump (in waste and EBS)	SECONDARY FEP
2.1.09.04.04	Precipitation and Dissolution (in waste and EBS)	SECONDARY FEP
2.1.09.04.05	Selective Dissolution of Contaminants Contained in SNF	SECONDARY FEP
2.1.09.04.06	Precipitation (release/migration factors)	SECONDARY FEP
2.1.09.04.07	Speciation control of contaminants by hyperalkaline plume formed in the EBS	SECONDARY FEP
2.1.09.04.08	Solubility within fuel matrix	SECONDARY FEP
2.1.09.04.09	Solubility and precipitation (contaminant speciation and solubility)	SECONDARY FEP
2.1.09.04.10	Solubility limit (contaminant speciation and solubility)	SECONDARY FEP
2.1.09.04.11	Radionuclide source term (contaminant speciation and solubility)	SECONDARY FEP
2.1.09.04.12	Elemental solubility/precipitation (contaminant speciation and solubility)	SECONDARY FEP
2.1.09.04.13	Speciation (contaminant speciation and solubility)	SECONDARY FEP
2.1.09.05.00	In-Package Sorption	ANL-WIS-MD-000009
2.1.09.05.01	Selective sorption of Pu from solution	SECONDARY FEP
2.1.09.05.02	Sorption	SECONDARY FEP
2.1.09.05.03	Radionuclide retardation	SECONDARY FEP
2.1.09.05.04	Sorption on filling material	SECONDARY FEP
2.1.09.06.00	Reduction-Oxidation Potential in Waste and EBS	ANL-WIS-MD-000009
2.1.09.06.01	Redox front (in waste and EBS)	SECONDARY FEP
2.1.09.06.02	Reduction-oxidation fronts (in waste and EBS)	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.09.06.03	Localized reducing zones (in waste and EBS)	SECONDARY FEP
2.1.09.06.04	Redox front (in buffer/backfill)	SECONDARY FEP
2.1.09.06.05	Fe control of oxidation state of contaminants	SECONDARY FEP
2.1.09.07.00	Reaction Kinetics in Waste and EBS	ANL-WIS-MD-000009
2.1.09.07.01	Chemical kinetics (in waste and EBS)	SECONDARY FEP
2.1.09.08.00	Chemical Gradients/Enhanced Diffusion in Waste and EBS	ANL-WIS-MD-000009
2.1.09.08.01	Enhanced diffusion (in waste and EBS)	SECONDARY FEP
2.1.09.08.02	Chemical gradients (in waste and EBS)	SECONDARY FEP
2.1.09.08.03	Diffusion in and through failed canister	SECONDARY FEP
2.1.09.09.00	Electrochemical Effects (electrophoresis and galvanic coupling) in Waste and EBS	ANL-WIS-MD-000009 ANL-EBS-PA-000002
2.1.09.09.01	Repository induced Pb/Cu electrochemical reactions	SECONDARY FEP
2.1.09.09.02	Natural telluric electrochemical reactions (in waste and EBS)	SECONDARY FEP
2.1.09.09.03	Electrochemical cracking (in waste and EBS)	SECONDARY FEP
2.1.09.09.04	Electrochemical effects/gradients (in waste and EBS)	SECONDARY FEP
2.1.09.09.05	Electrochemical effects of metal corrosion	SECONDARY FEP
2.1.09.09.06	Electrochemical effects (in waste and EBS)	SECONDARY FEP
2.1.09.09.07	Galvanic coupling (in waste and EBS)	SECONDARY FEP
2.1.09.09.08	Electrophoresis (in waste and EBS)	SECONDARY FEP
2.1.09.09.09	Electrochemical gradients (in waste and EBS)	SECONDARY FEP
2.1.09.09.10	Galvanic coupling (in waste and EBS)	SECONDARY FEP
2.1.09.09.11	Galvanic coupling (in waste and EBS)	SECONDARY FEP
2.1.09.10.00	Secondary Phase Effects on Dissolved Radionuclide Concentrations at the Waste Form	ANL-WIS-MD-000009
2.1.09.11.00	Waste Rock Contact	ANL-WIS-MD-000009
2.1.09.12.00	Rind (altered zone) Formation in Waste, EBS, and Adjacent Rock	ANL-WIS-MD-000009
2.1.09.12.01	Deep alteration of the porosity of drift walls	SECONDARY FEP
2.1.09.13.00	Complexation by Organics in Waste and EBS	ANL-WIS-MD-000009
2.1.09.13.01	Methylation (in waste and EBS)	SECONDARY FEP
2.1.09.13.02	Humic and fulvic acids	SECONDARY FEP
2.1.09.13.03	Complexation by organics	SECONDARY FEP
2.1.09.13.04	Fulvic acid	SECONDARY FEP
2.1.09.13.05	Humic acid	SECONDARY FEP
2.1.09.13.06	Complexing agents	SECONDARY FEP
2.1.09.13.07	Organics (complexing agents)	SECONDARY FEP
2.1.09.13.08	Organics (complexing agents)	SECONDARY FEP
2.1.09.13.09	Organic complexation	SECONDARY FEP
2.1.09.13.10	Organic ligands	SECONDARY FEP
2.1.09.13.11	Kinetics of organic complexation	SECONDARY FEP
2.1.09.13.12	Introduced complexing agents	SECONDARY FEP
2.1.09.14.00	Colloid Formation in Waste and EBS	ANL-WIS-MD-000012
2.1.09.14.01	Colloid generation-source (in waste and EBS)	SECONDARY FEP
2.1.09.14.02	Agglomeration of Pu colloids	SECONDARY FEP
2.1.09.14.03	Colloids (in waste and EBS)	SECONDARY FEP
2.1.09.14.04	Colloids/particles in canister	SECONDARY FEP
2.1.09.14.05	Colloid formation	SECONDARY FEP
2.1.09.14.06	Colloids	SECONDARY FEP
2.1.09.14.07	Colloids, complexing agents	SECONDARY FEP
2.1.09.14.08	Colloid generation and transport	SECONDARY FEP
2.1.09.14.09	Colloid formation, dissolution and transport	SECONDARY FEP
2.1.09.14.10	Colloid generation and transport	SECONDARY FEP
2.1.09.14.11	Colloid formation and stability	SECONDARY FEP
2.1.09.15.00	Formation of True (real) Colloids in Waste and EBS	ANL-WIS-MD-000012
2.1.09.16.00	Formation of Pseudo-Colloids (natural) in Waste and EBS	ANL-WIS-MD-000012
2.1.09.16.01	Pseudo-colloids	SECONDARY FEP
2.1.09.16.02	Pseudo-colloids	SECONDARY FEP
2.1.09.16.03	Natural colloids	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.09.16.04	Natural colloids	SECONDARY FEP
2.1.09.17.00	Formation of Pseudo-Colloids (corrosion products) in Waste and EBS	ANL-WIS-MD-000012
2.1.09.17.01	Colloid formation is associated with container hydrolysis products	SECONDARY FEP
2.1.09.18.00	Microbial Colloid Transport in the Waste and EBS	ANL-WIS-MD-000012
2.1.09.19.00	Colloid Transport and Sorption in the Waste and EBS	ANL-WIS-MD-000012
2.1.09.19.01	Colloid transport	SECONDARY FEP
2.1.09.20.00	Colloid Filtration in the Waste and EBS	ANL-WIS-MD-000012
2.1.09.20.01	Colloid Filtration by the Invert	SECONDARY FEP
2.1.09.20.02	Colloid Fltration (in pores and fractures)	SECONDARY FEP
2.1.09.20.03	Colloid Filtration	SECONDARY FEP
2.1.09.21.00	Suspensions of Particles Larger than Colloids	ANL-WIS-MD-000012
2.1.09.21.01	Suspended sediment transport	SECONDARY FEP
2.1.09.21.02	Rinse	SECONDARY FEP
2.1.09.22.00	Colloidal Sorption at the Groundwater Interface	ANL-WIS-MD-000012
2.1.09.23.00	Colloidal Stability and Concentration Dependence on Aqueous Chemistry	ANL-WIS-MD-000012
2.1.09.24.00	Colloid Diffusion	ANL-WIS-MD-000012
2.1.09.25.00	Colloidal Phases Produced by Coprecipitation (in the waste package or EBS)	ANL-WIS-MD-000012
2.1.09.26.00	Colloid Gravitational Settling	ANL-WIS-MD-000012
2.1.10.01.00	Biological Activity in Waste and EBS	ANL-WIS-MD-000012
2.1.10.01.01	Microbial activity accelerates corrosion of containers	SECONDARY FEP
2.1.10.01.02	Microbial activity accelerates corrosion of cladding	SECONDARY FEP
2.1.10.01.03	Microbial activity accelerates corrosion of contaminants	SECONDARY FEP
2.1.10.01.04	Microbes (in waste and EBS)	SECONDARY FEP
2.1.10.01.05	Microorganisms (in waste and EBS)	SECONDARY FEP
2.1.10.01.06	Microbiological effects (in waste and EBS)	SECONDARY FEP
2.1.10.01.07	Microbial activity (in waste and EBS)	SECONDARY FEP
2.1.10.01.08	Microbial activity (in waste and EBS)	SECONDARY FEP
2.1.10.01.09	Microbial activity (in waste and EBS)	SECONDARY FEP
2.1.10.01.10	Microbial interactions	SECONDARY FEP
2.1.10.01.11	Biofilms	SECONDARY FEP
2.1.11.01.00	Heat Output/Temperature in Waste and EBS	ANL-WIS-MD-000009
2.1.11.01.01	Glass temperature (in waste and EBS)	SECONDARY FEP
2.1.11.01.02	Canister temperature	SECONDARY FEP
2.1.11.01.03	Temperature, bentonite buffer	SECONDARY FEP
2.1.11.01.04	Temperature, canister	SECONDARY FEP
2.1.11.01.05	Temperature, tunnel backfill	SECONDARY FEP
2.1.11.01.06	Heat generation from waste containers	SECONDARY FEP
2.1.11.01.07	Radioactive decay heat	SECONDARY FEP
2.1.11.01.08	DOE SNF expected waste heat generation	SECONDARY FEP
2.1.11.01.09	DOE SNF expected waste heat generation	SECONDARY FEP
2.1.11.03.00	Exothermic and other Thermal Reactions in Waste and EBS	ANL-WIS-MD-000009
2.1.11.03.01	Concrete hydration	SECONDARY FEP
2.1.11.04.00	Temperature Effects/Coupled Processes in Waste and EBS	ANL-WIS-MD-000009
2.1.11.04.01	Thermal (processes)	SECONDARY FEP
2.1.11.04.02	Temperature effects (unexpected effects) (in waste and EBS)	SECONDARY FEP
2.1.11.04.03	Heat from radioactive decay (in waste and EBS)	SECONDARY FEP
2.1.11.04.04	Long-term transients (in waste and EBS)	SECONDARY FEP
2.1.11.04.05	Time dependence (in waste and EBS)	SECONDARY FEP
2.1.11.04.06	Coupled processes (in waste and EBS)	SECONDARY FEP
2.1.11.05.00	Differing Thermal Expansion of Repository Components	ANL-WIS-MD-000009 ANL-EBS-MD-000015
2.1.11.05.01	Differential thermal expansion of near-field barriers	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.11.05.02	Shearing of waste containers by secondary stresses from thermal expansion of the rock	SECONDARY FEP
2.1.11.05.03	Differential elastic response (in waste and EBS)	SECONDARY FEP
2.1.11.05.04	Non-elastic response (in waste and EBS)	SECONDARY FEP
2.1.11.07.00	Thermally Induced Stress Changes in Waste and EBS	ANL-WIS-MD-000009 ANL-WIS-MD-000008
2.1.11.07.01	Changes in *in situ* stress field (in waste and EBS)	SECONDARY FEP
2.1.11.07.02	Stress field changes, settling, subsidence or caving	SECONDARY FEP
2.1.11.08.00	Thermal Effects: Chemical and Microbiological Changes in the Waste and EBS	ANL-WIS-MD-000009
2.1.11.09.00	Thermal Effects on Liquid or Two-Phase Fluid Flow in the Waste and EBS	ANL-WIS-MD-000009
2.1.11.09.01	Convection effects on transport (Enhanced vapor diffusion)	SECONDARY FEP
2.1.11.09.02	Multiphase flow and gas-driven transport (water transport)	SECONDARY FEP
2.1.11.10.00	Thermal Effects on Diffusion (Soret effect) in Waste and EBS	ANL-WIS-MD-000009
2.1.11.10.01	Soret effect (in waste and EBS)	SECONDARY FEP
2.1.11.10.02	Thermal effects: Transport(diffusion) effects (in waste and EBS)	SECONDARY FEP
2.1.11.10.03	Soret effect (water transport)	SECONDARY FEP
2.1.12.01.00	Gas Generation	ANL-WIS-MD-000009
2.1.12.01.01	Formation of gases (in wastes and EBS)	SECONDARY FEP
2.1.12.01.02	Gas generation	SECONDARY FEP
2.1.12.01.03	Gas generation, buffer/backfill	SECONDARY FEP
2.1.12.01.04	Chemotoxic gases (in waste and EBS)	SECONDARY FEP
2.1.12.01.05	Pressurization (in waste and EBS)	SECONDARY FEP
2.1.12.02.00	Gas Generation (He) from Fuel Decay	ANL-WIS-MD-000009
2.1.12.02.01	Helium gas production	SECONDARY FEP
2.1.12.02.02	Internal pressure (in waste and EBS)	SECONDARY FEP
2.1.12.02.03	Gas generation, canister	SECONDARY FEP
2.1.12.02.04	Internal pressure (in waste and EBS)	SECONDARY FEP
2.1.12.02.05	He gas production (in waste and EBS)	SECONDARY FEP
2.1.12.03.00	Gas Generation (H_2) from Metal Corrosion	ANL-WIS-MD-000009
2.1.12.03.01	Chemical effects of corrosion	SECONDARY FEP
2.1.12.03.02	Effect of hydrogen on corrosion	SECONDARY FEP
2.1.12.03.03	Hydrogen production (in waste and EBS)	SECONDARY FEP
2.1.12.03.04	Hydrogen production by metal corrosion	SECONDARY FEP
2.1.12.04.00	Gas Generation (CO_2, CH_4, H_2S) from Microbial Degradation	ANL-WIS-MD-000009
2.1.12.04.01	Effect of temperature on microbial gas generation	SECONDARY FEP
2.1.12.04.02	Effect of pressure on microbial gas generation	SECONDARY FEP
2.1.12.04.03	Effect of radiation on microbial gas generation	SECONDARY FEP
2.1.12.04.04	Effect of biofilms on microbial gas generation	SECONDARY FEP
2.1.12.04.05	Methane and carbon dioxide by microbial degradation	SECONDARY FEP
2.1.12.06.00	Gas Transport in Waste and EBS	ANL-WIS-MD-000009
2.1.12.06.01	Thermo-chemical effects (related to gas in waste and EBS)	SECONDARY FEP
2.1.12.06.02	Gas transport	SECONDARY FEP
2.1.12.06.03	Gas effects (in waste and EBS)	SECONDARY FEP
2.1.12.06.04	Gas escape from canister	SECONDARY FEP
2.1.12.06.05	Gas flow and transport, buffer/backfill	SECONDARY FEP
2.1.12.06.06	Gas transport	SECONDARY FEP
2.1.12.06.07	Unsaturated flow due to gas production (in waste and EBS)	SECONDARY FEP
2.1.12.06.08	Gas permeability (in buffer/backfill)	SECONDARY FEP
2.1.12.07.00	Radioactive Gases in Waste and EBS	ANL-WIS-MD-000009
2.1.12.07.01	Radioactive gas (in waste and EBS)	SECONDARY FEP
2.1.12.07.02	Gaseous and volatile isotopes	SECONDARY FEP
2.1.12.08.00	Gas Explosions	ANL-WIS-MD-000009
2.1.12.08.01	H2/O2 explosions (in waste and EBS)	SECONDARY FEP
2.1.12.08.02	Flammability (in waste and EBS)	SECONDARY FEP

Table 3. Primary Waste-Form FEPs with Associated Secondary FEPs (Gray shading indicates primary miscellaneous waste-form FEPs and the associated secondary FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)
2.1.12.08.03	Explosions	SECONDARY FEP
2.1.12.08.04	Explosion	SECONDARY FEP
2.1.13.01.00	Radiolysis	ANL-WIS-MD-000009
2.1.13.01.01	Radiolysis (in waste and EBS)	SECONDARY FEP
2.1.13.01.02	Radiolysis	SECONDARY FEP
2.1.13.01.03	Radiolysis (in waste and EBS)	SECONDARY FEP
2.1.13.01.04	Radiolysis (in waste and EBS)	SECONDARY FEP
2.1.13.01.05	Radiolysis prior to wetting (in waste and EBS)	SECONDARY FEP
2.1.13.01.06	Radiolysis of brine	SECONDARY FEP
2.1.13.01.07	Radiolysis of cellulose (in waste and EBS)	SECONDARY FEP
2.1.13.01.08	Radiolysis	SECONDARY FEP
2.1.13.01.09	Radiolysis	SECONDARY FEP
2.1.13.02.00	Radiation Damage in Waste and EBS	ANL-EBS-MD-000015 ANL-WIS-MD-000009 ANL-WIS-MD-000010
2.1.13.02.01	Radiation effects (in waste and EBS)	SECONDARY FEP
2.1.13.02.02	Radiation effects on bentonite	SECONDARY FEP
2.1.13.02.03	Material property changes (due to radiation in waste and EBS)	SECONDARY FEP
2.1.13.02.04	Radiation damage (in waste and EBS)	SECONDARY FEP
2.1.13.02.05	Radiation shielding (in waste and EBS)	SECONDARY FEP
2.1.13.02.06	Radiation effects on buffer/backfill	SECONDARY FEP
2.1.13.02.07	Radiation effects on canister	SECONDARY FEP
2.1.13.02.08	Radiological effects on waste	SECONDARY FEP
2.1.13.02.09	Radiological effects on containers	SECONDARY FEP
2.1.13.02.10	Radiological effects on seals	SECONDARY FEP
2.1.13.02.11	Radiation effects on canister	SECONDARY FEP
2.1.13.03.00	Mutation	ANL-WIS-MD-000012
2.2.08.12.00	Use of J-13 Well Water as a Surrogate for Water Flowing into the EBS and Waste	ANL-WIS-MD-000009
3.1.01.01.00	Radioactive Decay and Ingrowth	ANL-WIS-MD-000009
3.1.01.01.01	Radioactive decay	SECONDARY FEP
3.1.01.01.02	Radioactive decay	SECONDARY FEP
3.1.01.01.03	Radioactive decay	SECONDARY FEP
3.1.01.01.04	Radioactive decay and ingrowth	SECONDARY FEP
3.1.01.01.05	Radioactive decay	SECONDARY FEP
3.1.01.01.06	Radioactive decay	SECONDARY FEP
3.1.01.01.07	Radioactive decay of mobile nuclides	SECONDARY FEP
3.1.01.01.08	Radionuclide decay and ingrowth	SECONDARY FEP
3.1.01.01.09	Radiological events and processes	SECONDARY FEP
3.2.07.01.00	Isotopic Dilution	ANL-WIS-MD-000009
3.2.07.01.01	Mass, isotopic and species dilution	SECONDARY FEP
3.2.07.01.02	Natural radionuclides/elements (in host rock disturbed zone)	SECONDARY FEP

1.4 ORGANIZATION OF FEP DATABASE

Under a separate scope of work, the YMP FEP team is constructing an electronic FEP database to assist the project during the license-review process, i.e., the YMP FEP database. Here, each FEP is entered as a separate record. Fields within each record provide a unique identification number, a description of the FEP, its origin, identification as a primary or secondary FEP, and mapping to related FEPs and to the assigned Analysis Model Reports (AMRs). Fields also provide summaries of the screening arguments, with references to supporting documentation, and, for all included FEPs, statements of disposition of the FEP within the Total System

Performance Assessment (TSPA). The FEPs discussed in this AMR provide summaries and dispositions for the 54 miscellaneous WF FEPs.

2. QUALITY ASSURANCE

An activity evaluation (CRWMS M&O 2000b, Enclosure 3), which was prepared per AP-2.21Q, *Quality Determinations and Planning for Scientific, Engineering, and Regulatory Compliance Activities*, determined that the Quality Assurance (QA) program applies to the activity under which this analysis was developed. This analysis does not affect any items on the *Q-List* (YMP 2000). Control of electronic management of data was accomplished in accordance with the controls specified by CRWMS M&O (2000b, Enclosure 5).

3. COMPUTER SOFTWARE AND MODEL USAGE

This AMR uses no computational software; therefore, this analysis is not subject to software controls. The analyses and arguments presented herein are based on guidance and proposed regulatory requirements, results of analyses presented and documented in other AMRs, or technical literature.

This AMR was developed using only commercially available software (Microsoft® Word 97 SR-2) for word processing, which is exempt from qualification requirements in accordance with AP-SI.1Q, *Software Management.* In addition, Sigma Plot, Scientific Graphic Software, Version 3.06, Jandel Corporation, and Microsoft Excel 97 SR-2 are commercial software packages used in this analysis to visually display the data using standard built-in mathematical functions. SigmaPlot, Scientific Graphic Software, Version 3.06, Jandel Corporation is used to plot data from the analysis. No calculations are performed with this software. There were no additional applications (Routines or Macros) developed using this commercial software. The values used and displayed are approximate in nature and are used only to identify a range of expected values. The FEPs in this AMR are scoping in nature and intended to provide the basis for the decision to either include or exclude the FEP. If a FEP is to be included in a model, any values or data is developed according to appropriate QA procedures in the applicable subject Summary or Model Abstraction AMR.

4. INPUTS

4.1 DATA AND PARAMETERS

The nature of the FEPs Screening Arguments and TSPA Dispositions is such that cited data and values are often used to support reasoned FEP Screening Arguments or TSPA Dispositions, rather than being used as direct inputs to computational analysis or models. Consequently, the data cited in FEPs Screening Arguments and TSPA Dispositions are largely corroborative in nature, and the FEP Screening Decisions will not be affected by any anticipated uncertainties in the cited data. As a result, the data are not listed as inputs in this section, but are cited in the individual FEP screening arguments and dispositions.

4.2 CRITERIA

This AMR complies with the NRC's proposed Rule 10 CFR Part 63 (64 FR 8640) and DOE's Interim Guidance (Dyer 1999). The subparts of the Interim Guidance that apply to this analysis are those pertaining to the characterization of the Yucca Mountain Site (Dyer 1999, Subpart B, Section 15). In particular, relevant parts of the guidance include the compilation of information regarding geology, hydrology, and geochemistry of the site (Dyer 1999, Subpart B, Section 21(c)(1)(ii)); the definition of geologic, hydrologic, and geochemical parameters; and conceptual models used in performance assessment (Dyer 1999, Subpart E, Section 114(a)).

Technical screening criteria are provided in DOE's Interim Guidance (Dyer 1999) and have also been proposed by the EPA in 40 CFR Part 197 (64 FR 46976). FEPs can be excluded from the TSPA if they are of low probability (less than one chance in 10,000 of occurring in 10,000 years ($10^{-4}/10^{4}$ years) or if occurrence of the FEP can be shown to have no significant effect on expected annual dose. There is no quantified definition of "significant effect" in the guidance or proposed regulations.

4.2.1 Low Probability

The probability criterion is explicitly stated in the DOE's Interim Guidance (Dyer 1999, Section 114(d)):

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

The EPA provides essentially the same criterion in proposed 40 CFR Part 197.40 (64 FR 46976):

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

The TSPAI IRSR (NRC 1998b) provides additional guidance on the screening process to follow in order to include or exclude FEPs. The guidance states that a low probability argument to exclude a FEP can be based on either quantitative estimates of the probability or qualitative arguments on the credibility of the FEP because of waste characteristics, repository design, or site characteristics.

4.2.2 Low Consequence

Criteria for low-consequence screening arguments are provided in DOE's Interim Guidance (Dyer 1999, Section 114(e) and (f)), which indicates that performance assessments shall:

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

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

The EPA provides essentially the same criteria in proposed 40 CFR Part 197.40 (64 FR 46976):

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

The terms "significantly changed" and "changed significantly" are undefined terms in the DOE Interim Guidance and in the EPA's proposed regulations. These terms are inferred for FEPs screening purposes to be equivalent to having negligible or no effect. Because the relevant performance measures differ for different FEPs (e.g., effects on performance can be measured in terms of changes in concentrations, flow rates, travel times, or other measures as well as overall expected annual dose), there is no single quantitative test of "significance."

4.2.3 Reference Biosphere

DOE's Interim Guidance (Dyer 1999), NRC's proposed regulations (10 CFR Part 63, 64 FR 8640), and EPA's proposed regulations (40 CFR §197.15, 64 FR 46976) specify assumptions (which in effect serve as criteria) pertinent to screening many of the Miscellaneous Waste-Form FEPs, such as explicit assumptions regarding the reference biosphere. An assumption pertaining to the characteristics of the reference biosphere is presented in DOE's Interim Guidance in Section 115 (a)(1) (Dyer 1999):

> Features, events, and processes that described the reference biosphere shall be consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site.

The EPA has specified a similar assumption in proposed 40 CFR §197.15 (64 FR 46976). This assumption can be summarized as follows:

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

These criteria are of particular interest because they impose a constraint on the use of probabilistic assessments to the TSPA.

4.2.4 Critical Group

The characteristics of the critical group to be used in exposure calculations are given in NRC's proposed regulations found at 10 CFR Part 63 (64 FR 8640) and DOE's Interim Guidance (Dyer 1999, Section 115(b)). Pertinent to the Miscellaneous Waste-Form FEPs is the guidance that:

> The critical group shall reside within a farming community located approximately 20 km south from the underground facility (in the general location of U.S. route 95 and Nevada Route 373, near Lathrop Wells, Nevada) (Dyer 1999, Section 115(b)(1))

The EPA-specified assumptions regarding biosphere characteristics are provided in proposed 40 CFR Part 197.21(a-c) (64 FR 46976) and describe the "reasonably maximally exposed individual" (RMEI). The characteristics of the RMEI are similar to those described for the critical group, but there is a significant difference in the approach of using a "critical group" versus the RMEI concept. The difference lies in the conceptual approach to calculating dose, the explanation of which is beyond the scope of this AMR.

4.3 CODES AND STANDARDS

There are no Codes or Standards directly applicable to this analysis.

5. ASSUMPTIONS

There are four general assumptions used throughout this AMR, either directly or indirectly, in screening the miscellaneous WF FEPs.

Assumption 5.1: The TSPA is based on an assumption that the potential repository will be constructed, operated, and closed according to the design used as the basis for the FEP screening.

Unless a FEP can be excluded because of a low probability of the phenomenon ever occurring, the FEP screening decision is based, at least in part, on the design used for the comparison. For example, the *License Application Design Selection Report* (CRWMS M&O 1999c, pp. 0-21 to 0-26, Section 7) indicates that the repository design includes backfill of the drift and installation of drip shields. These design features minimize the potential for rockfall or drift degradation damage to the waste packages, which could in turn affect the waste forms. The presence of these components strengthened the *Excluded* screening decision when REV 00 of this document was drafted. As stated in Section 1.1, the current design does not include backfill but this change does not affect the FEPs that feed the Waste Form Degradation Model.

This assumption is justified because a change in the design may require a reevaluation of the screening decision for FEPs that are dependent on design requirements.

Assumption 5.2: Only the wastes described in CRWMS M&O (2000j) were considered in these analyses. This assumption was based on management guidance (Stroupe 2000).

Assumption 5.3: The assumptions provided by proposed 10 CFR Part 63 (64 FR 8640) and the interim guidance (Dyer 1999) will be used within this analysis. The basis for using these assumptions is management guidance. These assumptions include the details of a human intrusion scenario, reference biosphere, and critical group.

Assumption 5.4: It has been assumed that FEPs may be excluded based on beneficial consequences. That is, if the only significant consequences of the FEP are improved repository performance and the potential repository performs adequately without taking credit for those benefits, then the TSPA will have bounded the performance as allowed by the interim guidance (Dyer 1999).

6. ANALYSIS/MODEL

This AMR addresses the 54 FEPs that have been identified as Miscellaneous Waste Form FEPs. Primarily, these FEPs represent areas of waste-form processes that could impact repository performance. The FEPs are related to geologic and hydrologic processes and contain detailed discussion. The FEPs discussions are arranged to make it easier to insert these FEPs into the FEP database.

The method used for this analysis is a combination of qualitative and quantitative screening of FEPs. The analyses are based on the criteria provided in the DOE's Interim Guidance (Dyer 1999), by the EPA in 40 CFR Part 197 (64 FR 46976), and by the NRC in proposed 10 CFR Part 63 (64 FR 8640). The criteria are used to determine whether or not each FEP should be included in the TSPA.

For FEPs that are *Excluded* based on proposed regulatory requirements (e.g., requirements regarding the location and composition of the critical group as described in Section 4.2.4), the screening argument includes the regulatory reference and a short discussion of the applicability of the standard. No primary miscellaneous waste-form FEPs have an *Excluded* screening decision based solely on proposed regulatory requirements or regulatory-specified conditions.

For FEPs that are *Excluded* based on NRC's proposed 10 CFR Part 63 (64 FR 8640) and DOE's Interim Guidance (Dyer 1999) or criteria from EPA's proposed regulations (40 CFR Part 197, 64 FR 46976), the Screening Argument includes the basis of the exclusion (low probability [Section 4.2.1] or low consequence [Section 4.2.2]) and provides a short summary. As appropriate, Screening Arguments cite work done outside this activity, such as in other AMRs. A more detailed discussion is typically provided in the Analysis/Discussion section.

For FEPs that are *Included* (the TSPA Disposition discussion for each FEP in Section 6.2 describes how the FEP has been incorporated in the process models or the TSPA abstraction.

For importance purposes as discussed in *Managing Technical Product Inputs*, AP-3.15Q, this analysis is classified as "Level 3" since it does not provide estimates of any of the Factors or Potentially Disruptive Events listed in the Screening Criteria for Grading of Data attachment in AP-3.15Q.

6.1 ALTERNATIVE APPROACHES

To ensure clear documentation of the treatment of potentially relevant future states of the system, the DOE has chosen to adopt a scenario-development process based on the methodology developed by Cranwell et al. (1990) for the NRC. The approach is fundamentally the same as that used in many performance assessments. The approach has also been used by the DOE for the Waste Isolation Pilot Plant (DOE 1996), by the NEA, and by other radioactive waste programs internationally (e.g., Skagius and Wingefors 1992). Regardless of the “scenario” method chosen for the performance assessment, the initial steps in the process involve development of a FEPs list, and screening of the FEPs list for inclusion or exclusion (see Section 1.2).

The approach used to identify, analyze, and screen the FEPs (as described in Section 1.2 and 1.3) was also considered. Alternative classification of FEPs as Primary or Secondary FEPs is possible in an almost infinite range of combinations. Classification into Primary and Secondary FEPs is based primarily on redundancy and on subject matter. Subsequent assignment and analysis by knowledgeable subject-matter experts for evaluation appeared to be the most efficient methodology for ensuring a comprehensive assessment of FEPs as they relate to the TSPA. Alternative classification and assignments of the FEPs are entirely possible but would still be based on subjective judgment. Alternative approaches for determining probabilities and consequences used as a basis for screening are discussed in Section 6.2 under the individual FEP analyses.

In practice, regulatory-type criteria were examined first, and then either probabilities or consequences were examined. However, there is no prescribed order in which the screening criteria should be applied. FEPs that are retained on one criterion were also considered against the others. Consequently, the application of the analyst's judgment regarding the order in which to apply the criteria does not affect the final decision. Allowing the analyst to choose the most appropriate order to apply the criteria prevents needless work, such as developing quantitative probability arguments for low-consequence events or complex, consequence models for low-probability events. For example, there is no need to develop detailed models of the response of waste packages to fault shearing, if it can be shown that fault-shearing events have a probability below the criteria threshold.

Regardless of the specific approach chosen to perform the screening, the screening process is in essence a comparison of the FEP against the criteria specified in Section 4.2. Consequently, the outcome of the screening is independent of the particular methodology or assignments selected to perform the screening.

Alternative interpretations of data as they pertain directly to the FEPs screening are provided in the Analysis and Discussion section for each FEP, as discussed below. The FEPs screening decisions may also rely on the results of analyses performed and documented as separate activities. Alternative approaches related to separate activities and analyses are addressed in the specific AMRs for those analyses and are not discussed in this AMR.

6.2 MISCELLANEOUS WASTE-FORM FEPS EVALUATION AND ANALYSIS

6.2.1 Magma Interacts with Waste—YMP No. 1.2.04.04.00

YMP Primary FEP Description: An igneous intrusion in the form of a dike occurs through the repository, intersecting the waste. This leads to accelerated waste-container failure (e.g., attack by magmatic volatiles (YMP No. 1.2.04.04.01), damage by fragmented magma (YMP No. 1.2.04.04.06), thermal effects (YMP No. 1.2.04.04.04), and dissolution of waste (YMP No. 1.2.04.04.02, 1.2.04.04.03).

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs: None.

IRSR Issues: TSPAI3, ENF3, CLST1, CLST3, CLST4

Screening Argument: The consequences of igneous activity were found (CRWMS M&O 1996) to contribute less than 1% to overall repository performance measures (to either releases or doses). However, the probability of occurrence of igneous activity is greater than the lower limit of one chance in 10,000 of occurring over 10,000 years adopted by DOE as the level of concern (Dyer 1999, Section 114(d)). Therefore, magma interaction with waste is considered as a disruptive event as discussed in the *TSPA Disposition* for this FEP.

Waste containers near magma intrusions will experience temperature increases as the magma intrudes and cools. Such increased temperature would accelerate corrosion and possibly cause cladding failure. Volatiles outgassing from the magma could alter the waste into a dispersible particulate form. While UO_2 is refractory, experiments have shown dissolution in molten basalt. Consequences from interaction with the fragmented magma are expected to be similar to rock fall damage (see YMP No. 2.1.07.01.00).

DSNF and HLW were evaluated with regard to their effect or contribution to the technical issue(s) discussed in this FEP. There was not any unique or significant effect not already accounted for by CSNF. The inventories of DOE-SNF are small enough that it probably would add very little to the total radioactivity released compared with commercial spent fuels (and HLW glass). The total mass of heavy metal for all DOE-SNF is less than about 3.5% of the total mass of heavy metal in the repository (or about 3.6% if HLW glass is ignored). In terms of activity, it is even less of an issue, with the activity of all DOE SNF contributing less than 0.5% of the radioactivity in the repository (with or without glass).

TSPA Disposition: Indirect volcanism was modeled (CRWMS M&O 1999h, Sections 3.9.2.1, 3.9.2.2) with the performance assessment numerical model using a modified source term, specifically different solubilities (because of different mineral phases) for some of the actinides. An example of an indirect volcanic effect is the contact of a magmatic dike with waste packages in which the waste form is recrystallized into other mineral phases that then dissolve at an increased or decreased rate in comparison to the undisturbed case.

Direct volcanic effects (i.e., radionuclides carried by ash plumes from volcanic eruptions) are modeled completely outside the performance assessment numerical model program using the code for contaminated ash dispersal and disposition (Jarzemba et al. 1997). For contaminated ash dispersal and disposition input requires a particle-size distribution for repository wastes following a volcanic event. Attachment I addresses this issue by reasonably concluding that the results of laboratory crushing and physical degradation due to a volcanic event are analogous. For further details on both direct and indirect volcanism modeling see CRWMS M&O (1999h, Section 3.9.2).

Supplemental Discussion: The probability of occurrence of igneous activity as estimated by the Probabilistic Volcanic Hazard Analysis (CRWMS M&O 1996, Section 4.3) is greater than the lower limit of 1 × 10^{-8} per year adopted by DOE as the level of concern (Dyer 1999, Section 114(d)). Therefore, this FEP must be considered as a possible disruptive event.

Volatiles outgassing (YMP No. 1.2.04.04.01) (Symonds et al. 1994, Chapter 1) from a magma dike or sill near to or through the repository can alter the waste into a dispersible particulate form. These volatiles exsolve from magma as it rises and as the confining pressure is reduced. The components are variable and include H_2O, H_2S, CO_2, SO_2, etc. The scenario develops as follows. (1) At temperatures of 700°C to 800°C fuel rods can fail by ductile rupture due to internal gas pressure (Guenther 1983, p. vii). (2) Oxidation of UO_2 to U_3O_8 can occur in tens of hours at temperatures as low as 305°C (Hanson 1998, Figures A.3 and A.16 to A.19). (3) With H_2O as an oxygen source, rapid oxidation of UO_2 fuel could be expected as soon as the cladding ruptures. The oxidized fuel is expected to be finely divided.

Dissolution of spent fuel in magma (YMP No. 1.2.04.04.02) can occur in spite of the fact that spent fuel is refractory and has a melting temperature 1000°C higher than the ambient magma temperature. It has been determined experimentally (Westrich 1982) that some substantial percentage of UO_2 will dissolve (form a multiphase eutectic) in molten basalt (used here as an approximation to the intrusive magma).

Waste containers that are in direct contact with magma (YMP No. 1.2.04.04.05) will experience a substantial temperature increase as the magma intrudes and cools. Because the magma is expected to be saturated with respect to Fe, dissolution of the containers is not expected. However, thermally induced structural collapse and cladding failure and other chemical interactions are possible (Guenther 1983, p. vii; Hanson 1998, p. 2.5).

Waste containers that are not in direct contact with magma (YMP No. 1.2.04.04.04) could still experience a substantial temperature increase because of proximity to the magma. Such increased temperature would accelerate corrosion and possibly cause cladding failure.

The inventories of DOE-SNF are small enough that it probably would add very little to the total radioactivity released compared with commercial spent fuels (and HLW glass) according to Table 1-1 in DOE (1998d). The total mass of heavy metal for all DOE-SNF is less than about 3.5% of the total mass of heavy metal in the repository (or about 3.6% if HLW glass is ignored). In terms of activity, it is even less of an issue with the activity of all DOE SNF contributing less than 0.5% of the radioactivity in the repository (with or without glass).

Attachments: See the following attachment in this AMR:

Attachment I, "An Estimate of Fuel-Particle Sizes for Physically Degraded Spent Fuel Following a Disruptive Volcanic Event Through the Repository"

6.2.2 Waste Inventory—YMP No. 2.1.01.01.00

YMP Primary FEP Description: The waste inventory includes all potential sources of radio toxicity and chemical toxicity. It consists of the radionuclide inventory (typically in units of curies), by specific isotope, of anticipated radionuclides in the waste, and the non-radionuclide inventory (typically in units of density or concentration) that consists of both physical, e.g., CSNF, DNSF, and HLW, and chemical waste constituents. The radionuclide composition of the waste will vary due to initial enrichment, burn-up, the number of fuel assemblies per container, and the decay time subsequent to discharge of the fuel from the reactor.

Also consider that the fuel types, matrices, radionuclide mixes, and non-radionuclide inventories in DSNF and HLW may differ from CSNF. Additional waste types should be considered if they are proposed for disposal at Yucca Mountain.

Screening Decision and Regulatory Basis: *Included* (those radionuclides that could potentially contribute to inhalation and ingestion doses or are relevant to groundwater protection).

Excluded based on low consequence (the effects of hazardous waste).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.01.00	DSNF Inventory and DSNF Alteration
2.1.02.07.00	Gap and Grain Release of Cs, I

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: The radionuclide inventory varies considerably both within and between each of the waste forms. Nearly 200 radionuclides are in these waste forms. Previous analyses indicate only a limited number of radionuclides are important to repository performance. Average and bounding inventories of 27 isotopes for 3 waste categories (CSNF, DSNF, HLW) have been developed: 24 isotopes from screening arguments based on human dose and 3 isotopes mandated by the Groundwater Protection Requirement of the proposed 40 CFR 197. Different subsets of isotopes are used for direct release, nominal release and human intrusion calculations.

Radionuclides that could potentially contribute to inhalation and ingestion doses or are relevant to groundwater protection are included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

Gap inventory, in general, is discussed in "Gap and Grain Release of Cs, I" FEP 2.1.02.07.00. Because the DSNF UO_2 waste form is not specifically modeled in the PA, gap inventory and release are subsumed in the conservatism of the DSNF inventory and DSNF alteration (see FEP 2.1.02.01.00.

The effects of hazardous wastes are excluded based on low consequence because they are limited by the Waste Acceptance System Requirements (DOE 1999a). The insignificant quantities of hazardous wastes that might be present will not significantly affect calculated expected annual dose.

TSPA Disposition: Examination of relative inhalation and ingestion doses from 100 to 10,000 years for average and bounding spent nuclear fuel and high-level waste leads to the following disposition in the TSPA-SR.

For a direct release from a disruptive event scenario, ^{90}Sr, ^{137}Cs, ^{227}Ac, ^{229}Th, ^{231}Pa, ^{232}U, ^{233}U, ^{234}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, and ^{243}Am are modeled. These are the isotopes that contribute most to the dose when release is not mitigated by either solubility or transport (CRWMS M&O 2000j).

Nominal release calculations include ^{14}C, ^{99}Tc, ^{129}I, ^{227}Ac, ^{229}Th, ^{232}U, ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am. By modeling the plutonium isotopes (^{238}Pu, ^{239}Pu, ^{240}Pu), the americium isotopes (^{241}Am, ^{243}Am), ^{229}Th, and ^{227}Ac, doses that could result from colloidal transport of radioactive material to the biosphere are adequately represented in the TSPA-SR. By modeling ^{14}C, ^{99}Tc, ^{129}I, the uranium isotopes (^{233}U, ^{234}U, ^{236}U, ^{238}U), and ^{237}Np, doses that could result from transport of solutes, either by fracture flow or matrix diffusion, are adequately represented in the TSPA-SR (CRWMS M&O 2000j).

Human-intrusion calculations should include ^{14}C, ^{99}Tc, ^{129}I, ^{227}Ac, $^{229\text{Th}}$, ^{232}U, ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am. The justification for this list of radionuclides is the same as that for nominal release discussed above. However, ^{63}Ni, ^{90}Sr and ^{137}Cs are included for the human-intrusion scenario because a human-intrusion event could occur as early as 100 years (CRWMS M&O 2000j). The recommended average per-package inventory for CSNF, DSNF, and HLW is shown in Table 4 (ANL-WIS-MD-000009 REV 00 ICN 01). The information in Table 4 is from Table 34 in CRWMS M&O (2000j).

Table 4. Recommended Average, per-Package Inventory

Isotope ID	Grams per CSNF Waste Package	Grams from DSNF in a Codisposal Waste Package	Grams from HLW in a Codisposal Waste Package
^{227}Ac	3.09E-06	1.05E-04	4.36E-04
^{241}Am	8.76E+03	7.87E+01	5.43E+01
^{243}Am	1.29E+03	1.68E+00	1.55E+00
^{14}C	1.37E+00	6.63E-01	7.11E-03
^{137}Cs	5.34E+03	5.52E+02	4.04E+02
^{129}I	1.80E+03	8.08E+01	4.41E+01
^{63}Ni	5.53E+01	6.48E-01	3.17E-01
^{237}Np	4.74E+03	4.26E+02	1.78E+02
^{231}Pa	9.87E-03	3.02E-01	7.44E-01
^{210}Pb	0.00E+00	1.38E-08	1.31E-07
^{238}Pu	1.51E+03	8.79E+01	5.69E+01
^{239}Pu	4.38E+04	2.13E+03	3.52E+03
^{240}Pu	2.09E+04	4.55E+02	3.39E+02
^{242}Pu	5.41E+03	1.15E+01	6.25E+00
^{226}Ra	0.00E+00	2.21E-06	1.52E-05
^{228}Ra	0.00E+00	6.46E-06	6.51E-06
^{90}Sr	2.24E+03	3.01E+02	2.67E+02
^{99}Tc	7.68E+03	4.53E+02	7.01E+02
^{229}Th	0.00E+00	2.46E-02	3.79E-03
^{230}Th	1.84E-01	1.75E-02	7.00E-03
^{232}Th	0.00E+00	1.38E+04	1.59E+04
^{232}U	1.01E-02	1.37E-01	7.64E-04
^{233}U	7.00E-02	1.98E+02	1.02E+01
^{234}U	1.83E+03	2.77E+02	3.39E+01
^{235}U	6.28E+04	1.74E+04	1.56E+03
^{236}U	3.92E+04	5.27E+03	3.65E+01
^{238}U	7.92E+06	4.67E+05	7.86E+05

Hazardous waste has been excluded from TSPA modeling as discussed in the *Screening Argument* for this FEP.

Supplemental Discussion: The radionuclide inventory varies considerably both within and between each of the waste forms. Nearly 200 radionuclides are in these waste forms. Previous analyses indicate only a limited number of radionuclides are important to repository performance (CRWMS M&O 2000j). The analytical method used to determine the relative importance of each radionuclide to the dose involved five steps. First, for the waste form under consideration, the relative dose contribution from an individual radionuclide was calculated. Second, the individual radionuclides were ranked with the highest contributor to the dose given the highest ranking. Third, each relative dose value calculated in the first step was converted to a percent contribution to the dose. Fourth, a cumulative sum of the percent dose contributions was calculated for each radionuclide in its ranked order. Finally, the fifth step was to choose radionuclides for the calculation (starting with the highest ranked radionuclide) to assure a reasonable estimate of the dose. For this screening analysis, the goal was to identify the radionuclides that contribute to 95% of the dose. (CRWMS M&O 2000j)

The analytical method described above identifies which radionuclides should be modeled if all of the radionuclides in a waste form are released to the environment in proportion to their inventory abundance. Thus, Input Transmittal for *Status of Radionuclide Screening for the Total System Performance Assessment - Site Recommendation (TSPA-SR)*, R&E-PA-99217.Tb (CRWMS M&O 1999i; 2000j) examined eight waste forms: an average Pressurized-water reactor (PWR) SNF assembly, a bounding PWR SNF assembly, an average Boiling-water reactor (BWR) SNF assembly, a bounding BWR SNF assembly, an average DSNF canister, a bounding DSNF canister, an average HLW canister, and a bounding HLW canister.

However, radionuclides are not generally released in proportion to their inventory abundance. Factors that can affect releases of radionuclides include radionuclide longevity, element solubility, and element transport affinity (CRWMS M&O 2000j). Consequently, screening on inventory alone is not bounding. The screening approach used involved grouping the radionuclides into subsets that have similar characteristics (radionuclide longevity, element solubility, and/or element transport affinity). Radionuclides that may be important given a variety of operative transport mechanisms are identified by grouping those radionuclides into subsets and screening each subset by itself.

The screening process was applied, using appropriate conditions, to obtain a relevant set of isotopes for each of three specific scenarios: direct release, nominal release, and human intrusion. For details see CRWMS M&O (1999i; 2000j).

Waste Package Radionuclide Inventory Approximations for TSPA-SR (CRWMS M&O 2000o) provides a description of how average radionuclide activities for each of the waste-package configurations were derived. While this input transmittal describes how the calculation of average radionuclide activities for the waste-package configurations was performed, the data from the transmittal have been superceded by the data from CRWMS M&O (2000j).

6.2.3 Codisposal/Collocation of Waste—YMP No. 2.1.01.02.00

YMP Primary FEP Description: Codisposal and collocation refer to the disposal of CSNF, DSNF, HLW, and possibly other wastes in close proximity within the repository. Codisposal and collocation might affect thermal outputs, chemical interactions, or radionuclide mobilization. At Yucca Mountain, the DSNF will be combined with HLW canisters within a waste package. This codisposal with HLW within a waste package is unique to the DSNF and does not apply to the CSNF placement within waste packages.

The DSNF will be contained within canisters that will be placed within the waste packages. The use of canisters within the waste package is not typical of the CSNF placement within waste packages. Also, some DSNF waste packages may contain only DSNF canisters, while others may contain both DSNF and HLW canisters.

Screening Decision and Regulatory Basis: *Included* (codisposal of DSNF and HLW glass, the effects of chemical interactivity between DSNF and HLW glass within a waste package on both DSNF degradation and radionuclide mobilization).

Excluded low consequence (chemical interactions between waste packages; the effects DSNF cladding [except for Naval SNF]; the effects of DSNF and glass-pour canisters as barriers to DSNF degradation, to HLW-glass dissolution, or to radionuclide release; DSNF geometry and surface area dependence; and preferential condensation).

Excluded low probability, (the possible effects of collocation with low- and intermediate-level nuclear waste, of toxic chemical wastes, and wastes disposed by future human activity in a nearby facility).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.01.03.00	Heterogeneity of waste forms
2.1.02.01.00	DSNF degradation
2.1.02.03.00	HLW degradation
2.1.02.25.00	DSNF Cladding Degradation.

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: Chemical interactivity between waste packages can be excluded from consideration based on low consequence to dose. Omission of chemical interaction between waste packages will not significantly affect calculated expected annual dose, because the unsaturated flow through the repository is vertical and the small but sufficient separation between adjacent waste packages implies only insignificant chemical interaction between waste packages. Insignificant chemical interactions between waste packages should not significantly affect the release of radionuclides. See *Supplemental Discussion* for details.

Preferential condensation can be excluded from consideration based on low consequence to dose. Omission of preferential condensation will not significantly affect calculated expected annual dose because thermal shielding by the near-field results in spatial averaging of the thermal field resulting in an environment that is not conducive to significant preferential condensation. See *Supplemental Discussion* for details.

No credit is taken for beneficial effects of the DSNF cladding (except for Naval SNF cladding). Omission of DSNF cladding will not significantly affect the interpretation of calculated expected annual dose and can, therefore, be excluded based on low consequence to dose. No credit is taken for the beneficial the effects of DSNF and glass-pour canisters as barriers to DSNF degradation, to HLW-glass dissolution, or to radionuclide release. Omission will not significantly affect the interpretation of calculated expected annual dose and can, therefore, be excluded based on low consequence to dose. DSNF geometry area dependence can be excluded based on low consequence to dose. The dose from uranium-metal fuel does not change as surface area is increased (CRWMS M&O 1998c, Section A-4). Therefore, omission of the geometry area dependence will not significantly change calculated expected annual dose. Also, the geochemical calculations of CRWMS (M&O 2000am, Section 6.4.3.1) indicate that, for several hundred years following waste-package failure, corrosion of the steel components of the waste package dominates the geochemistry resulting in a low-pH, acidic environment. In this environment, degradation of the SNF occurs within a few hundred years, which is small compared to regulatory time frame of 10,000 years.

The possible effects of co-location with low- and intermediate-level nuclear waste, toxic chemical wastes, and wastes disposed by future human activity in a nearby facility are moot and should be excluded from the TSPA based on low probability (not credible). These waste forms are not scheduled for disposal at the Yucca Mountain repository, and proposed 10 CFR Part 63 (64 FR 8640) and Dyer (1999) stipulates that future human behavior in the region should be considered to be the same as that occurring at the present time.

TSPA Disposition: The In-Package Chemistry Component in the TSPA-SR model accounts for chemistry effects due to the presence of both DSNF and HLW. See the in-package chemistry abstraction in CRWMS M&O (2000i). Degradation modeling of codisposed DSNF and HLW are treated separately within the Waste Form Degradation Component of the TSPA. Details of DSNF and HLW degradation can be found in CRWMS M&O (2000e; 2000d) respectively.

Chemical interactivity between waste packages, preferential condensation, DSNF cladding (except for Naval SNF cladding), and HLW pour canisters as a barrier to flow and transport have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: The principal waste forms to be disposed at Yucca Mountain are (CRWMS M&O 1998c, Section 6.1):

- Commercial spent nuclear fuel (CSNF) from boiling-water and pressurized-water reactors
- DOE-owned spent nuclear fuels (DSNF), dominated by N-reactor fuels from Hanford
- High-level radioactive waste (HLW) in the form of glass logs in stainless steel canisters.

These waste forms will be placed in specially designed waste packages with all of the HLW and some of the DSNF form designated for codisposal. Present plans call for five glass logs and one DSNF canister to be placed in each codisposal package (DOE 1999b, Section 9). Containing a relatively small fraction of the total waste, the codisposal packages will be co-located randomly within an array comprised predominantly of CSNF waste packages. This FEP and its secondary FEPs Included a number of issues relative to codisposal and collocation, issues with contrasting dispositions.

Collocation Issues—The collocation issues identified by this FEP may be excluded from the TSPA analyses.

Collocation with Other Waste Forms—Secondary FEPs 2.1.01.02.01 and 2.1.02.02.03 focus on the possible effects of collocation with waste forms other than those listed above.

Specifically, these FEPs refer to low- and intermediate-level nuclear waste, to toxic chemical wastes, and wastes disposed by future human activity in a nearby facility. It should be noted that wastes other than those listed above are not scheduled for disposal at the Yucca Mountain repository, and that proposed 10 CFR Part 63 (64 FR 8640) and Dyer (1999) stipulates that future human behavior in the region should be considered to be the same as that occurring at the present time. These issues therefore are moot and are excluded from the TSPA based on low probability (not credible).

Chemical Interactions between Waste Packages—FEPs 2.1.01.02.02, 2.1.02.10.00, and 2.1.02.11.00 identify the possibility of chemical interactions between neighboring waste packages. However, the nature of flow through fractured, unsaturated host rock and a small, but sufficient, separation of neighboring waste packages mean that chemical interactions between waste packages are not expected. Interactions are not expected because unsaturated flow is largely vertical and packages in the same disposal elevation would not interact geochemically. Hence, changes in the rates of waste-package degradation and radionuclide mobilization are not expected. These arguments imply only insignificant chemical interaction between waste packages, and the issue of chemical interactivity between waste packages is, therefore, excluded based on low consequence to dose.

Thermal Interactions between Waste Packages—FEP 2.1.01.02.09 notes that thermal interactions between neighboring waste packages can occur. This issue alludes to the fact that although commercial design seeks to minimize temperature differences, the DSNF codisposal and HLW waste packages are expected to be generally cooler (CRWMS M&O 2000aa and CRWMS M&O 2000ab) than CSNF waste packages at the time of disposal (CRWMS M&O 1998b, Table 3-22). Consequently, the codisposal packages may experience preferential condensation relative to the CSNF packages. In developing an argument for this issue, it is useful to identify a characteristic length. The waste packages in the VA design were considered to be placed in an in-line end-to-end placement of waste packages with a relatively small separation between waste packages. This means that a waste-package length of ~ 5.3 m (CRWMS M&O 1998d, Table 5-1) represents an approximate center-to-center separation distance, which also represents an appropriate characteristic length for inferring changes in the thermal field within and surrounding the repository. At distances greater than a few waste-package lengths, the thermal field reflects the presence of an average heat source rather than the presence of an individual waste package. This averaging effect tends to "hide" the exact location of the cooler packages and to shield them from preferential condensation.

The period of concern only occurs if the repository cools to temperature levels that are lower than those of surrounding rock. At that time the vapor-pressure gradient reverses direction and turns vapor transport toward the repository. Determining whether such a gradient reversal can occur at all is a rather complex problem because it depends on both the power-level, the rate of decay of the waste within the repository, and the thermal properties of surrounding rock. If a reversal does occur, the repository then becomes a target for vapor transport and condensation. It should be noted that relative waste-package temperatures at the time of gradient reversal, typically several hundred years into the post-operational period, represent the most appropriate basis for assessing preferential condensation. The fact that DSNF and codisposal packages are cooler at the time of disposal does not mean that they will be cooler several hundred years later.

This discussion takes the position that preferential condensation may be excluded from TSPA analyses. There are two reasons. First, thermal shielding caused by the near-field averaging of the thermal field renders preferential condensation a process of low consequence. Second, uncertainty regarding the occurrence of repository condensation and added uncertainty regarding the occurrence of preferential condensation render preferential condensation a process of low probability. This issue becomes even less important with the proposed line loading of the EDA II design (CRWMS M&O 1999c, Table 5-3) which is designed to keep the waste packages at nearly equal temperatures. Therefore, preferential condensation is screened out, based on both low consequences to dose and low probability (not credible).

Codisposal Issues—In contrast to collocation issues, one of the codisposal issues identified by this FEP that must be included in TSPA analyses is the geochemistry in the failed waste package. This issue must be either included or bounded by the TSPA.

DSNF/HLW Glass Interactions within a Codisposal Waste Package—Secondary FEPs 2.1.01.02.04, 2.1.01.02.07, 2.1.01.02.10, 2.1.01.02.11, and 2.1.01.02.12 identify the likelihood of chemical reactions within a codisposal package and note that such reactions may affect DSNF degradation, radionuclide mobilization, and the integrity of DSNF cladding. To respond to such concerns, the National Spent Nuclear Fuel Program has sponsored analyses (CRWMS M&O 2000am, Section 6.4) using the geochemical model EQ3/6 (Wolery 1992, general reference to entire document).

These calculations indicate that, for several hundred years following package failure, corrosion of the steel internal components of waste package dominates the geochemistry, resulting in a low-pH, acidic environment. Here, the steel components of the codisposal package, which holds the HLW glass-pour canisters and DSNF in place, cause the low-pH environment by corrosion. This environment is much like what one would expect in the absence of glass. After the steel is exhausted, glass dissolution dominates the geochemistry, resulting in a high-pH, alkaline environment. HLW glass then controls the geochemical environment in which the DSNF radionuclides are dissolved and released (CRWMS M&O 2000am, Section 6.4.3.1).

Various parameters affect the geochemical environment, sometimes substantially altering the rates of release for uranium, plutonium, and neptunium. Such parameters include the corrosion rate of low-carbon steel, the dissolution rate of HLW glass, the oxygen fugacity, and the DSNF mass loading. Note: This is the DSNF and other waste degradation model (CRWMS M&O 2000e). Thus, chemical interactivity between DSNF and HLW glass affects both DSNF degradation and radionuclide mobilization and must be accounted for in TSPA analyses.

Most likely, such chemical interactivity would negatively impact the integrity of DSNF cladding, and the geochemical calculations include cladding degradation in determining the geochemical environment of the codisposal package. Here, however, a special consideration takes precedence. No credit is taken for DSNF cladding (except for Naval SNF) either as a barrier to chemical interactivity or as a barrier to radionuclide release (CRWMS M&O 1998c, Section A-3.3); consequently, such effects may be excluded from TSPA analyses based on low consequence because there would be no significant increase in dose.

DSNF and Glass-Pour Canisters—Stainless steel canisters facilitate the transport and handling of both DSNF and HLW glass, and the geochemical calculations discussed above include their dissolution in determining the geochemical environment of the codisposal package. For the analyses of DSNF (CRWMS M&O 1998c, Section 6.5.1) however, like DSNF cladding, no credit is taken for any beneficial effects of these canisters as barriers to DSNF degradation, to HLW-glass dissolution, or to radionuclide release. Consequently, such effects may be excluded from TSPA analyses based on low consequence because there would be no significant increase in dose.

DSNF Geometry—Secondary FEP 2.1.01.02.08 notes that the DSNF to be disposed of in Yucca Mountain will have a variety of fuel geometries, some of which may be significantly different from the commercial SNF geometries. The concern here is the variability of surface area, its possible effect on radionuclide release, and the need to include this effect in TSPA analyses.

Generally, however, the dependence of radionuclide release on surface area is small, and the consequences of ignoring the effect are also small. The dose from uranium-metal fuel does not change, as surface area is increased (CRWMS M&O 1998c, Section A-4). The geochemical calculations of CRWMS (M&O 2000am, Section 6.4.3.1) indicate that, for several hundred years following package failure, corrosion of the steel components of the waste package dominates the geochemistry, resulting in a low-pH, acidic environment. In this environment, degradation of the SNF occurs within a few hundred years.

Thus, although congruent degradation, together with an area dependence, most likely occurs, the time period is so short (a few hundred years) compared with the regulatory period (10,000 years) that this area dependence is of little consequence and may be ignored by the calculations. For ^{129}I and ^{99}Tc, two dose-sensitive fission products, total release occurs sufficiently rapidly following degradation that it may be considered as instantaneous by TSPA analyses. For ^{237}Np, an actinide with a strong sensitivity to dose during a time frame of a few hundreds of thousands of years, limited solubility is the dominant constraint on release, not surface area.

For uranium-metal fuel, e.g., N-Reactor fuel, the area-to-volume ratio increases relatively rapidly because of the manner in which the fuel degrades to UO_2 particulates in the presence of water (CRWMS M&O 2000e, Section 6.3.7). Here, again, degradation time is relatively short (tens of years), and, as in the case of Pu, release is dominantly limited by solubility and by the formation of minerals (CRWMS M&O 2000am, Section 6.4.3.2). Therefore, DSNF geometry is excluded from the TSPA-SR due to low consequence to dose.

6.2.4 Heterogeneity of Waste Forms—YMP No. 2.1.01.03.00

YMP Primary FEP Description: Waste forms to be placed in Yucca Mountain will have physical, chemical, and radiological properties that vary.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.01.04.00	Heterogeneity of Emplaced Waste
2.1.02.01.00	DSNF Degradation
2.1.02.02.00	CSNF Alteration, Dissolution, and Radionuclide Release

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Commercial spent nuclear fuel (CSNF), DOE-owned spent nuclear fuel (DSNF), and high-level waste (HLW) shipped to the repository may contain quantities of radionuclides that vary from waste package to waste package, fuel assembly to fuel assembly, and from canister to canister. The composition of each of these waste forms may vary due to initial uranium enrichment, possible plutonium enrichment, and burnup of the fuel, among other factors. The physical state within the waste form may also vary. For example, damaged fuel pellets or extremely high-burnup fuels may have greater surface area exposed to any water penetrating a waste package than undamaged, low-burnup spent fuel. Given these potential differences in isotopic composition and physical condition, the mass of radionuclides available for transport may vary significantly among waste packages.

The different physical (structure, geometry), chemical, and radiological properties of the many forms CSNF, DSNF, and HLW could result in differences in the corrosion and alteration rates of the waste-package composition. This could affect repository chemistry, breach times, dissolution rates, and availability of radionuclides for transport. However, the behavior of the repository is not expected to be significantly influenced by the presence of the DSNF because the DSNF represents only ~10%, by activity, of the total waste to be stored at Yucca Mountain.

Heterogeneity of waste forms is included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: Three major categories of waste forms are currently planned for disposal at Yucca Mountain: CSNF, DSNF, and HLW. These will be broken down into a various number of configurations based on properties of the distinct types and codisposal options as appropriate for determining average and bounding inventories and representative waste package geometry.

Coefficients of corrosion-rate equations, specific areas, inventory, cladding-failure rate and the number of packages for each waste-form category are the primary TSPA modeling parameters (CRWMS M&O 1999h, Table 3.10-1).

CSNF—CSNF (CRWMS M&O 2000c) consists of two categories: pressurized water reactors (PWR) and boiling water reactors (BWR). Radionuclide activities for each assembly in the waste stream were estimated, and the waste-package configuration that could accommodate each assembly based on its potential criticality level was determined. The result was a grouping of the 230,000 CSNF assemblies into five proposed waste-package configurations. For each group, average and bounding radionuclide activities for the fuel assemblies were calculated.

The five proposed CSNF waste package configurations are (CRWMS M&O 2000j, Table I-1):

Configuration	Number of Packages	Number of Fuel Assemblies PWR	BWR	Length Classification	Criticality Control
21 PWR	4500	21	0	regular	Absorber Plate
21 PWR	100	21	0	regular	Control Rod
12 PWR	170	12	0	long	Absorber Plate
44 BWR	3000	0	44	regular	Absorber Plate
24 BWR	90	0	24	regular	Absorber Plate (thick)

Note: MCO–Multi-Canister Over-pack. "Short," "long," and "regular" are waste-container length classifications and do not necessarily represent a specific length.

The average CSNF radionuclide activity for one of these waste packages is the number of fuel assemblies times the average per-assembly radionuclide activity. Details are provided in CRWMS M&O (2000j, Attachment I).

The effects of heterogeneity on CSNF degradation can be found in FEP 2.1.02.02.00 and on waste inventory in FEP 2.1.01.01.00.

DSNF and HLW—For the most part, DSNF and HLW (CRWMS M&O 2000e; 2000d) are planned to be disposed together in codisposal packages. The nuclear-fuel-inventory information from DOE (1999b) was used to calculate average and bounding per-canister inventories for eight proposed waste-package configurations (CRWMS M&O 2000j, Table I-1):

Configuration	Number of Packages	Canisters-per-Package DSNF	HLW
Codisposal Short	1100	1 short	5 short
Codisposal Long	1500	1 long	5 long
DSNF Short/HLW Long	130	1 short	5 long
HLW Long	600	0	5 long
N-Reactor	160	2 MCO	2 long
Naval Long	110	1 Naval long	0
Naval Short	210	1 Naval short	0
Immobilized Pu	100	0	5

Note: MCO–Multi-Canister Over-pack. "Short" and "long" are waste-container length classifications and do not necessarily represent a specific length.

The average DSNF radionuclide activity for one of these waste-packages is the number of DSNF canisters times the average per-canister radionuclide activity. The average HLW radionuclide activity from for one of these waste-packages is the number of HLW canisters times the average per-canister radionuclide activity. Details are provided in CRWMS M&O (2000j, Attachment I).

The effects of heterogeneity on DSNF degradation can be found in FEP 2.1.02.01.00; HLW Glass degradation in FEP 2.1.02.03.00; and on waste inventory in FEP 2.1.01.01.00.

For the in-package chemistry abstraction, see CRWMS M&O (2000i); for the inventory abstraction, see CRWMS M&O (2000j), and for the cladding degradation abstraction, see CRWMS M&O (2000w).

Supplemental Discussion: CSNF, DSNF, and HLW shipped to the Yucca Mountain repository may contain quantities of radionuclides that vary from fuel assembly to fuel assembly, from canister to canister and from waste-package to waste-package. The composition of each of these waste forms (waste-package fuel type, fuel assembly or canister) may vary due to initial uranium enrichment, possible plutonium enrichment, and burnup of the fuel, among other factors. The physical state within the waste form may also vary. For example, damaged fuel pellets or extremely high-burnup fuels may have greater surface area exposed to any water penetrating a waste package than undamaged, low-burnup spent fuel. Given these potential differences in isotopic composition and physical condition, the mass of radionuclides available for transport may vary significantly among the waste packages.

The different physical (structure, geometry), chemical, and radiological properties of the many forms of CSNF, DSNF, and HLW could result in differences in the corrosion and alteration rates of the waste-package elements and matrices. This could affect repository chemistry, breach times, dissolution rates, and availability of radionuclides for transport. However, the behavior of the repository is not expected to be significantly influenced by the presence of the DOE SNF because the DOE SNF represents such a small fraction, ~10% by radionuclide activity, of the total waste to be stored at Yucca Mountain (CRWMS M&O 2000j, Tables I-5 and I-6).

Heterogeneity in the physical (structure, geometry), chemical, and radiological properties of waste packages will be included, in some form, in this TSPA. At a minimum, waste categories of CSNF, DSNF, and HLW will be modeled separately. Though within a category (CSNF, DSNF, HLW), the variations may not be considered in detail except in determining the representative, average, or surrogate properties of the category.

The argument excluding waste-form heterogeneity based on the random placement of waste package (see FEP 2.1.01.04.00) also is a supporting argument for using representative, averaged, or surrogate properties for each category of waste (CSNF, DSNF, HLW).

Commercial Spent Nuclear Fuel—The CSNF consists of two categories: pressurized water reactors (PWR) and boiling water reactors, (BWR). The size of the fuel rods and assemblies differ. Therefore, the amount of decay heat produced by disposal containers holding PWR or BWR SNF also differs. Most of the fuel within these two categories is clad with Zircaloy. However, approximately 1% is old fuel that is clad with stainless steel.

CSNF is the dominant type of waste that will be emplaced at the Yucca Mountain repository. A total of about 63,000 MTHM of CSNF, comprising around 90% of the total inventory, will be emplaced in 7,860 waste packages and modeled using 5 representative configurations (CRWMS M&O 2000j, Table I-1). The CSNF will be shipped to the repository from multiple commercial sites and are expected to arrive at the Yucca Mountain repository in random order with regard to radioactivity level, damage, and other factors influencing heterogeneity of the waste.

High-Level Waste—The HLW, incorporated in a solid solution with borosilicate glass, is processed at various sites, and, therefore, the radioisotope inventory will vary slightly depending on the point of origin. The HLW comprises less than 7% of the total waste on a radionuclide activity basis (CRWMS M&O 2000j, Tables I-5 and I-6). The HLW will be in the form of glass logs encased in stainless steel canisters from DOE facilities at Savannah River, Hanford, West Valley, and the Idaho National Engineering and Environmental Laboratory. The HLW canisters are expected to arrive in random order with regard to radioactivity level and other factors influencing heterogeneity of the waste.

The HLW canisters for the base case will be emplaced in the 3910 codisposal waste packages and modeled using five representative configurations. Waste packages containing canisters filled with immobilized Pu-in-ceramic will comprise 100 waste-packages and will be modeled using 1 representative configuration. (CRWMS M&O 2000j, Table I-1).

DOE Spent Nuclear Fuel—In general, the more than 250 forms of DSNF have been reduced to a tractable number of representative types and packaging configurations. For example, the TSPA-VA (CRWMS M&O 1998c, Section 6.3.2.1) considered DSNF as being represented by five waste-forms: metal, oxide, carbide, ceramic, and glass. Spent Navy reactor fuel, and both spent mixed U-Pu oxide (MOX) fuel and immobilized Pu from the disposition of excess weapons fissile material were considered in separate analyses. The variety of waste-from structures (fuel assemblies for example) will also be taken into account in determining the representative types.

The DSNF comprises less than 4% of the total waste mass on a radionuclide activity basis and has been grouped into 11 categories with similar physical characteristics (CRWMS M&O 2000e, Section 1). One fuel type, Uranium metal, comprises most (~85%) of the DSNF (CRWMS M&O 2000e, Table 3). Zirconium clad N-Reactor constitutes over 95% of this category. Although N-Reactor is Zirconium clad, the cladding will not be included in the TSPA because it is generally in poor condition. The N-Reactor waste will be co-disposed with HLW in 160 waste-packages and modeled using 1 representative configuration (CRWMS M&O 2000j, Table I-1). Also, the TSPA will use the degradation behavior of N-Reactor fuel matrix to conservatively represent the degradation of all DSNF (CRWMS M&O 2000e, Section 7.1).

The remaining DSNF inventory will be co-disposed with HLW in 2730 waste-packages (CRWMS M&O 2000j, Table I-1) and modeled using 3 representative configurations. Naval fuel will be disposed of in 320 waste-packages and modeled using 2 representative configurations. The degradation behavior of the Naval SNF will be conservatively bounded by the Zirconium clad CSNF degradation model (CRWMS M&O 2000e, Section 6.3.1).

6.2.5 Spatial Heterogeneity of Emplaced Waste—YMP No. 2.1.01.04.00

YMP Primary FEP Description: Waste placed in Yucca Mountain will have physical, chemical, and radiological properties that will vary spatially.

CSNF, SNF, and HLW shipped to the repository may contain quantities of radionuclides that vary from fuel assembly to fuel assembly, from waste package to waste package, and from canister to canister. The composition of each of these waste forms (fuel assembly or canister) may vary due to initial uranium enrichment, possible plutonium enrichment, and burnup of the fuel, among other factors. The physical state within the waste form may also vary. For example, damaged fuel pellets or extremely high-burnup fuels may have greater surface area exposed to any water penetrating a waste package than undamaged, low-burnup spent fuel. Given these potential differences in isotopic composition and physical condition, the mass of radionuclides available for transport may vary by several orders of magnitude among waste packages.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence (the effect of spatial heterogeneity of the waste on repository-scale response. No secondary FEPs associated with this primary FEP).

Included (heterogeneity within a waste package is implicitly included in the evaluation of in-package temperature used to determine perforation of the CSNF cladding).

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.01.02.00	Codisposal/Collocation
2.1.01.03.00	Heterogeneity of waste forms

IRSR Issues: TSPAI3, CLST3

Screening Argument: The time scale for convective transport and diffusion of aqueous species in the fully flooded waste package is rapid in relation to alteration of the contents (waste form, cladding, and support material); consequently, the contents are modeled as being uniformly mixed and spatial heterogeneity in the waste is excluded. Spatial heterogeneity of the two types of waste packages modeled is also excluded based on low consequence. Waste packages of CSNF and codisposed DSNF and HLW are to be closely packed together so that variations in temperature (which can influence aspects of waste-form degradation and in-package chemistry) of individual waste packages will not vary significantly. The effect of spatial heterogeneity of the waste on repository-scale response can be excluded based on low consequence. Omission of repository-scale heterogeneity from the TSPA will not significantly affect the calculated, expected annual dose because random placement and close packing of the CSNF and codisposal waste packages will result in spatial averaging of temperatures and release of radionuclides.

Heterogeneity within a waste package is implicitly included in the TSPA as discussed in the *TSPA Disposition* section.

TSPA Disposition: Spatial heterogeneity within waste packages is implicitly included in the evaluation of temperature within a waste package used to determine perforation of the CSNF cladding. For details, see the cladding-degradation abstraction in CRWMS M&O (2000w).

The effect of spatial heterogeneity of the waste on repository-scale response has been excluded from consideration in the TSPA as discussed in the *Screening Argument* for this FEP.

Supplemental Discussion:

Commercial Spent Nuclear Fuel—CSNF is the dominant type of high-level waste that will be emplaced at the Yucca Mountain repository. A total of 63,000 metric tons heavy metal (MTHM) of CSNF, comprising 90% of the total inventory, will be emplaced in 7,860 waste packages (CRWMS M&O 2000j, Tables I-1). The CSNF will be shipped to the repository from multiple commercial sites. It is reasonably concluded that it will arrive randomly with regard to radioactivity level, fuel damage, and other factors influencing heterogeneity of the spent fuel as a source of radionuclides.

On the basis of this conclusion, packages of CSNF will be emplaced in such a way that any heterogeneity in radionuclide release from package-to-package variation will be spatially averaged by random placement of waste packages. Waste packages of CSNF and codisposed DSNF and HLW are to be closely packed together so that variations in temperature (which can influence aspects of waste-form degradation and in-package chemistry) of individual waste packages will not vary significantly. However, temperature variations within a waste package are evaluated to determine perforation through creep rupture (cladding FEP 2.1.02.19.00) and stress corrosion cracking. Any heterogeneity that remains after spatial averaging is expected to have a negligible impact on total system performance.

Within a waste package, the time scale for convective transport and diffusion of aqueous species in the fully flooded waste package is rapid in relation to the alteration of the contents (waste form, cladding, and support material). Consequently, the contents are considered to be uniformly mixed, and spatial heterogeneity of the waste relative to the evaluation of in-package chemistry can be excluded. Spatial heterogeneity of the two types of waste packages modeled is also excluded based on low consequence. Spatial averaging will be effective in eliminating heterogeneity when multiple waste packages fail. It will not be effective if only one package fails. However, a single package failure of CSNF produces negligible release as demonstrated by the recent defense-in-depth calculations for the license application (LA) design (CRWMS M&O 1999f). The defense-in-depth calculation for the LA design shows that peak dose rate from a single juvenile waste-package failure is 0.001 mrem/year for the first 20,000 years and 0.005 mrem/year for the first 100,000 years after closure (CRWMS M&O 1999f, Figure 1). Even a hypothetical increase in release rate by two orders of magnitude results in peak dose rates of 0.1 and 0.5 mrem/year, substantially below the anticipated 15 mrem/yr regulatory limits for the repository, the EPS's proposed standard for Yucca Mountain (40 CFR Part 197.20) (64 FR 4976). Hence, the potential heterogeneity due to failure of a single waste package of CSNF can be ignored in performance assessment.

High-Level Waste—The Yucca Mountain repository is designed to hold 4,667 MTHM of HLW, and the HLW comprises approximately 6.7% of the total waste on a MTHM basis (CRWMS M&O 1998c, Section 6.2.1). The HLW will be in the form of glass logs encased in stainless steel canisters from DOE facilities at Savannah River, Hanford, and West Valley and from the Idaho National Engineering and Environmental Laboratory (DOE 1999b). It is reasonably concluded that the HLW canisters will arrive randomly with regard to radioactivity level, damage, and other factors influencing heterogeneity of the vitrified waste as a source of radionuclides.

The canisters will be emplaced in 1,663 waste packages (CRWMS M&O 1998c, Section 6.2.1.3). Each package will hold four canisters of HLW and one canister of DSNF. The waste packages of DSNF will be emplaced based on their random distribution on arrival. Any heterogeneity in radionuclide release will be spatially averaged by random placement of waste packages and further averaged by the fact that four canisters of HLW are collocated in each waste package. Any heterogeneity that remains is expected to have a negligible impact on total system performance.

DOE Spent Nuclear Fuel—The YMP repository is designed to hold 2,333 MTHM of DSNF (CRWMS M&O 1998c, Section 6.2.1). The DSNF comprises 3.3% of the total waste on a MTHM basis and has been grouped into 11 groups with similar physical characteristics (DOE 1999b, pp. 8, 9). The categories of DSNF and its placement in waste packages provide a basis for ignoring heterogeneity of DSNF. The uranium metal group comprises most (85%) of the DSNF on an MTHM basis (DOE 1999b, pp. 8, 9). This waste will be contained in 101 waste packages that are to be emplaced randomly in the repository (CRWMS M&O 1998c, Table A6-10). Most packages (95) will contain four canisters of DSNF. While container-to-container variations will occur, placing four canisters in each waste package and random placement of these packages throughout the repository will tend to spatially average any effects from container-scale heterogeneity.

6.2.6 DSNF Degradation, Alteration, and Dissolution—YMP No. 2.1.02.01.00

YMP Primary FEP Description: DSNF to be disposed of in Yucca Mountain has a variety of fuel types that include metallic uranium fuels; oxide and MOX fuels; Three Mile Island rubble; and heterogeneous fuels such as UAl_x, $U\text{-}ZrH_x$, and graphite fuels. In general, the composition and structure of these spent fuels are significantly different from the commercial spent nuclear fuel (CSNF), and degradation, alteration, and dissolution may be different from the CSNF degradation.

Processes to be considered in this FEP include alteration and dissolution of the various DSNF waste forms, phase separation, oxidation of spent fuels, selective leaching, and the effects of the high-integrity can (HIC) on DSNF degradation.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.01.02.00	Codisposal/Collocation
2.1.01.03.00	Heterogeneity of waste forms
2.1.02.28.00	Various features of the ~250 DSNF fuel types and grouping into categories
2.1.09.04.0	Radionuclide solubility

IRSR Issues: TSPAI3, ENF3, CLST1, CLST3, CLST4

Screening Argument: The DSNF inventory is diverse and includes metallic uranium fuels; oxide and MOX fuels; Three Mile Island rubble; and heterogeneous dispersion fuels such as UAl_x, $U\text{-}ZrH_x$, and graphite fuels. The diversity of the DSNF is such that degradation will differ from that of the CSNF (Eide 2000). Furthermore, the YMP will allocate a portion of the maximum allowable mass in the repository to the disposal of DSNF. For these two reasons, diversity of degradation rate and specified allocation, the DSNF is included in the TSPA.

DSNF degradation, alteration, and dissolution are included in the TSPA as discussed in the *TSPA Disposition*.

TSPA Disposition: The degradation rate of DSNF was set at a fixed, bounding rate that applied to all DSNF types. (The DSNF inventory, however, is a composite of all ~250 types of DSNF [including Naval fuels–see Waste Inventory FEP, 2.1.01.01.00; furthermore, the Pu ceramic waste form is included in the TSPA by including its inventory in the HLW category (CRWMS M&O 2000e, Section 6.3.12)]). The degradation rate was based on an evaluation of available corrosion data for uranium metal, uranium oxide, and the experimental oxidation data obtained recently for irradiated N-Reactor fuel. Ten times the experimentally obtained oxidation data for N-Reactor Uranium metal (which makes up 90% of DSNF category) provides a conservative bound for the other available alteration data as shown in Figure 2. Ten times the highest observed rate in recent Pacific Northwest National Laboratory (PNNL) studies is 1.75×10^6 mg/m^2-d (CRWMS M&O 2000e, Section 6.3.7).

Radionuclides are mobilized congruently in proportion to the fraction of waste altered. The fraction of waste altered is a function of the surface area available for degradation. This available surface area is conservatively considered to remain constant during the degradation process.

The concentration of each radionuclide is calculated based on the mass available and the minimum of rind (altered zone) (CRWMS M&O 2000p) (see FEP 2.1.09.12.00), water volume, and seepage volume. The concentration of each radionuclide is limited by the specific radionuclide solubility limit (see FEP 2.1.09.04.00) (CRWMS M&O 2000y, Section 7). Colloid formation is also considered in the TSPA model (see Colloid FEP 2.1.09.14.00).

Supplemental Discussion: The DSNF inventory is diverse and includes metallic uranium fuels; oxide and MOX fuels; TMI rubble; and heterogeneous dispersion fuels such as UAlx, U-ZrHx, and graphite fuels. Groups 2 through 11 (DOE 1999b) of the DSNF will be conservatively represented by N-Reactor Uranium metal (which makes up 90% of this category) with a composite inventory derived from all DSNF Groups (except Pu ceramic and Group 1 Naval fuels). Aqueous corrosion of the metallic uranium-based SNF produces particulate materials (Gray and Einziger 1998) and, thereby, potentially enhances radionuclide release.

Current TSPA Bounding of DSNF—CRWMS M&O (2000e, Section 7.2.1) recommends that for TSPA purposes, the N-Reactor SNF be used as the surrogate for radionuclide release kinetics for all DSNF since the rate of degradation of this SNF is higher than that of other DSNF waste forms (CRWMS 2000e). This approach thus conservatively bounds the effects of oxidation for all DSNF. Therefore, the TSPA analyses, which employ a constant rate degradation model, will use the recommended rate of 1.75 x 10^6 mg/m^2-d for N-Reactor fuel as a bounding rate for all DSNF (CRWMS 2000e). The recommended degradation rate was established by an evaluation of the available corrosion data for uranium, uranium oxide, and other metals and materials that comprise the DSNF, and the experimental oxidation data obtained recently for *irradiated* N-Reactor fuel (CRWMS M&O 2000e). Presently, the Release Rate Testing Program, managed by the NSNFP, is experimentally determining the corrosion rate and solubility of N-Reactor and other irradiated DSNF fuel types under conditions more representative of the anticipated repository conditions. These results will be incorporated into the TSPA model to confirm the bounding of the DSNF, including the heterogeneous fuels.

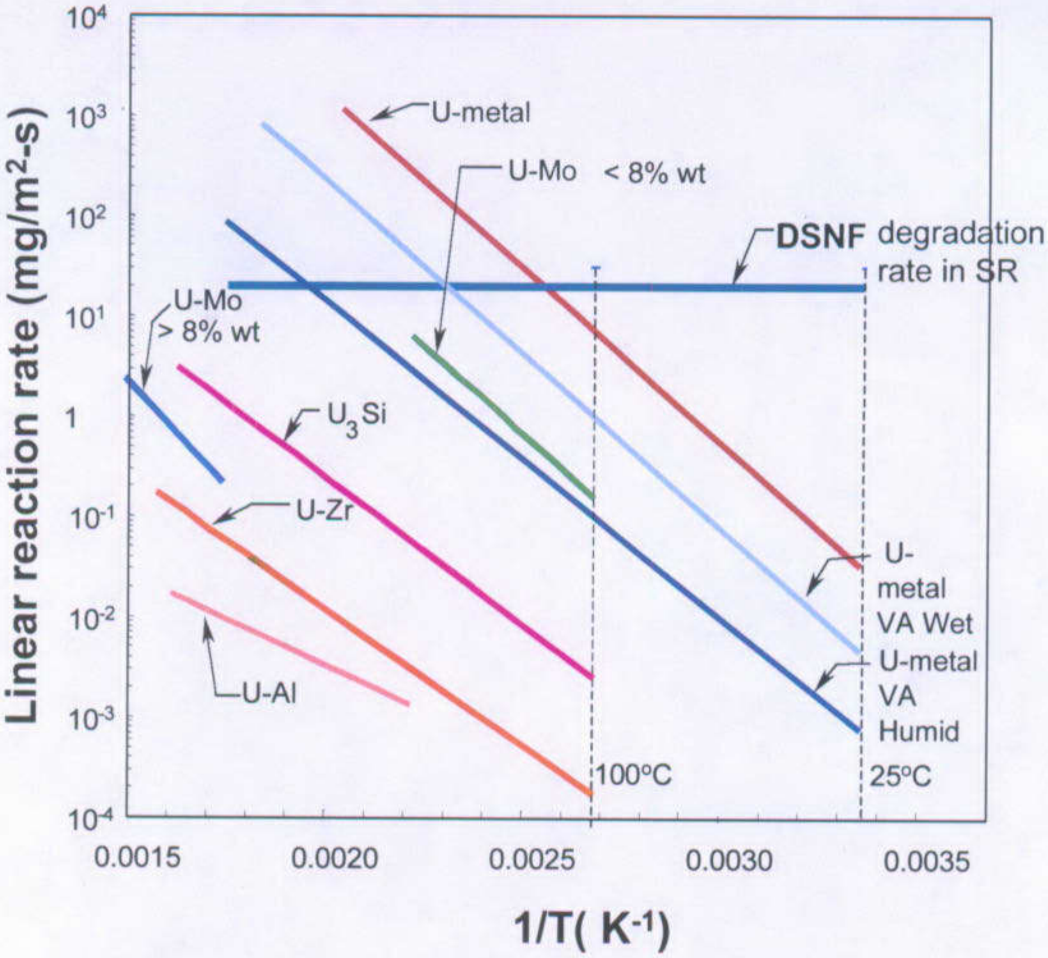


Figure 2. Constant Degradation Rate Used for DSNF Category Bounds all Measured Degradation Rates (DOE 2000, Figure 1-2)

Influence of Heterogeneous Fuels—Heterogeneous SNF are (1) particle-dispersion fuels (DOE 1999b), wherein the fuel was fabricated with the fuel meat as a separate phase from the fuel matrix, or (2) fuel in which the fuel meat itself experienced "phase separation" in reactor service or storage.

The heterogeneous particle-dispersion DSNFs were fabricated as composites, in which the continuous fuel matrix differs in composition from the dispersed particulate, fuel-meat phase. In these fuel types, the encapsulating matrix is the dominant, continuous phase. Examples of the particle-dispersion fuels are (DOE 1999b):

- Aluminum-based UAl_x and U_xSi_y fuels. In this fuel type, particles of intermetallic UAl_x, U_3O_8, or U_xSi_y are dispersed in a continuous aluminum matrix.
- Graphite fuels. The graphite fuels consist of highly enriched UC_2 and fertile ThC_2 particles dispersed in a continuous graphite matrix. The spherical particles of UC_2 and ThC_2 are individually encapsulated by multiple coatings of highly protective SiC and/or pyrolytic carbon. The graphite matrix is a continuous binding matrix that forms an additional encapsulant for the coated particles.

- $U-ZrH_x$ (TRIGA) fuels. In TRIGA fuels, minute particles of uranium metal are dispersed in a continuous ZrH_x phase.

It is generally recognized that the dissimilar fuel-matrix and fuel-meat materials have different corrosion characteristics, and could exhibit selective dissolution or leaching. The differences in corrosion can be qualitatively described as follows (DOE 1999b):

- UAl_x corrodes more slowly than the aluminum matrix in unirradiated samples. However, the reaction rates for the two materials appear to be comparable in irradiated fuel. Data generated recently for aluminum-based fuels at Pacific Northwest National Laboratory (PNNL) indicated that there is no differential corrosion or selective leaching.

- The carbonaceous (graphitic) matrix of graphite fuels, and the pyrolytic carbon and SiC coatings on the UC_2 and ThC_2 kernels, are highly resistant to aqueous corrosion and corrode much more slowly than UC_2 and ThC_2.

- The ZrH_x matrix of TRIGA fuels is highly resistant to corrosion, and corrodes much more slowly than uranium metal.

The above matrix materials, and the pyrolytic carbon and SiC coating materials, all are more resistant to corrosion than uranium metal, the material used to bound the DSNF in the TSPA dose calculations. Of the fuel-meat compounds, only Pu/U carbides (Group 3) corrode faster than uranium metal; however, their coatings render them highly resistant to corrosion, much more so than uranium metal (CRWMS M&O 2000w). Existing data (DOE 1999b, Section 6.3) also show that Pu/U carbide of the non-graphite, Group 3 SNF also has a dissolution rate potentially greater than that of the uranium metal SNF; however, the inventory of this material is extremely small.

In dispersion fuels, only the degradation, alteration, and dissolution of the dominant continuous-fuel matrix phase are of concern. The dominant matrix phase encapsulates and effectively isolates the dispersed phase that contains the all of the fissile material and most of the fission products. Because in all cases the dominant phase is much more resistant to corrosion than uranium metal, the use of uranium-metal corrosion rates in the TSPA model conservatively bounds the corrosion and dissolution rate of dispersion fuels.

While the UC_2 and UH_3 may represent a small fraction of the fuel mass, the UC_2 and UH_3 phases are important because they may be more reactive than uranium metal and could significantly accelerate fuel corrosion. However, the uranium-metal mass is the dominant matrix phase that encapsulates the UC_2 and UH_3. Therefore, this microstructure is not expected to affect the corrosion of uranium metal matrix to a detectable extent. While the UH_3 in the corrosion product occlusions in the N-Reactor fuel may be reactive to water and air, it is not expected to affect the rate of corrosion of the adjacent metal surface. In fact, recent NSNFP results (DOE 1999b) from testing of N-Reactor fuel samples have indicated that the degradation of uranium-metal fuel is non-selective because it is controlled by the dissolution of the dominant metal phase. A NSNFP corrosion testing program is in progress to determine experimentally the oxidation rates of UH_3 and the corrosion rates of irradiated N-Reactor fuel, as well as characterizing the UH_3 content in the corrosion samples and the corrosion product. The results will be incorporated in to future analyses as they become available.

Influence of Pu Ceramic—Pu is incorporated in a discrete crystalline phase that is embedded in a vitrified glass phase. The crystalline phase is currently envisioned to be a titanate-based ceramic similar to Synroc (Shaw 1999). This material is more degradation resistant than the HLW glass. Thus, release from the waste form may be conservatively bounded by specifying that the Pu is uniformly distributed in the glass phase and by applying glass dissolution kinetics (CRWMS M&O 2000d) to estimate the release rate (CRWMS M&O 2000e, Section 6.3.12)

Influence of Selective Leaching—Except for gap and grain-boundary release, leaching of all radionuclides is expected to be non-selective because it will be controlled by the dissolution of the dominant phase in relatively homogeneous fuels such as metallic-uranium fuel and MOX fuel.

For high-enrichment, high-burnup fuels, such as some of the aluminum-based fuels that may consist of two or more phases, selective leaching would appear to be feasible. However, based on data generated recently for aluminum-based fuels at PNNL, there appears to be little or no selective leaching (CRWMS M&O 2000e).

Because of the limited data base that is available to support explicit modeling of selective leaching and other separate effects, the TSPA-SR approach is to use conservative bounding models to calculate the rate of release from the DSNF inventory (CRWMS M&O 2000e).

Influence of Rapid Oxidation —For modeling the effect of combustion of the metallic uranium on dose, the reasonable conclusion (CRWMS M&O 2000e, Section 7.2.1) is also made that all metallic uranium oxidizes within one time step following failure of the waste package. This conservatively bounds the release of radionuclides resulting from a pyrophoric event (CRWMS M&O 1999m) (see FEP 2.1.02.08.00, "DOE SNF Pyrophoricity").

Influence of High-Integrity Can—A high-integrity can (HIC) is a special-purpose container being designed and evaluated for the disposal of certain parts of the DOE spent-fuel inventory (DOE 1998a). The spent fuel designated for HIC disposal will be fuel that can not be disposed in standard waste package because the material has lost its integrity. The material will be some combination of fuel sections used for laboratory analysis, powdered fuel, damaged or broken parts of fuel, severely degraded fuel, and small lots of fuel used for experiments in research facilities in the United States and around the world. The HIC is needed to allow containment of poorly categorized, fragmented, or damaged spent fuel to meet near-term environmental and/or regulatory requirements.

Because the spent fuel designated for HIC disposal will contain fuels with substantial inventories (CRWMS M&O 2000am, Section 6.5) of radionuclides, the HIC will be manufactured from material designed to provide a barrier that delays release of these materials until well after waste-package failure. The design of the HIC will allow the fuel in the HIC to be isolated from other spent fuel in the waste package, thus preventing any unintended synergistic reactions or criticality. (CRWMS M&O 2000am, Section 6.5).

Preliminary results show that the peak dose from the HIC inventory is approximately three orders of magnitude less than the TSPA-VA base case dose. The results also show that the magnitude of the difference in total dose between the TSPA-VA base case and the HIC cases is smaller at early times than later times. The peak dose from disposal of the DOE SNF in the HIC is delayed by about 60,000 years as compared to disposing of the same inventory without the

HIC. In both cases the peak dose is the same magnitude (CRWMS M&O 2000am, Section 6.5.3).

Influence of In-Package Water Chemistry—In-package water chemistry can vary significantly from the seepage water chemistry because of reactions with waste-package and waste-form materials. The primary processes that are likely to control the major element chemistry include equilibration with the gas phase CO_2 and O_2, dissolution of steel alloys (baskets and waste package), dissolution of fuel elements, and growth of secondary oxides and carbonate minerals and possibly sorption (CRWMS M&O 1999d). For DSNF the degradation dependence on water chemistry is subsumed in the constant degradation rate.

6.2.7 CSNF Alteration, Dissolution, and Radionuclide Release—YMP No. 2.1.02.02.00

YMP Primary FEP Description: Alteration of the original mineralogy of the commercial spent nuclear fuel (CSNF) (under wet or dry conditions) and dissolution of the uranium-oxide matrix can influence the mobilization of radionuclides. The degradation of UO_2 could be affected by a number of variables, such as surface area, burnup, temperature, overall solution electrochemical potential (Eh), pH, and especially solutions containing significant concentrations of calcium, sodium, carbonate and silicate ions, as well as availability of organic complexing materials. In turn, these water properties are affected by the alteration of the cladding and matrix.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.01.02.00	Codisposal/Collocation
2.1.01.03.00	Heterogeneity of waste forms
2.1.09.04.00	Radionuclide solubility
2.1.09.10.00	Secondary phase effects
2.1.02.12.00 through 2.1.02.27.00	Cladding related FEPs

IRSR Issues: *TSPAI3, ENF3, CLST3*

Screening Argument: The rate of the alteration, degradation, and dissolution of the CSNF matrix is a very fundamental process that can control the rate of release of several moderately soluble to fairly soluble radionuclides (e.g., ^{99}Tc, ^{129}I) and, therefore, has been included under CSNF Degradation Component of the Waste Form Degradation Model. The rate of the alteration of the protective cladding is evaluated separately in the CSNF Cladding Degradation Component (see cladding unzipping FEP 2.1.02.23.00). Less soluble radionuclides (e.g., ^{239}Pu, ^{241}Am, ^{237}Np) are potentially mobilized based on their elemental solubility limit and included as a separate model component, Dissolved Radionuclide Concentration Component (see solubility

FEP 2.1.09.04.00). In addition, the potential for colloidal mobilization of several radionuclides has been included as the Colloidal Radionuclide Concentration Component (see colloid FEP 2.1.09.14.00).

CSNF alteration, dissolution, and radionuclide release is included in the TSPA as discussed in the *TSPA Disposition.*

TSPA Disposition: Degradation and dissolution of the CSNF matrix is included in the Waste Form Degradation Model as a specific component, CSNF Matrix Degradation Component as summarized in the CSNF degradation AMR (CRWMS M&O 2000c). The component consists of a kinetic rate equation (also known as an intrinsic dissolution rate or forward reaction rate equation under saturated conditions). The coefficients of the equation have been evaluated through regression analysis on high-flow rate experimental data on irradiated SNF and unirradiated UO_2 obtained over a range of temperatures and water chemistry (CRWMS M&O 2000c), specifically pH, CO_3, and oxygen potential. This rate equation is used as the upper bound on the availability of radionuclides for potential mobilization. (CRWMS M&O 1998c, Section 6.3.1.3.2; CRWMS M&O 1999h, Section 6.2.2.2). The chemistry of the water passing through the package is evaluated by the In-Package Chemistry Component using response surfaces generated by a geochemistry process model (CRWMS M&O 2000i).

Supplemental Discussion: Adequately discussed in this section under *Screening Argument* and *TSPA Disposition.*

6.2.8 Glass Degradation, Alteration, and Dissolution—YMP No. 2.1.02.03.00

YMP Primary FEP Description: Glass waste forms are thermodynamically unstable over long time periods, and will alter on contact with water. Radionuclides can be mobilized from the glass waste by a variety of processes, including degradation and alteration of the glass, phase separation, congruent dissolution, precipitation of silicates co-precipitation of other minerals including iron corrosion products, and selective leaching.

Screening Decision and Regulatory Basis: *Included* (in-package minor phase separation, congruent dissolution, and chemistry-dependent corrosion rates).

Excluded (based on low consequence, phase separation, precipitation of silicate and other minerals, and selective leaching).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.01.02.00	Codisposal/Collocation
2.1.01.03.00	Heterogeneity of waste forms
2.1.02.01.00	DSNF degradation
2.1.09.04.00	Radionuclide solubility

IRSR Issues: TSPAI3, ENF3, CLST1, CLST3, CLST4

Screening Argument: In-package minor phase separation, congruent dissolution, and chemistry-dependent corrosion rates are included in the TSPA discussed in the *TSPA Disposition* for this FEP.

Phase separation can be excluded based on low consequence because controls will be implemented as part of the waste-production and acceptance processes to ensure that significant phase separation will not occur. Also, available data indicate that the radionuclide-release performance of waste glass is not sensitive to minor phase separation. Therefore, any minor phase separation that is omitted from the TSPA will not significantly change calculated expected annual dose.

Precipitation of silicates and other minerals can be excluded based on low consequence. Omission of precipitation of silicates and other minerals will not significantly change the calculated expected annual dose. No credit is taken either for the internal structure of the waste package in delaying the movement of water or for the retarding effects of clays and zeolites on the transport of radionuclides. Also, precipitation of most silicates will be slower than glass dissolution. Consequently, the removal of silica by precipitation of secondary solids is overcompensated for by the release of silica from the dissolving glass (CRWMS M&O 2000d), at least in the vicinity of the glass surfaces. Thus, silicate precipitation is not expected to lead to increased glass-corrosion rates.

Selective leaching can be excluded based on low consequence. Inclusion of selective leaching in the TSPA-SR calculation would reduce calculated expected annual dose. Therefore, its omission as a system wide feature from the performance assessment models will not significantly change the interpretation of the calculated expected annual dose. Selective leaching of alkali metals and alkaline earths is expected to occur upon initial contact of waste glass with groundwater. However, this process is a transient and unimportant process for radionuclide release. Also, the basis for the modeling approach is that the rate of corrosion of the waste-glass matrix represents an upper bound on the rate of release (or leaching) of the radionuclides in the glass matrix.

TSPA Disposition: HLW degradation and dissolution is included in the TSPA as a source term for the mobilization of contaminants in the HLW Degradation Component of the Waste Form Degradation Model as detailed in HLW degradation AMR (CRWMS M&O 2000d). The chemistry of the water passing through the package is altered based on response surfaces generated from the geochemical numerical model (CRWMS M&O 2000i). The glass-dissolution model used in the TSPA is derived from a reference glass composition and will be applicable to glasses whose compositions lie within specified bounds. The glass-dissolution model takes into account the effects of current and future glass compositions, vapor hydration of the glass prior to contact by liquid water, and alteration-phase formation on the dissolution rate. The corrosion rate of the glass is determined by its composition and temperature as well as the composition and pH of the solution with which it has contact. However, a bounding parameter value is used in the model to account for the effect of the silicic-acid concentration.

The degradation, alteration, and dissolution model for HLW in the TSPA can be summarized in the following steps:

- In-package water chemistry is determined from initial conditions, seepage flow rate, waste-package and matrix dissolution via a response surface(s) generated using geochemical-model calculations. The WF dissolution information comes in as a feedback from the results of the degradation model on the previous performance-assessment-model iteration.

- HLW is altered under full-immersion conditions using a kinetic rate equation, which is defined with two sets of parameters over two pH regimes. The rate per unit area is a function of pH and temperature. A bounding parameter value is used in the model to account for the effect of the silicic acid concentration. The rate parameters are determined by linear regression fit to experimental data over a range of temperatures and pH from flow through experiments (CRWMS M&O 2000d, Section 6.2).

- Radionuclides are made available for mobilization proportional to the fraction of waste altered (congruently). The fraction of waste altered is the rate times an effective surface area. The effective area is some factor (20 in the TSPA-VA analyses) (CRWMS M&O 1998c, pp. 6-79; CRWMS M&O 2000n, Section 3.6.1) times the geometric area.

- Concentration of each radionuclide is calculated based on the minimum of rind (altered zone) water volume and seepage volume.

- Concentration of each radionuclide is limited by a sampled or empirical elemental solubility limit. Highly soluble radionuclides (Tc, I) are released congruently as the matrix is altered. Low-solubility radionuclides (Pu, Am, Np) are solubility limited.

- The contribution of colloid formation to the radionuclide concentration is determined in the Colloid Radioisotope Concentration Component of the TSPA. The component determines the mass of radioisotopes released from the waste either reversibly or irreversibly attached to mobile colloids.

Phase separation, precipitation of silicate and other minerals, and selective leaching have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Glass is a thermodynamically metastable, covalent/ionic solid whose degradation depends on ion exchange, surface complexation, and silica concentration (CRWMS M&O 1999h, Section 3.5.2.6). These three processes, in turn, depend upon temperature and pH. When a disposal container breaches, water vapor may be the first reactant to enter the container. Water vapor could alter the glass to produce a gel layer containing high concentrations of radioisotopes. This altered glass would then have a higher release rate than unaltered glass, when liquid water enters the breached disposal container. The model to be developed for aqueous degradation of HLW in the TSPA-SR will be similar to that in the TSPA-VA (CRWMS M&O 1998c, Chapter 6, p. 6-73) where silica concentration and temperature were the primary variables. However, a bounding parameter value is used in the model to account for the effect of the silicic acid concentration and pH is included in the empirical equation. TSPA-SR will evaluate the sensitivity to this sequence of events by developing a vapor hydration model for the borosilicate glass.

Details of specific HLW degradation processes are discussed below:

Rate of Glass Dissolution—The dissolution rate of glass depends on the glass composition and on the temperature, pH and concentration of dissolved silica in the solution with which the glass is in contact. Because of the small amounts of liquid water expected to contact the glass in the disposal system, transformation of the glass to thermodynamically stable phases, such as clays and zeolites, will occur slowly and mitigate the release of radionuclides. The glass dissolution rate may increase on precipitation of certain alteration phases. The durability of glass is expected to enhance the performance of the disposal system. The glass-dissolution model that was included in the TSPA-VA has been evaluated and updated to ensure that the effects of possible differences in the values of model parameters for current and future glass compositions, vapor hydration of the glass prior to contact by liquid water, and alteration-phase formation were taken into account in the calculated dissolution rate (CRWMS M&O 2000d, Section 5). The model provides upper limits to glass-dissolution rates when contacted by humid air or dripping water, and upon immersion. The rate calculated using the model bounds the rates observed in laboratory tests in which rate-increasing alteration phases formed during the test or were added separately. Phases present in the solution other than glass alteration phases can affect the glass dissolution rate. For example, the presence of ductile iron (McVay and Buckwalter 1983) and clay particles (Van Iseghem and Lemmens 1993) have been observed to increase the glass-dissolution rate. As in the case of glass-alteration phases, the effect is attributed to the removal of dissolved silica from solution.

The model developed for use in the TSPA-SR and TSPA-LA includes parameters that account for the effects of the glass composition, temperature, pH, and silicic acid in the solution. The presence of alteration phases and other phases affects the pH due to changes in the silicic-acid concentration. The model includes a term that combines parameter values for the glass composition and the silicic-acid concentration that bound the dissolution rates of reference glasses after alteration phases have formed. The effects of the pH (in the case of immersion) and temperature (in the case of contact by humid air, dripping water, or immersion) on the rate are calculated explicitly.

Congruent Dissolution—The dissolution rate is defined as the rate of decomposition (grams/area/time or volume/area/time = length/time) of the glass-lattice-structure surface in contact with an aqueous solution. Thus, for a homogeneous radioactive glass, the dissolution rate is congruent by definition. However, because of potential precipitation, colloidal, and adsorption kinetic processes, the release rate of radionuclide concentrations may not be congruently related to the solid-state concentrations of the glass waste form. Also, see YMP No. 2.1.02.03.05 for additional related comments.

Selective Leaching—Selective alkali metals and alkaline earths leaching involves the incongruent release of these elements from the waste form when it is initially contacted by aqueous solutions. It is attributed to a variety of processes (e.g. ion exchange and selective dissolution) that occur near the glass/solution interface (Cunnane et al. 1994a, 1994b). This leaching is, however, a transient process; the important process that controls the radionuclide release is the hydrolysis and dissolution (often referred to as "corrosion") of the glass matrix (Cunnane et al. 1994a; 1994b).

A conservative bounding approach will be used to model the release of radionuclides from corroding HLW glass in the repository. The basis for this modeling approach is the contention that the rate of corrosion of the waste-glass matrix represents an upper bound on the rate of release (or leaching) of the radionuclides in the glass matrix (CRWMS M&O 2000d, Section 5). It does not require explicit modeling of the selective leaching of alkaline earth or other elements from the waste glass.

Phase Separation—Improper heat treatment of glass can produce macroscopic phase separation, which can lead to a reduction of the chemical resistance of the glass. In particular, this process would favor the selective leaching of Cs and Sr. However, product-acceptance specifications and production controls will preclude significant phase separation in the HLW glass waste forms that are scheduled for geological disposal at Yucca Mountain (DOE 1995). The specific product-acceptance specifications, from DOE (1995), that are pertinent to phase separation are the waste-form specifications: 1.1 Chemical Specification, 1.3 Specification of Product Consistency, and 1.4 Specification of Phase Stability.

Even if minor phase separation were to occur during production and if the glass was accepted for disposal, available information indicates that the phase separation would not significantly influence the waste glass performance in the repository (Cunnane et al. 1994a; 1994b, Section 2.1).

Precipitation of Silicate—Precipitation of silicate alteration phases has been observed to occur as glass dissolves (CRWMS M&O 2000d, Section 6.2.2). An increase in the glass dissolution rate has also been observed to occur coincident with the formation of these alteration phases. Although the relationship between the test conditions under which the effect of alteration phase formation on the dissolution rate were observed and the anticipated conditions in the disposal site is not known, the small amounts of water that may contact the glass will be conducive to phase formation. The possibility that phase formation could increase the dissolution rate was included in development of the glass-dissolution model for the TSPA-SR and TSPA-LA (CRWMS M&O 2000d, Section 5). Model parameter values were selected so that the dissolution rate calculated by the model for a particular glass was higher that the rates that had been measured in laboratory tests with that glass in which rate-affecting alteration phases had formed. The effects of the glass composition and the solution composition are combined in a single term in the model. The effects of the pH (in the case of immersion) and temperature (in the case of contact by humid air, dripping water, or immersion) on the rate are calculated explicitly.

In-Package Water Chemistry—In-package water chemistry can vary significantly from the seepage-water chemistry because of reactions with waste-package and waste-form materials. The primary processes likely to control the major-element chemistry include: equilibration with the gas phases CO_2 and O_2; dissolution of steel alloys (baskets and waste package); dissolution of fuel elements and growth of secondary oxides and carbonate minerals and possibly sorption. The variation in major-element composition of in-package fluids was examined with reaction-path simulations, which deal primarily with reactions that occur after the original thermal pulse and temperatures have returned to near ambient levels (CRWMS M&O 2000m).

The in-package water chemistry and WF degradation are coupled processes. WF degradation is a function of the water chemistry (pH, Eh, dissolved carbon, silica and ionic strength), and the water chemistry is altered by the degradation process. This coupling is handled through a feedback mechanism where in-package water chemistry is calculated using degradation parameters from the previous performance-assessment-model time step.

6.2.9 Alpha Recoil Enhances Dissolution—YMP No. 2.1.02.04.00

YMP Primary FEP Description: During decay of certain radionuclides, alpha particles may be emitted with sufficiently high energies that the daughter nuclide recoils appreciably to conserve system momentum. A result of recoil is that certain radionuclides, such as ^{234}U exhibit substantially greater dissolution rates (with the same solubility limits) and can be transported preferentially.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.02.00	CSNF degradation

IRSR Issues: CLST3, CLST4

Screening Argument: The effects of alpha recoil can be excluded based on low consequence. Omission of the effects of alpha recoil will not significantly change the calculated expected annual dose because the alpha-recoil dissolution rates have been shown to be much less than the dissolution due to chemical processes. Even when reasonably concluding that all radioactive decay results in an alpha recoil, it does not cause significant increases in the dissolution rate of the different waste forms. The processes investigated in this argument only involve direct nuclear effects and not indirect non-nuclear effects such as thermal-enhanced dissolution due to heat generation from the radioactive decay of SNF. For details see the *Supplemental Discussion* for this FEP (YMP No. 2.1.02.04.00).

TSPA Disposition: The effects of alpha recoil have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Radioactive decay is a mechanism for increasing the transfer of radionuclides from degraded waste forms into groundwater in the immediate vicinity. This contaminated groundwater may then be ultimately released to the subsurface unit boundary and then to the biosphere. The transport of the radionuclides through natural geologic media is dependent upon many site-specific factors such as: mineralogy, geometry, and microstructure of the rocks, as well as the geometric constraints on the type of groundwater flow (e.g., rock matrix or fracture flow). Radioactive decay of radionuclides after they are being transported in groundwater is not of concern since they will not, to any measurable quantity, increase the release of radionuclides released from the waste forms into the groundwater. The decay within the groundwater will only transmute the specific radionuclide inventory already being transported by the groundwater (i.e., due to colloids, dissolution, etc.), and the subsequent decay chains from the transported radionuclides can be modeled using radionuclide-transport computational codes.

Of the various radioactive decay modes (i.e., alpha (α) decay, beta (β) decay, gamma (γ) decay, spontaneous fission [SF], isomeric transition [IT], etc.) the most important for fissile materials is the alpha-decay mechanism. This mechanism is the dominant decay mode for heavy radionuclides. All the heavy nuclides above ^{209}Bi are radioactive and tend to decay into more stable nucleus configurations (i.e., atomic masses less than 200 AMUs). Because these heavy radionuclides need to lose significant quantities of mass in order to become more stable, in general they will decay by the mode that results in the largest mass loss. Thus, the decay mode with the highest probability of mass loss will be alpha decay, which has the largest mass and associated kinetic energy. Although many of the heavy radionuclides emit alpha particles with energies greater than 4.0 MeV (see Parrington et al. 1996, p. 48), there is no gamma-decay or beta-decay with energy greater than 4.0 MeV and only a very few with energy greater than 1.0 MeV. Thus, alpha-recoil mechanisms will bound the effects due to beta- and gamma-recoil. Other special decay modes such as IT and SR decay have probabilities of occurrence that are orders of magnitude less than that of alpha decay. (Information in Attachment II indicates that IT occurs for ^{108m}Ag, ^{242m}Am, ^{93m}Nb, and ^{121m}Sn; none of which are significant in terms of mass contribution. Also, information from Lederer and Shirley (1978, p. 1464) indicates that the SF half-lives are several orders of magnitude longer than that for other decay modes and are, thus, insignificant.)

The number of atom displacements per alpha decay can be calculated by noting that when a recoil nucleus strikes an atom, it requires a minimum displacement energy, E_d, of approximately 25 ev to eject the struck atom from its lattice site (Foster and Wright 1973, p. 296). The total number of displacements caused by a single alpha-decay event is given by Equation 1 (Foster and Wright 1973, p. 296. Equation gives displacement units.):

$$Disp(E) = \int_0^\infty P(T)K(E,T)\nu(T)dT = \int_{E_d}^{T_m} P(T)K(E,T)\nu(T)dT \qquad \text{(Eq. 1)}$$

where

P(T) is the probability that an atom (primary *knock-on*), struck by either the emitted alpha or the alpha-recoil atom, receiving energy *T* is displaced.

K(E,T) is the probability for the transfer of kinetic energy *T* to the primary knock-on atom of energy *E*.

ν(T) is the total number of displacements in a cascade originating from a primary knock-on atom whose energy is *T*.

The expression is integrated over the energy range starting at the displacement threshold energy, E_d, and ending at the maximum energy that can be transferred to an atom, T_m. Since the displacement of atoms corresponds to a threshold event, *P(T)* is modeled as a Heaviside step function (Foster and Wright 1973, p. 297):

$$P(T) = \langle T - E_d \rangle^0 = \begin{cases} 0 & if\ T < E_d \\ 1 & if\ T \geq E_d \end{cases} \qquad \text{(Eq. 2)}$$

To simplify the analysis, the probability for kinetic energy transfer is treated as being a uniform distribution over the applicable energy range and is:

$$K(E,T) = \frac{1}{T_m - E_d} \qquad \text{(Eq. 3)}$$

The total number of displacements which originate from a primary knock-on is described by the Kinchin-Pease Model and is (Foster and Wright 1973, p. 297):

$$\nu(T) = \frac{T}{2E_d} + \frac{E_i - T}{2E_d}\langle T - E_i \rangle^0 = \begin{cases} T/2E_d & \text{if } T \le E_i \\ E_i/2E_d & \text{if } T > E_i \end{cases} \qquad \text{(Eq. 4)}$$

where

E_i is the energy required for ionization.

The $2E_d$ in the denominator accounts for the displacement of the knock-on atom and the additional E_d for the striking atom to also leave the displacement site. The model also reasonably concludes there is an ionization threshold ($E_I \approx 1000A$) below which displacements take place and above which only ionization takes place.

The total number of displacements is given by Equation 5, which includes ionization interactions.

$$Disp(t_m) = \begin{cases} \displaystyle\int_{E_d}^{T_m} \frac{T}{2E_d T_m - 2E_d^2}\,dT & \text{if } T_m \le E_i \\ \displaystyle\int_{E_d}^{E_i} \frac{T}{2E_d T_m - 2E_d^2}\,dT + \int_{E_i}^{T_m} \frac{E_d}{2E_d T_m - 2E_d^2}\,dT & \text{if } T > E_i \end{cases}$$

$$= \begin{cases} \dfrac{T_m + E_d}{4E_d} & \text{if } T_m \le E_i \\ \dfrac{E_i^2 - E_d^2}{4E_d(T_m - E_d)} + \dfrac{E_i T_m - E_i^2}{2E_d(T_m - E_d)} & \text{if } T_m > E_i \end{cases} \qquad \text{(Eq. 5)}$$

For the example involving ^{238}U decaying to $^{234\text{Th}}$, there are two cases to examine. The first is for the alpha-recoil atom, and the second is for the emitted alpha particles. From the analysis in Attachment III, it can be seen that the alpha-recoil atom (^{234}Th) has a value 0.072 MeV for T_m, which is much less than the ionization energy, typically $\approx 1000A$ = 0.234 MeV. This results in $Disp_{max}(\alpha)$ = 720 displacements per alpha decay. (This does not include replacement collisions along with focusing and channeling effects, which will significantly lower the estimate for displacements.) If each of the secondary displacements follows a bifurcation process (i.e., 2^N = 720), (standard mathematical terminology) this would correspond to 9.49 bifurcation levels. This means that the maximum number of atom mono-layers of the SNF fuel meat through which recoil-nuclei (due to alpha decay) could pass through and enter bounding groundwater is approximately 10. This is only for recoil atoms traveling in the direction of the groundwater. Attachment III indicates that only half of the recoil could be in the proper direction. This also does not mean that all the (secondary) displaced atoms within ten mono-layers of the material surface would enter the groundwater, it only indicates that it is possible.

The second case, the emitted alpha particles, involves a much larger number of possible displacements. This is due to its larger kinetic energy. From Lederer and Shirley (1978), it can be identified that the maximum energy for alpha particles emitted from ^{238}U is 4.196 MeV. Since this energy is above the ionization value, the second version of Equation 5 that contains two terms must be used to take into account the ionization by the alpha particles. Using the maximum alpha energy results in a value of 4550 for $Disp(\alpha)$. When added to the recoil-atom value, the net displacements are 5270, which corresponds to approximately 12.4 bifurcation levels.

The enhancement to the dissolution rate due to alpha decay can now be upper bounded. This rate will correspond to the number of alpha decays per unit time within the first thirteen half mono-layers of the material surface that result in nuclei recoiling toward the groundwater. The worst case density for thorium (or even uranium) will approach that due to pure plutonium metal, with a maximum density of 19.84 g/cm^3 (Wick 1980, Table 7.1) and a mono-layer thickness of approximately 3.0 Å ($3.0\text{x}10^{-10}$m). The combination of these two parameters results in 0.039 grams of SNF material that is within the first thirteen mono-layers of a surface area of 1.0 m^2 that are in the direction of the material surface. When this surface density (0.039 g/m^2) is multiplied by the fractional rate at which the SNF material experiences radioactive α-decay, it can be expressed in the units used in the Technical Basis Document (CRWMS M&O 1998c, A6-29) for dissolution of the SNF material due to chemical processes. The fractional rate was determined from radiological half-lives along with some data from Attachment II, which contains inventory information for the Yucca Mountain repository. These data identify that the major constituents, on a mass basis, of the final waste forms (spent nuclear fuel, and high-level waste) are: ^{238}U, ^{235}U, ^{239}Pu, ^{236}U, and ^{240}Pu. Key radionuclides from Attachment II are analyzed in Table 5, which identifies their maximum alpha-recoil enhancement for dissolution rates. Data in column 5 of this table are computed by noting that the fractional decay rate for the radionuclides is given by their decay constant ($\lambda = ln(2)/\tau_{½}$). To put these values from Table 5 into proper perspective, comparison to Figure 3 needs to be made. Figure 3 identifies the dissolution rates of different waste forms due to chemical (non-nuclear) processes. As can be identified, the alpha-recoil dissolution rates are much less than the values due to chemical processes.

In summary, the radioactive decay processes that directly increase fuel-meat dissolution are bounded by alpha-recoil rates. Even when reasonably concluding that all radioactive decays result in an alpha-recoil, they will not cause significant increases to the dissolution rate of the different waste forms. The processes investigated in this argument are only for direct nuclear effects and not indirect non-nuclear effects such as thermal-enhanced dissolution due to heat generation from the radioactive decay of SNF. Thus, the overall dissolution due to direct nuclear-decay processes has been eliminated from performance-assessment calculations on the basis of low consequence to the performance of the disposal system.

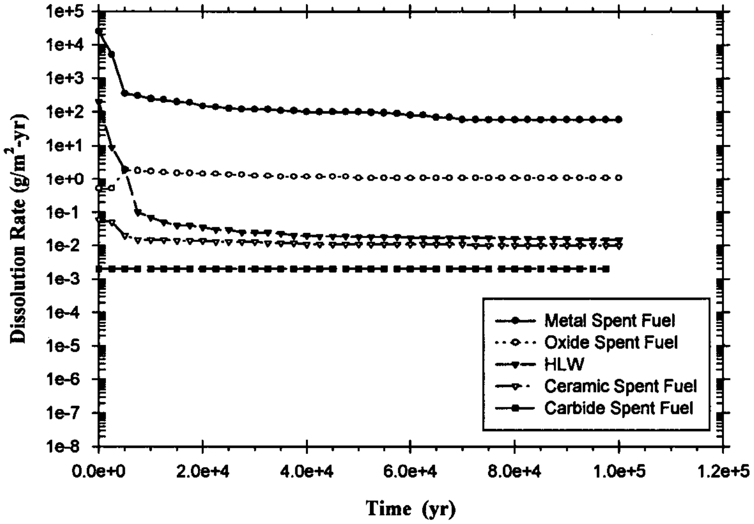
Table 5. Alpha-Recoil Enhanced (from Both α and α-Recoil Atom) Dissolution Rates Due to the Major Mass-Based Constituents of SNF and HLW to be Disposed in the Yucca Mountain Potential Repository

Nuclide ID	Decay Mode	Half-Life [(a)] yr)	Fraction Decay Rate [(b)] (1/yr)	α–Decay Rate in 13 Mono-Layers[(c)] (g/m^2–yr)
^{235}U	α, γ, SF	4.47 x 10^9	1.55 x 10^{-10}	6.05 x 10^{-12}
^{235}U	α, γ, SF	7.04 x 10^8	9.85 x 10^{-10}	3.84 x 10^{-11}
^{239}Pu	α, γ, SF	2.410 x 10^4	2.88 x 10^{-5}	1.12 x 10^{-6}
^{236}U	α, γ, SF	2.342 x 10^7	2.96 x 10^{-8}	1.15 x 10^{-9}

(a) Data obtained from Attachment II. The original source of information, in radioactivity units, was DOE (1998d). Mass unit values were derived in Sanchez et al. (1998).

(b) The fraction decay rate, also known as the decay constant, is given by $\lambda = \ln(2)/t_{½}$, where $t_{½}$ is the radionuclide half-life given by values in column 3.

(c) Each mono-layer thickness is 3.0 Å (3.0x10^{-10}m), and the density is upper bounded at 19.86 g/cm^3 (theoretical density of pure plutonium metal [Wick 1980]).



NOTE: (CRWMS M&O 1998c, p. A6-29, Figure A6-1)

Figure 3. Comparison of Dissolution Rates for High-Level Waste, Metallic Carbide, and Ceramic Spent Nuclear Fuel

Attachments:

See the following attachments:

Attachment II, "Radionuclide Inventory for Final Waste Forms"

Attachment III, "Alpha-Recoil Mechanics"

6.2.10 Glass Cracking and Surface Area—YMP No. 2.1.02.05.00

YMP Primary FEP Description: Cracking of the HLW glass on cooling and during handling means that the surface area of the glass is greater than the surface area of a monolithic block. The increase in the surface area could affect the rate of glass alteration and radionuclide dissolution.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.03.0	HLW degradation

IRSR Issues: CLST4

Screening Argument: A robust container and relatively cool repository temperatures, relative to the glass transition temperature (approximately 400°C) of glass, are quite favorable to the longevity of HLW glass. These two factors should ensure that significant glass cracking, beyond that accompanying the manufacturing process, does not occur either during handling or during the thermal period. Nonetheless, any possible increase in surface area due to cracking is included in the TSPA as discussed in the *TSPA Disposition.*

TSPA Disposition: The effect of using an area-enhancement factor will appear in the characterization of mobilization for HLW within the HLW-Degradation Component of the TSPA. For details, see the HLW degradation AMR (CRWMS M&O 2000d).

Supplemental Discussion: During manufacture, molten HLW glass is poured into a robust container at a slow rate and cooled slowly. Despite such precautions, temperature gradients between inner and outer regions, heterogeneity of the HLW-glass mixture, and differing thermal coefficients of expansion between the glass and the canister cause cracking.

During handling, the robust canister and container used at Yucca Mountain should provide adequate protection against additional glass cracking. The effects of manufacturing and handling processes on the surface area are included in modeling the rate of glass corrosion (CRWMS M&O 2000d, Section 6.1.2).

With the most current design, repository temperatures should be within a few degrees of ambient temperatures, and temperature changes affecting the glass should occur much more slowly than during manufacture.

During operational and post-operational periods, therefore, repository temperatures are expected to cause relatively minor cracking and fragmentation beyond that which occurs during manufacture. Thus, the use of a surface-area-enhancement factor of 20, as currently employed by the TSPA, represents an adequate approach for characterizing the surface-area effect of glass cracking and fragmentation during both operational and post-operational periods.

6.2.11 Glass Recrystallization—YMP No. 2.1.02.06.00

YMP Primary FEP Description: HLW glass recrystallization could occur and would lead to a less corrosion-resistant waste form. However, recrystallization is a very slow process and possible only if a high glass temperature is maintained over a prolonged period. It is not expected to occur below 400°C.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.03.0	HLW degradation

IRSR Issues: CLST4

Screening Argument: HLW glass recrystallization can be excluded from the TSPA based on low consequence because controls are to be implemented as part of the waste production and acceptance processes to ensure that significant glass recrystallization will not occur. Also, available data indicate that the radionuclide-release performance of waste glass is not sensitive to minor recrystallization. However, minor effects of recrystallization are subsumed in the degradation model because bounding parameters were derived directly from dissolution experiments. Therefore, omission of glass recrystallization will not significantly change the calculated expected annual dose.

TSPA Disposition: HLW glass recrystallization has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Product acceptance specifications and production controls will preclude significant glass recrystallization in the HLW glass waste forms that are planned for disposal of HLW at Yucca Mountain (DOE 1995). The specific product-acceptance specifications that are pertinent to glass recrystallization are the waste-form specifications: 1.1 Chemical Specification, 1.3 Specification of Product Consistency, and 1.4 Specification of Phase Stability.

Even if minor glass recrystallization were to occur during production and if this glass was accepted for disposal, available information indicates that the recrystallization would not significantly influence the waste-glass performance in the repository (Cunnane et al. 1994a, Volume 2, Section 2.2.2).

6.2.12 Pyrophoricity—YMP No. 2.1.02.08.00

YMP Primary FEP Description: DSNF can contain pyrophoric material. Pyrophoric material could ignite and produce an adverse effect on repository performance. Pyrophoric events could affect the thermal behavior of the system and could contribute to degradation of the waste package, waste form, and cladding.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.11.01.0	Heat output/ temperature in waste.

IRSR Issues: CLST3

Screening Argument: The effects of pyrophoric events can be excluded based on low consequence. Omission of pyrophoricity will not significantly change the calculated expected annual dose because an analysis given in the *Supplemental Discussion* shows that a pyrophoric event (PE) would have only a minimal impact on postclosure repository performance. In terms of the potential increase in total radionuclides released over the repository's lifetime, the impact of PEs would probably not exceed a 1% increase in the total amount of radionuclides released. In terms of the peak offsite dose that could result from a single PE, it was demonstrated that regardless of the model used, a PE would, at most, result in a 4% increase in peak offsite dose above the dose that would be obtained if PEs were not possible. As for clustered events, unrealistic scenarios involving incredible mechanisms would be required to generate more than a 4% increase in peak offsite dose. Therefore, pyrophoricity of DSNF can be excluded based on low consequence.

TSPA Disposition: The effects of pyrophoricity have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: For the purposes of this analysis, pyrophoricity is understood to be the capability to ignite spontaneously under temperature, chemical, or physical/mechanical conditions specific to the storage, handling, or transportation environment (ASTM C 1454-00 2000, pp.1-10), and a pyrophoric event (PE) is defined as ignition followed by rapid chemical oxidation or self-sustained burning. The probability of a pyrophoric event (PE) occurring needs to be addressed because a PE has the potential for increasing the release rate of radionuclides, which could impact repository performance. This analysis will examine such issues as: which types of SNF are a concern, the conditions required for a PE to occur, the impact on adjacent waste packages, and the impact a PE would have on repository performance.

The likelihood of a PE occurring is small. First, a sufficient ignition source must be present, and U-hydrides in N-reactor fuel may provide this source. Although hydrides observed so far are present only in small quantities (less than 2%), they tend to be concentrated near the exposed uranium metal fuel surface (Marschman et al. 1997). It has not been demonstrated that hydrides would not spontaneously ignite upon access to air, thereby initiating an event that would cause the bulk fuel to oxidize. Therefore, at least a small, finite chance of spontaneous ignition must be considered to be possible.

However, there also must be sufficient oxygen to support continuing combustion of the uranium metal after ignition for a full-scale PE to occur. The waste packages containing such potentially pyrophoric material as the N-reactor SNF will be filled with an inert gas prior to emplacement in the repository. The inert environment will exclude all but small quantities of water and oxygen, thus reducing the potential for oxidation reactions prior to breach of the waste package (DOE 1998d, Section 5.1.2.1). After that, waste-package breach oxygen would tend to be introduced into the waste package in a slow and controlled manner, given the identified waste-package failure modes (pitting and general corrosion). The slow introduction of oxygen would tend to limit the oxidation rate of any existing hydrides, thus limiting the ability of the hydrides to initiate a PE. However, there is not sufficient data available at this time to definitely exclude PEs based on their low probability of occurrence. Therefore, this analysis will presume that PEs are possible and estimates the impact they would have on repository performance if they were to occur.

The DOE Office of Environmental Management/Office of Civilian Radioactive Waste Management (EM/RW) Repository Task Team (DOE 1997, pp. 20-21) has addressed three fuel types with regard to their potential for pyrophoric reactions: oxide fuels, metal fuels, and carbide fuels. A fourth type (metallic sodium-bonded fuel) has been eliminated from further consideration because it is a listed RCRA hazard and it will be treated prior to repository emplacement. There are numerous varieties of the three identified fuel types. The characteristics and pyrophoric nature of these three types of fuel are discussed in the following paragraphs. These fuels are all DSNF and, thus, only a small part of the total waste inventory in the repository.

Oxide fuels consist of U oxide fuels, Th/U oxide fuels, and U/Pu oxide fuels of varying enrichments and concentrations. Bulk UO_2 fuels have been exposed in oxidizing environments at temperatures considerably higher than expected repository temperatures with no indication of ignition. It is not expected that the addition of significant amounts of ThO_2 or PuO_2 in the mixed oxide (MOX) fuels will make the fuel pyrophoric as long as the matrix form is composed of heavy-metal oxides (DOE 1997, p. 20).

Most metals will oxidize and can be pyrophoric when in a fine particulate form. Some metal fuels are also potentially pyrophoric when hydrides are present. The corrosion of uranium metal tends to form uranium hydrides as finely divided inclusions dispersed in the uranium metal. Although examination and testing of damaged N-reactor zirconium-clad U-metal fuel showed that there were only small amounts of uranium hydride formed by corrosion as precipitates within the metal and in thin coatings on the surfaces of internal cracks (Marschman et al. 1997, Section 3.4.2), the presence of these hydrides are believed to be responsible for the observed decreased ignition temperature of damaged/corroded N-reactor SNF samples. The possibility exists that additional U-hydride will form during interim storage (Reilly 1998, p. 30). This additional hydride could potentially act as an ignition source if concentrated in a small area.

TRIGA fuels, which are predominantly U-Zr-hydride, may not be as susceptible to pyrophoric reactions as uranium metal fuel because this fuel does not display the extensive damage/corrosion evident in the N-reactor SNF.

Most (i.e., SRS-located) aluminum-clad uranium metal DSNF does not have the amount of damage/corrosion (and consequent extent of hydriding and potential for pyrophoricity) nor the propensity to form uranium hydrides as a result of corrosion, shown by the N-reactor SNF (Lam et al. 1997, Section 8). The small fraction (e.g., SPR SNF) of the total aluminum-clad uranium-metal SNF inventory that is currently located in the K-Basins at Hanford could display damage/corrosion characteristics similar to N-reactor SNF due to its similar handling and wet-storage history. This Al-U SNF will be disposed in the N-reactor SNF canisters.

The U-Mo fuels should behave similarly to the Al-U SNF. The electrochemical potential for uranium-hydride formation is not favorable, so the uranium/aluminum alloy matrix fuel is much less reactive than U-metal (DOE 1997, p. 21). Because of the low U loading of these fuels (<9% of their total mass is U), any UH_3 would be present at less than 3 mole %.

Carbide fuels are not a pyrophoric hazard in a dry atmosphere except as high surface area powders. In a moist atmosphere (as would be required to corrode through a waste package), uranium carbide (UC_2) will corrode to form UO_2, carbon dioxide (CO_2), and hydrocarbon gases. The major constituent of the hydrocarbon gases is methane with minor constituents being ethane, ethylene, and acetylene. These gases are flammable in air, but only acetylene is autocatalytically explosive. An analysis of carbide fuels (Propp 1998) indicated that hydrocarbon production is not a problem. It should also be noted that less than 1% of the DSNF is carbide SNF.

The following considerations were used in this analysis to create a worst-case scenario. This worst-case scenario will then act as a bounding analysis to determine the maximum impact pyrophoricity can have on repository performance.

- A PE cannot occur until after both the waste-package and SNF canister have breached.

- There is sufficient oxygen available to support a PE.

- An ignition source is available.

- A PE would cause nearly instantaneous release of the soluble and volatile radionuclide inventory in the affected packages.

- A PE would cause the two adjacent waste-packages to breach.

- N-reactor SNF poses the greatest risk of pyrophoric behavior because it is composed of damaged and corroded uranium-metal clad in zirconium alloy; and this and similar types of uranium-metal-based spent fuels have displayed pyrophoric behavior in the past. Also, N-reactor SNF represents the overwhelming majority of the DSNF (85%) but only ~10% of the total waste to be stored at Yucca Mountain (CRWMS M&O 2000j, Tables I-5 and I-6)

- The N-reactor waste-packages will be evenly distributed throughout the repository.

- A PE will not propagate beyond the two adjacent waste packages.

A PE can impact a number of repository components: the affected waste-package itself (cladding, fuel and in-package chemistry), nearby waste-packages, and nearby geohydrology. It can be postulated that changes in local percolation rates could result from changes in the surrounding rock strata caused by a PE's thermal energy. However, modeling stipulated that waste packages must be breached prior to undergoing a PE. Therefore, changes in local percolation rates would not impact waste-package failure rates. Also, the complete dissolution of the waste form within a breached waste package is modeled to occur within one time step following its breach. Therefore, changes in local percolation rates would not significantly impact SNF dissolution rates. The use of titanium drip shields will tend to shield the surrounding geohydrologic system from any sudden release of thermal energy and distribute it along the drift.

The following argument addresses pyrophoricity in terms of both the total radionuclides that could be released due to a PE and the effect a PE could have on the peak offsite dose. A sensitivity analysis is included to evaluate the effects of clustering.

Impact on Radionuclide Release—Over the period of one million years following repository closure, some percentage of the approximately 160 N-reactor and 12,000 total waste-packages (CRWMS M&O 2000j, Table I-1) will fail. Adjacent waste packages equal to twice the number of failed N-reactor waste packages could potentially fail due to PEs. In reality, some portions of these adjacent waste packages may have failed prior to a nearby PE and another portion would have failed independently sometime later. Therefore, if (1) the various types of waste packages are evenly distributed throughout the repository and (2) the probability of failure of a waste package containing pyrophoric (i.e., N-reactor) SNF is the same as that of any other waste package, the increase in the total number of waste-package failures during the million-year period due to PEs can be calculated as follows:

$$N_{add} = N_{nr} * F_b * N_a * (1 - P_b)$$

$$F_{add} = N_{add} / N_{tot} \qquad \text{(Eq. 6)}$$

where,

N_{add} = number of additional breached waste packages

N_{nr} = number of N-reactor waste packages

F_b = fraction of waste packages breached

N_a = number of adjacent waste packages breached per-PE

F_{add} = fractional increase in waste packages breached due to PEs

Equation 6 predicts 1.1, 0.8, 0.5, 0.3 and 0 percent increases in the number of waste packages failing due to a PE for normal failure rates of 60, 70, 80, 90, and 100 percent, respectively. Because the various waste package types were considered to be evenly distributed, these results should correspond to the increase in the total amount of radionuclides released over the one-million-year period. It should be noted that the TSPA-VA (DOE 1998c, Section 4.1.7) analysis indicated that approximately 100% of the waste packages will be failed at 1,000,000 years and the Alternative-II (EDA-II) (CRWMS M&O 1999c) estimated that approximately 89% of the waste packages failed at 1,000,000 years. Also, the better the repository performs (i.e., the lower the percentage of failed waste packages at 1,000,000 years), the greater the impact PEs could have in term of percent increase in radionuclide release. However, the total repository release would also be significantly lower as repository performance increases.

Impact of PE on Peak Offsite Dose—The impact a single PE would have on peak offsite dose can be bounded by probabilistically assessing the potential worst-case release from a failed waste package, in terms of the potential percent increase in radionuclide release. It is independent of any particular repository model.

A pyrophoric event involving a single N-reactor waste package with the simultaneous failure of the two adjacent CSNF waste packages could be estimated to result in a peak offsite dose that is equivalent to approximately three times the value of a single waste-package failure.

In the event a waste package containing N-reactor SNF fails and subsequently undergoes a pyrophoric event, the initial release would be approximately three waste-package equivalents (WPEs) of radionuclides rather than one WPE. However, because whether or when a waste package fails is independent of the type of SNF contained in it, then the probability that any given failed waste package contains N-reactor SNF is equal to:

$$N_{nr}/N_{tot} = 0.013 \qquad \text{(Eq. 7)}$$

Therefore, the average increase in radionuclide release from the repository due to a PE involving a waste package containing N-reactor SNF can be represented as the probability that the failed waste package contains N-reactor SNF multiplied by the probability that that waste package undergoes a PE multiplied by three WPE. Even considering that every breach of a waste package containing N-reactor fuel results in a PE, the increase is:

$$(0.013 \text{ N-reactor WP/total WP})(1 \text{ PE per failed N-reactor WP})(3 \text{ WPEs}) \cong 0.04 \text{ WPEs} \qquad \text{(Eq. 8)}$$

This represents a 4% increase in peak offsite dose above that which would result if PEs were not possible. Because a PE event will most likely not occur following an N-reactor waste-package breach, the actual increase is probably much smaller.

Additionally, preliminary, postclosure site-boundary dose-sensitivity analyses using the RIP performance assessment code have indicated that the dose rate at the site boundary due to the release of the N-reactor SNF radionuclide inventory during one performance assessment time step is insensitive to the actual rate of release of the radionuclides from the waste package (CRWMS M&O 1999p). Thus, even the postulated instantaneous release due to a PE has little effect on overall repository-boundary, radionuclide-release performance.

Clustering Sensitivity Model—Clustering can be defined as multiple waste packages failing in a short time period. They can be postulated as being either induced by some initiating event that is not associated with pyrophoricity or DSNF (non-pyrophoric-induced cluster) or induced by an initiating PE that results in subsequent PEs (pyrophoric-induced cluster). Clustering events can be potentially important in that they could result in a higher peak offsite dose. Although no credible mechanisms have been identified that would result in a clustering of PEs, the impact a clustering event could have on peak offsite dose is addressed.

Regarding non-pyrophoric-induced clusters, an argument can be made similar to that in the preceding section. It does not matter how many waste packages are involved in some random event that results in clustering, each waste package involved has approximately 0.01 probability of being an N-reactor waste package, which might then result in a PE. Therefore, the maximum impact PEs could have on the peak dose associated with a non-pyrophoric induced clustering event would be a 4% increase in the dose resulting from that clustered event (see Eq. 8). This conservatively considers that the waste packages adjacent to the waste packages undergoing PEs were not failed during the clustering event. It also considers that none of the adjacent waste packages had failed at some previous time and that all the involved N-reactor waste-package failures result in a PE. (Because a PE event will most likely not occur following an N-reactor waste-package breach, the actual increase is probably much less.)

The PE-induced clustering event can be dismissed based on the expected separation that will exist between N-reactor waste packages. Also, there is probably insufficient oxygen available in a drift to support multiple PEs occurring simultaneously.

Conclusions—From the proceeding consequence-based analysis, it can be seen that a pyrophoric event involving single waste packages containing N-reactor SNF would have only a minimal impact on postclosure repository performance. In terms of the potential increase in total radionuclides released over the repository's lifetime, the impact of PEs would probably not exceed a 1% increase in the total amount of radionuclides released. In terms of the peak offsite dose that could result from a single PE, it was demonstrated that regardless of the model used a PE would, at most, have a 2% increase in peak offsite dose above the dose that would be obtained if PEs were not possible. As for clustered events, unrealistic scenarios involving incredible mechanisms would be required to generate more than a 2% increase in peak offsite dose. Therefore, pyrophoricity of DSNF can be excluded from further consideration with respect to its effect on repository postclosure performance in the FEP process.

6.2.13 Flammable Gas Generation from DSNF—YMP No. 2.1.02.29.00

YMP Primary FEP Description: Defense Spent Nuclear Fuel (DSNF) to be disposed of in Yucca Mountain will have a small percent of carbide fuel. When carbide is exposed to water, flammable gasses such as methane and its minor constituents ethane, ethylene, and acetylene (referred as ethyne by the International Union of Pure and Applied Chemistry–IUPAC) are generated. If these gases ignite, localized increases in temperature can occur, which might affect fuel degradation. The area around the ignition point may be mechanically and/or thermally perturbed, which could affect waste container or host-rock properties in the area of the EBS.

Screening Decision and Regulatory Basis: *Excluded* based on low-consequence.

Related Primary FEPS: None.

IRSR Issues: CLST3

Screening Argument: Flammable-gas generation from DSNF can be excluded based on low consequence. Omission of flammable-gas generation from DSNF will not significantly change the calculated expected annual dose because in the total repository, only 0.001% of the waste is carbide that can react with water to produce flammable gases after the waste packages and canisters are breached, and only if water flows in and contacts the spent carbide fuel. Such gas explosions from carbide fuels will not result in a significant release of radionuclides from the waste form. Consequently, there will not be a significant change in either time or magnitude of the expected annual dose.

DSNF has a small percent (~1%) of carbide fuels (DOE 1999b, p. D-2). The chemical reaction of U/Th carbide and water could produce flammable gases. Only DSNF from the Peach-Bottom Core 1 has a potential to generate flammable gases and given the small percent of carbide fuel in the total waste inventory, any consequences from a gas explosion would be low. Additionally, the repository is located in the unsaturated zone (UZ), which is overlaid by ~250 meters of welded and non-welded tuffs. The tuffaceous layers are sub-divided into units having abundant fractures and/or fracture networks; some units are highly fractured, and others have only moderate to minimal fracturing. With the exception of intermittent perched zones, both the matrix and the fractures within the overlying host rock have a relatively high degree of gas saturation (generally >50%) above and below the repository horizon, making these area more gas than liquid permeable. Within the repository horizon, under ambient conditions, gas saturation in the matrix ranges between 10-20%. Gas saturation in fractures is much higher. These conditions will promote a dispersive gas flowpath between the repository and host rock, thus diluting any potential flammable-gas concentrations to levels below the ignition point.

TSPA Disposition: Flammable-gas generation from DSNF has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: DSNF has a small percentage (~1%) of U/Th carbide fuels (DOE 1999b, p. D-2). Since DSNF comprises only about 4% of the total waste packages in the repository (DOE 1999b, Section 1), the carbide spent fuels will amount to less than 0.04% of the waste packages. The DNSF waste will be co-disposed in some 3910 waste packages (DOE 1999b; CRWMS M&O 2000j, Table I-1), with one DSNF canister packaged with five CSNF canisters in each waste package, so the carbide fuel is inherently dispersed.

The chemical reaction of U/Th carbide with water produces carbon dioxide and hydrocarbon gases. The major constituent of the hydrocarbon gases is methane, with minor constituents being ethane, ethylene, and acetylene. In sufficiently high concentrations, all of the hydrocarbon gases are flammable in air. Flammable gases, such as acetylene, are autocatalytically explosive; in a relatively pure state, it will decompose to carbon and hydrogen if it is compressed to several tens of MPa. Acetylene is used commercially under high pressure only if it is mixed with an inert gas and handled in equipment with a minimum of free volume; for example, acetylene for welding is dissolved under about 1.4 MPa pressure in acetone and contained in cylinders packed with diatomaceous earth. The repository is located in an unsaturated tuffaceous unit, the Topopah Spring welded tuff, that consists of numerous fracture networks. The Topopah Springs unit is over- and underlain by other tuffaceous unsaturated units. These units are sub-divided; some have abundant fractures and/or fracture networks while others have moderate to minimal fracturing. Mean fracture permeabilities range between 2.5×10^{-14} to 3×10^{-11} m^2; matrix permeabilities range between 2×10^{-19} to 2×10^{-12} m^2 (CRWMS M&O 2000an, Section 6). Both the matrix and the fractures above and below the repository horizon have a gas saturation that is generally greater than 50% (CRWMS M&O 2000ag, Section 3.6), making these areas more gas than liquid permeable. In the repository horizon, gas saturation in the matrix varies from 10 to 30% (CRWMS M&O 2000ag, Section 3.6); gas saturation and permeability is higher in fractures than in the matrix. Consequently, any gas produced in the repository will tend to flow through the fracture complex with some gas diverted to the matrix at unit interfaces or where fracture networks pinch out (CRWMS M&O 2000ag Section 3.3). These flowpaths will disperse flammable gas concentrations to levels below the ignition point. Additionally, because of this fracture network system, repository pressures will stay within only a few pascals of atmospheric pressure. Given these conditions, if DSNF canisters were to be breached, repository pressures would be similar to commercial safe-handling conditions for acetylene, that is, low pressures with high dilution conditions due to the presence of inert gases (nitrogen in the air, water vapor, and CO_2 and CH_4 from carbide reactions). An analysis of carbide fuels (Propp 1998) indicated that hydrocarbon production is not a problem.

The U/Th carbide was used to fuel gas-cooled reactors. For fuels used by the Fort St. Vrain and Peach-Bottom Core-2 Reactors, the fuel pellets are encased in tough silicon-carbide shells that are highly resistant to corrosion. These shells are designed to contain all gases produced within the pellets, and spent fuel pellets have demonstrated high integrity, with typical failure rates of 0.3% to 0.5% (Rechard 1995b, Section A.4.2, p. 48). However, spent-fuel pellets from the Peach-Bottom Core-1 had no protective SiC layer. As a result, many pellets have broken open, with an estimated cladding failure rate of 35% (DOE 1998d, Appendix A, Section 6.9, p. 41), thus opening the possibility that flammable hydrocarbons could be produced within DSNF waste packages after the waste package is breached and water contacts the fuel pellets. However, the amount of low-integrity Peach-Bottom Core 1 fuel is very small, about 1.663 MTHM (DOE 1998d, Appendix B, p. 14) contained in 103 waste packages (DOE 1998d, Table 1-1, p.1-8), compared with 24.667 MTHM of high-integrity fuel from Fort St. Vrain and Peach-Bottom Core 2 (DOE 1998d, Appendix B, p. 14) in 545 waste packages (DOE 1998d, Table 1-1, p.1-8). Thus, less than 7% [1.663/(1.663 + 24.667)] of the carbide fuel is contained in low-integrity pellets. This equates to about (0.35 x 7%) 2.5% of the carbide fuel that is readily susceptible to reaction with water. In the total repository, (2.5% x 0.04%) 0.001% of the waste is carbide that can react with water to produce flammable gases after the waste packages and canisters are breached and if water flows in and contacts the spent carbide fuel. If all of the Peach-Bottom Core 1 spent fuel were located together in one small area, the possibility exists that sufficient quantities of flammable gases could be generated to be a concern. However, this waste will be widely distributed over a large area of the repository and codisposed in 648 waste packages with CSNF, so the probability of generating problematical amounts of flammable gases is low.

A screening argument can be based on three mitigating factors: (1) a small quantity of spent fuel produced by the Peach-Bottom Core 1 and the minor increases in dose which it could possibly effect (consequence); (2) relatively good air circulation within the repository would disperse the flammable gases and dilute potentially autocatalytically explosive gases in the drifts (consequence); and (3) because of the high permeable set of stress-induced and *in situ* fractures and fracture networks that extend outward from the repository drift wall, thus providing multiple dispersive flowpaths away from the repository. There will not be sufficient pressure buildup to cause flammable gases to autocatalytically decompose.

6.2.14 Void Space (in Waste Package)—YMP No. 2.1.02.09.00

YMP Primary FEP Description: If waste packages and/or DSNF canisters are not completely filled, then the unfilled inert-gas or air-filled volume could influence water-chemistry calculations.

Screening Decision and Regulatory Basis: *Included* (the use of void space in determining radionuclide concentrations). *Excluded* based on low consequence (the use of void space in determining in-package chemistry).

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.03.00	HLW degradation
2.1.09.04.00	Radionuclide solubility, solubility limits
2.1.09.12.0	Rind (altered zone) formation

IRSR Issues: CLST4

Screening Argument: Void space volume is included in determining radionuclide concentrations as discussed in the *TSPA Disposition.*

The use of void space for determining in-package chemistry has been Excluded based on low consequence. Omission of the void space volume in determining in-package chemistry will not significantly change the calculated expected annual dose. The influence of void volume on the uncertainty in the in-package chemistry is small compared to variability in water inflow rates into the waste package used to define the uncertainty in the in-package chemistry.

For determining the radioisotope concentration, the Dissolved Radioisotope Concentration Component of the TSPA uses a time varying void volume of the altered waste form (see Solubility FEP 2.1.09.12.00). In addition, in the process model for evaluating the In-Package Chemistry Component of the TSPA, the mixing-cell volume is equal to the fixed void volume in the waste package; hence, void space is indirectly included in the In-Package Chemistry Component.

The potentially time-varying void space available and its influence on determining the in-package chemistry was excluded based on low consequence. Rather, the uncertainty in the in-package chemistry that might be caused by different void volumes is small compared to various waste-package water-inflow rates that were used to define the various simulations and, thereby, the uncertainty used in evaluating the In-Package Chemistry Component of the TSPA.

TSPA Disposition: The concept of an unfilled void volume is included in some aspects of the TSPA-SR/LA. To evaluate the chemistry of water in the package, the entire void volume in the waste package including any void volume in a canister is reasonably concluded to be saturated and is included in the process-model calculations. However, in the waste-form dissolution model for the TSPA-VA and TSPA-SR/LA, after waste-package failure, radioisotopes released from the waste form are dissolved (up to a solubility limit) in a fixed-volume mixing reservoir. For the solubility calculations, the waste package is reasonably concluded to be unsaturated; the volume of the mixing reservoir is then only the pore volume in the altered fuel (corrosion rind) around the unaltered fuel. (See YMP No. 2.1.09.12.00.) For details, see the In-Package Source Term AMR (CRWMS M&O 2000p).

The use of void space for determining in-package chemistry has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: In the process-model for evaluating the In-Package Chemistry Component of the TSPA, the mixing cell volume is equal to the entire void volume in the waste package. The influence of the void volume within the waste package and waste form is evaluated within these AMRs: *In-Package Source Term Abstraction* (CRWMS M&O 2000p) and *Summary of In-Package Chemistry for Waste Forms* (CRWMS M&O 2000m).

6.2.15 Cellulosic Degradation—YMP No. 2.1.02.10.00

YMP Primary FEP Description: Degradation of cellulose in the waste could affect the long-term performance of the disposal system.

Screening Decision and Regulatory Basis: *Excluded* based on low probability (not credible), (the degradation of cellulose in the waste).

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.10.01.00	Biologic activity
2.1.12.04.0	Gas generation from microbial degradation

IRSR Issues: None.

Screening Argument: Degradation of cellulose in the waste can be excluded from consideration in the TSPA based on low probability (not credible). According to *Waste Acceptance System Requirements* (DOE 1999a), organic materials will not be included as part of the waste in the Yucca Mountain potential repository. Therefore, omission of the degradation of the insignificant quantities of cellulosic material that might be present is excluded on the basis of low probability (not credible) because degradation of very small quantities of cellulosic materials has a very low probability of occurrence.

TSPA Disposition: Cellulose degradation has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: No cellulosic materials will be included or stored as part of the waste in the Yucca Mountain repository. The *Waste Acceptance System Requirements* (DOE 1999a) currently states that "The waste form shall not contain detectable amounts of organic materials."

If cellulose were included in the Yucca Mountain Project (YMP) waste, degradation of cellulose must affect the long-term performance of the disposal system with the production of H_2, CO_2, CH_4, H_2S, and N_2O. The significant effect of these gases from microbial degradation is discussed in YMP No. 2.1.02.12.00.

Furthermore, because no cellulose is permitted (DOE 1999a, p. 16), cellulosic degradation in waste form is not an issue for CSNF, DSNF, or HLW.

6.2.16 DSNF Cladding Degradation—YMP No. 2.1.02.25.00

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

Screening Decision and Regulatory Basis: *Excluded* based on low consequence (for DSNF cladding degradation [except for Naval SNF]).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.01.0	DSNF degradation

IRSR Issues: CLST3

Screening Argument: DSNF cladding degradation can be excluded based on low consequence. No credit is taken for the protective performance of DSNF cladding because more than 80 percent of the DSNF is from the N-Reactor and is in poor condition. As much as 50 percent of the cladding may be already perforated. Also, unzipping of the cladding can be conservatively neglected since the inventory is small. Inclusion of the protective performance of DSNF cladding in the TSPA-SR calculation would reduce interpreted doses. Therefore, omission of DSNF cladding degradation as a system-wide feature from the performance assessment models will not significantly change the interpretation of expected dose.

TSPA Disposition: DSNF cladding degradation has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: For approximately 80% of the DSNF that is N-Reactor SNF, the cladding will be significantly damaged at the time of emplacement in their canisters. Up to 50 percent may be already perforated (Rechard 1995a, Section 11.3.1, p. 11-24). The large effort required to characterize the condition of the DSNF cladding and to establish the effectiveness of the cladding as a barrier was not judged to be cost effective. Because the cladding integrity of most DSNF will not be extensively characterized, the TSPA takes no credit for radionuclide retardation by the cladding. Also, the TSPA takes no credit for canister integrity, i.e., once the waste package has leaked/failed, it is conservatively concluded that the DSNF is exposed to water/air. Because no credit will be taken for the cladding, mechanisms that might enhance cladding degradation do not impact the predicted consequences (CRWMS M&O 1999h, Section 3.5.3; DOE 1998c, Section 3.5.7, Fig. 5-39).

If no credit is taken for DSNF cladding, then it may be reasonably concluded that the DSNF will have an immediate influence on the waste-stream plume chemistry. However, because DSNF is such a small percent of the total waste, its influence will be negligible and dominated by the CSNF. For the DSNF it is suggested that it is completely available for mobilization in one time step based on the preliminary screening decision for YMP No. 2.1.02.01.00, "DSNF Degradation, Alteration, and Dissolution."

6.2.17 Various Features of the Approximately 250 DSNF Types and Grouping into Waste Categories—YMP No. 2.1.02.28.00

YMP Primary FEP Description: Several hundred distinct types of DOE-owned SNF (DSNF) may potentially be stored at YMP. These represent many more types than can viably be examined for their individual effect on the repository. A limited number of representative or bounding degradation models must be selected and/or abstracted. As a result, the effects on repository performance of the details related to the many distinct types of DSNF can not be evaluated.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Included (N-reactor fuel as a bounding degradation model for all DSNF types.

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory
2.1.02.01.00	DSNF degradation
2.1.01.02.00	Codisposal/Collocation
2.1.01.03.00	Heterogeneity of waste forms
2.1.09.04.0	Radionuclide solubility

IRSR Issues: TSPAI3, ENF3, CLST1, CLST3, CLST4

Screening Argument: Details of the many distinct DSNF types have been excluded from the TSPA modeling based on low consequence. Omission of the details for the many distinct DSNF fuel type will not significantly change the calculated expected annual dose because a bounding surrogate degradation model based on N-reactor fuel will be used in the TSPA analyses. See FEP 2.1.02.01.00 on DSNF degradation.

The analysis/model report (AMR) titled *DSNF and Other Waste Form Degradation Abstraction* (CRWMS M&O 2000e) documents an examination of the available data and information concerning dissolution kinetics of DSNF fuel matrices for the purpose of selecting and/or abstracting a limited number of representative or bounding degradation models for the TSPA.

There are no secondary FEPs associated with this primary FEP.

TSPA Disposition: In evaluating the Waste Form Degradation Component of the TSPA, the degradation rate of DSNF was set at a fixed rate that bounded all DSNF types. The DSNF inventory, however, is a composite of ~250 types of DSNF including Naval fuels (see Waste Inventory FEP, 2.1.01.01.00). Furthermore, the Pu ceramic waste form is included in the TSPA by including its inventory in the HLW category (CRWMS M&O 2000e, Section 6.3.12). The degradation rate was based on an evaluation of available corrosion data for uranium metal, uranium oxide, and the experimental oxidation data obtained recently for irradiated N-Reactor fuel, which makes up 90% of DSNF category. A value of 1.75 x 10^6 mg/m^2-d provides a conservative bound for the other available alteration data. This value is ten times the highest observed rate (CRWMS M&O 2000e, Section 6.3.7) in recent Pacific Northwest National Laboratory (PNNL) experimental studies for N-Reactor Uranium metal. See FEP 2.1.02.01.00 on DSNF degradation.

Supplemental Discussion: The analysis/model report (AMR) titled *DSNF and Other Waste Form Degradation* (CRWMS M&O 2000e) documents an examination of the available data/information concerning dissolution kinetics of DSNF fuel matrices for the purpose of selecting and/or abstracting a limited number of representative or bounding degradation models for the TSPA (DOE 2000). The descriptions and associated degradation models of DSNF fuels are summarized in Table 6. A more detailed summary of each grouping is provided below. These excerpts were taken from CRWMS M&O (2000e).

Group 1 Classified Naval DSNF—The naval SNF is similar to commercial LWR SNF in that it is composed of sintered uranium dioxide encased in zirconium alloy cladding.

Detailed descriptions of the configuration of this SNF or experimental studies or analyses of its dissolution behavior are not available. However, it can be expected that the dissolution behavior of this SNF will be qualitatively similar to that of the commercial LWR SNF. Appropriate confirmation of this condition will be provided by the naval SNF disposition program.

Group 2 U/Pu Alloy—There are several individual types of U/Pu alloy-based DSNF primarily comprised of U-Mo and U-Zr alloys, although smaller quantities of U-Th and U-Fe alloy are part of this group. These alloy fuels are generally clad in zirconium alloy, but some small quantities have aluminum, stainless steel, or tantalum alloy cladding. The largest single fuel types in this group, (90% by weight of uranium), is the zirconium clad U-Mo Fermi reactor SNF and stainless steel clad U-Zr alloy Annular Core Research Reactor SNF.

Studies of the dissolution behavior of U-Mo and U-Zr alloys give dissolution rates, which depend on the amount of alloying molybdenum and zirconium, but which generally show U-Mo dissolution rates to be slightly higher than U-metal and U-Zr alloy slightly lower than U-metal. From this behavior, it is recommended that the best-estimate degradation rate for the Group 2 fuels be taken as the model for U-metal under wet oxic conditions from DOE (1999b, Section 6.7) as follows:

$$R\ [\mathrm{kg/m^2\text{-}s}] = 1.88 \times 10^3\ \mathrm{EXP}\ (-7970/T_K) \qquad \text{(Eq. 2a)}$$

This model was derived from data in the open literature for the corrosion of unirradiated metallic uranium and uranium alloy.

Group 3 Pu-U Carbide SNF—Group 3 SNF consists primarily of fuel from the Fast Flux Test Facility (FFTF) with most of the balance from the Sodium Reactor Experiment (SRE). Both consist of mixed carbide fissile fuel particles in a non-graphite matrix.

Only a very limited amount of information concerning the chemical reactivity is available. Because of the lack of specific information concerning degradation behavior and the indication in DOE (1999b, Section 6.3) that the fissile particles could be very reactive, it is recommended that both the best-estimate and conservative-degradation-rate models be taken as 100X the dissolution rate of the uranium metal SNF.

Group 4 MOX and Pu Oxide SNF—Group 4 SNF is composed of a mixture of uranium and Pu oxides with various cladding materials. Although several dozen SNF types are in this category, the largest single type is the FFTF DFA/TFA fuel, contributing over 50% of the uranium by weight.

Since the fuel material is either uranium oxide or Pu oxide, the dissolution kinetics of the fuel form is not expected to be materially different than that for commercial LWR SNF. The best-estimate and conservative models recommended are those for the LWR SNF.

Group 5 U/Th Carbide SNF—This SNF group consists primarily of thorium or uranium carbide particles coated with pyrolytic carbon or silicon carbide embedded in a carbonaceous matrix. Over 90% by weight of MTHM of this group is Fort St. Vrain (FSV) fuel, with the remainder being Peach Bottom (PB) fuel. The PB fuel may be more damaged than the FSV fuel, although there is little qualified information concerning the condition of either.

Fuel in this group whose protective coatings and matrix are intact would be expected have dissolution kinetics similar to that of pure silicon carbide. Damaged with a damaged coating or matrix would be expected to have dissolution kinetics similar to uranium carbide. DOE (1999b, section 6.5) contains a comparison of the corrosion rate of silicon carbide with that of uranium metal and shows that the uranium metal corrosion model is about 10X that of pure silicon carbide. For these reasons, the recommended best-estimate degradation model is that for silicon carbide given in DOE (1999b), and the conservative rate model should be taken as 10X that for uranium metal.

Group 6 Th/U Oxide SNF—Thorium-uranium oxide spent fuels primarily consist of the Shippingport Light Water Breeder Reactor SNF with the remainder from the Dresden and ERR thorium-uranium oxide SNF. The Shippingport fuel was clad in zirconium alloy, and the Dresden and Experimental Research Reactor (ERR) fuels were clad in stainless steel. The thorium-uranium oxide fuel consisted of sintered pellets similar to commercial LWR fuel pellets.

Several reports discussed in DOE (1999b, Section 6.6) attest that the mixed thorium-uranium was more corrosion resistant than pure uranium dioxide, as much as five orders of magnitude more corrosion resistant. Since the dissolution of the thorium-uranium oxide was not specifically measured, the use of the ceramic synroc model is suggested as the best-estimate model and 1000X this as the conservative model.

Group 7 U-metal SNF—The zirconium-clad N-reactor SNF constitutes over 95% of this group with small quantities of aluminum-clad single pass reactor (SPR) and EBR-II metallic U SNF. A significant quantity of the N-reactor fuel is significantly and visibly damaged, and much of the rest could have small pinholes/cracks in the cladding.

Experimental studies have been conducted in the past on uranium metal and uranium metal alloy, and some of this work is summarized in DOE (1999b, Section 6.7). PNNL has conducted dissolution tests on samples of the N-reactor SNF, showing that, in general, the rates of dissolution somewhat exceed those of the unirradiated uranium metal or alloy.

The PNNL work (Gray and Einziger 1998) showed that there were two stages in the dissolution behavior of the N-reactor SNF samples tested, an initial Stage 1 rate and after an incubation period a faster Stage 2 dissolution rate. The Stage 1 dissolution resembled the congruent dissolution noted in the similar PNNL dissolution experiments on uranium dioxide-based LWR SNF samples, that is the uranium went directly into solution as the soluble uranyl species (UO_2^{++}) via the following reactions:

$$U + O_2 \rightarrow UO_2$$

$$UO_2 + H_2O + \frac{1}{2} O_2 \rightarrow UO_2^{++} + 2OH^- \qquad \text{(Eq. 9)}$$

This Stage 1 congruent dissolution of the matrix was correlated with the expression:

$$\log R\ (\mathrm{mg/m^2\text{-}d}) = 8.52 + 0.347 \log[CO_3^{--}] + 0.088\ \mathrm{pH} - 1929/T_K \qquad \text{(Eq. 10)}$$

where CO_3^{--} is the molar concentration of carbonate in the contacting solution. The flow-through test data that was used to generate this expression qualified as service condition tests under the testing/modeling logic of ASTM C 1174-97 (1998). Vapor phase hydration testing which qualified as characterization testing under the logic of ASTM C 1174-97 (1998) established that alteration phases could form on the U-metal SNF which could retard release and, thereby, demonstrated that Equation 10 represented a conservative maximum dissolution rate.

During Stage 2, the dissolution ceased to be congruent and the disintegration of some of the fuel matrix, which formed a sludge-like material primarily consisting of U_4O_9 and/or the mineral form Schoepite ($UO_3 \cdot 2H_2O$), was observed. It was also postulated that the Stage 2 dissolution may have been faster because it coincided with the depletion of dissolved oxygen and consequent formation of an anoxic condition in the contacting water, represented by the equation:

$$U + 2H_2O \rightarrow UO_2 + H_2 \qquad \text{(Eq. 11)}$$

The rates of dissolution that were observed for Stage 2 ranged from 10,000 mg/m^2-d at 25°C to 290,000 mg/m^2-d at 75°C for some samples. Stage 2 dissolution generally began around sixty days into the testing. Table 2 in CRWMS M&O 2000e gives values for the Stage 1 dissolution rate as calculated from the above rate expression and for the experimental Stage 2 dissolution rates of the PNNL study. It should be noted that the N-reactor SNF dissolution testing is still underway, and these results upon which this model was derived are preliminary.

In most of the cases where Stage 2 dissolution was initiated, it began around 60 days into the test, a very short period in terms of TSPA analyses. A significant fraction of the N-reactor fuel has been stored in a damaged condition under water at the K-basins at Hanford and the metallic uranium has been exposed to the water environment. Therefore, the degradation model for unirradiated and unexposed U-metal derived in DOE (1999b) and recommended for the Group 2 (U/Pu alloy), Group 3 (Pu/U carbide), and Group 10 (Unknown) SNF is not recommended for the N-reactor SNF which forms the basis for Group 7. Although the PNNL experimental data is limited, it indicates that Stage 2 dissolution kinetics should apply for the N-reactor SNF. Based on these analyses, it is recommended that the best-estimate degradation rate model for the U-metal SNF be taken as the highest observed rate in the PNNL studies or 1.75×10^5 mg/m^2-d, and the conservative value be taken as 10X this rate.

Group 8 Uranium Oxide SNF—TSPA Group 8 consists of uranium dioxide-based SNF removed from commercial LWRs or similar SNF from test reactors. About half of the total inventory of approximately 178 MTHM comes from the TMI-2 core and, therefore, is not intact fuel. The other half is substantially undamaged SNF from other commercial reactors.

The dissolution kinetics per unit area of the damaged fuel, such as the TMI rubble, would be expected to be similar to the kinetics of the uranium dioxide-based LWR pent fuel reported in CRWMS M&O (1999c) but with a substantially enhanced effective surface area for release.

$$\mathrm{Log}_{10} R\ (\mathrm{mg/m^2\text{-}d}) = 5.4628 + [-2446.3\ (1/T_K)] + [1.4956\ \log_{10}(CO_3^{--})] + [-1.7016\ \log_{10}(O_2)] + [0.2346\ \mathrm{pH}] + [-0.8151\ \log_{10} BU] + [392.12\ (\log_{10} O_2)\ (1/T_K)] + [780.42\ (\log_{10} BU)\ (1/T_K)] + [0.1736\ (\log_{10} BU)\ (\log_{10}(CO_3^{--}))] + [0.1770\ (\mathrm{log10}BU)\ (\log_{10} O_2)] + [-0.2694\ (\log_{10} BU)\ (\mathrm{pH})] + [-0.3367\ (\log_{10}(CO_3^{--}))^2] \qquad \text{(Eq. 12)}$$

Where T_K is the temperature in kelvins, CO_3^{--} is the molar concentration of carbonate ion in the liquid phase, BU is the spent fuel burnup in MWD/kgU, and O_2 is the oxygen partial pressure in atmospheres in the gas phase.

This model for the degradation rate of uranium dioxide spent fuel was derived from dissolution tests which qualified as service condition tests under the testing/modeling logic of ASTM C 1174-97 (1998). These tests represented the maximum forward dissolution reaction rate for the material, with no back reactions that would inhibit dissolution. The fact that the reaction represented the maximum rate was verified by the observations of congruent dissolution and the lack of precipitated alteration phases on the test specimens during the tests. Vapor phase hydration tests were also performed on sibling samples of the SNF, which served as characterization tests per the logic of ASTM C 1174-97 (1998) and which verified that alteration phases could form on the SNF, and these phases would inhibit the dissolution rate. Therefore, the testing overall demonstrated the conservatism inherent in using Equation 12 for the dissolution rate for uranium dioxide based SNF.

DOE (1999b, Section 6.8) suggests a surface area enhancement factor of 100 for a release model representing the Group 8 SNF. It is, therefore, recommended that the best-estimate degradation model for intact uranium dioxide-based Group 8 SNF be the same as the degradation model for commercial LWR SNF [equation (12)] and that the conservative degradation model for intact Group 8 fuel be 100X the best-estimate model. It is also recommended that both the best-estimate and conservative degradation models for the non-intact Group 8 SNF be 100X the intact fuel best-estimate model.

Group 9 Al-based SNF—This group consists of fuels based on a uranium aluminide, uranium silicide, or uranium oxide particle phase dispersed in a continuous aluminum alloy matrix. The fission product radionuclides remain in the dispersed phase; and, therefore, the dissolution of the dispersed phase material is the parameter most germane to the release of the radionuclides upon contact with groundwater. The dissolution rate of interest is expressed in terms of mgU/m^2-day.

Much of this spent fuel (~36%) is from foreign research reactor sources with the balance from domestic research reactors, such as the High Flux Irradiation Reactor (HFIR), the Advanced Test Reactor (ATR), and university research reactors (DOE 1999b, Appendix D).

SRS has conducted dissolution studies on SRS reactor spent fuel samples at 25°C in both J-13 and bicarbonate solutions and on unirradiated samples at 25°C and 90°C. The data showed dissolution rates for the irradiated SNF of 0.19-0.22 mgU/m^2-day in J-13 and 22-36 mgU/m^2-day in bicarbonate solution. The corrosion data for the unirradiated fuel samples showed that the dissolution rate at 90°C was approximately 10X the rate at 25°C. Since the groundwater chemistry at the time of waste-package failure is expected to be approximately that of the J-13 water, the J-13 data is recommended for the best-estimate degradation model. Since a potentially aggressive water condition would be that containing bicarbonate, the bicarbonate data is recommended as the basis for the conservative model.

Group 10 Unknown SNF—Group10 SNF consists of a small amount of uranium nitride SNF and fuel with unknown matrices. This group consists of only about 0.2% of the total inventory of DSNF in MHTM.

Because of the unknown fuel matrix material and the small volume, it is recommended that the degradation models for this group be based on the dissolution kinetics of unirradiated uranium metal, similar to the Group 2 SNF. Therefore, the recommended best-estimate and conservative models are the same as for the Group 2 SNF.

Group 11 U-Zr-Hx—TRIGA test reactor fuel comprise ~97% of the total Group 11 SNF inventory of ~1.6 MTHM (DOE 1999b, Appendix D). The TRIGA fuel consists of a dispersion of fine particles of metallic uranium dispersed in a zirconium hydride matrix (U-Zr-Hx). Uranium loadings varied from approximately 8 to 45 wt%.

Unirradiated U-Zr-Hz fuel has been shown to have good elevated temperature corrosion resistance. However, there is no known qualified data for the dissolution rate of this material in repository-relevant water and temperature conditions. For this reason the NSNFP DSNF information report (DOE 1999b, section 6.11) recommends that the dissolution rate for this DSNF form be taken as 0.1X the uranium oxide SNF dissolution rate. The low total inventory in MHTM of the U-Zr-Hx SNF makes the contribution of the degradation of this material insensitive to the degradation model for this waste form.

Immobilized Ceramic Pu Disposition Waste Form—The waste form for the immobilized Pu will be cold-pressed, titania-based pyrochlore ceramic disks containing approximately 10.5 wt% embedded Pu as PuO2. These disks will be stacked in stainless steel cans, which are in turn embedded inside a canister filled with a vitrified borosilicate filler glass similar to the HLW glass waste form.

Although the ceramic waste form is similar to synroc in that it is titania-based, there is no known dissolution rate data specifically for the ceramic. Dissolution rates for similar ceramic materials show dissolution rates 3 to 4 orders of magnitude lower than that for borosilicate glasses. In the absence of qualified dissolution data from which to derive a model, it is recommended that the HLW glass degradation model be used for the immobilized ceramic form. The conservative conditions used in abstracting this model are that (1) the Pu is dispersed uniformly in the glass filler material at the time of contact with water, (2) the surface area exposed to the water is equivalent to 21X the geometric cylindrical surface area of the glass, and (3) the Pu would be released congruently at the dissolution rate of the glass matrix.

The HLW glass dissolution models that should be used in the immobilized ceramic Pu disposition model are as follows:

$$R/S_{im}\ (g/m2\text{-}d) = k_{eff} \bullet 10^{\eta \bullet pH} \bullet \exp(-E_a/RT) \qquad \text{(Eq. 13)}$$

Where

S_{im} = the effective reacting surface area of the glass in which the ceramic disks are embedded when immersed in the groundwater and is taken as 21X the geometric surface area,

$$R = 0.00831\ kJ/mol \bullet K \qquad \text{(Eq. 14)}$$

for $pH < pH_m$;

$$\log_{10} k_{eff} = 8.3 \pm 0.8,\ \ \eta = -0.6 \pm 0.2, \text{ and } E_a = 58 \pm 15\ kJ/mol \qquad \text{(Eq. 15)}$$

for $pH > pH_m$;

$$\log_{10} k_{eff} = 6.6 \pm 0.8,\ \ \eta = 0.4 \pm 0.1, \text{ and } E_a = 80 \pm 5\ kJ/mol$$

and

$$pH_m = 1.7 + 1045/T_K$$

Table 6. DSNF Groupings (CRWMS M&O 2000e)

Group	Name	Source	Description
1	Classified Naval SNF	Naval SNF	The naval SNF is similar to commercial LWR SNF in that it is composed of sintered uranium dioxide encased in zirconium-alloy cladding.
2	U/Pu Alloy	The largest single fuel types in this group, (90% by weight of uranium), is the zirconium-clad U-Mo Fermi reactor SNF and stainless-steel clad U-Zr alloy Annular Core Research Reactor SNF.	Primarily comprised of U-Mo and U-Zr alloys, although smaller quantities of U-Th and U-Fe alloy are part of this group. These alloy fuels are generally clad in zirconium alloy, but some small quantities have aluminum, stainless-steel, or tantalum-alloy cladding.

Table 6. DSNF Groupings (CRWMS M&O 2000e) (Continued)

Group	Name	Source	Description
3	Pu-U Carbide	Primarily from the Fast Flux Test Facility (FFTF) with most of the balance from the Sodium Reactor Experiment (SRE).	Consist of mixed-carbide fissile-fuel particles in a non-graphite matrix.
4	MOX and Pu Oxide	Although several dozen SNF types are in this category, the largest single type is the FFTF DFA/TFA fuel, contributing over 50% of the uranium by weight.	Composed of a mixture of uranium and Pu oxides with various cladding materials.
5	U/Th Carbide	Over 90% by weight of MTHM of this group is Fort St. Vrain (FSV) fuel, with the remainder being Peach Bottom (PB) fuel. The PB fuel may be more damaged than the FSV fuel.	Primarily thorium- or uranium-carbide particles coated with pyrolytic carbon or silicon carbide embedded in a carbonaceous matrix.
6	Th/U Oxide	Shippingport Light Water Breeder Reactor SNF with the remainder from the Dresden and ERR thorium-uranium oxide SNF. The Shippingport fuel was clad in zirconium alloy, and the Dresden and Experimental Research Reactor (ERR) fuels were clad in stainless steel.	Thorium-uranium oxide spent fuels. The thorium-uranium oxide fuel consisted of sintered pellets similar to commercial LWR fuel pellets.
7	U Metal	N-reactor SNF constitutes over 95% of this group with small quantities of Single Pass Reactor (SPR) and EBR-II	The zirconium clad (N-Reactor) or aluminum-clad (SPR and EBR-II) metallic U SNF.
8	U Oxide	About half of the total inventory of approximately 178 MTHM comes from the TMI-2 core and, therefore, is not intact fuel. The other half is substantially undamaged SNF from other commercial reactors.	Uranium dioxide-based SNF removed from commercial LWRs or similar SNF from test reactors.
9	Al based	Much of this spent fuel (~36%) is from foreign research reactors with the balance from domestic research reactors such as the High Flux Irradiation Reactor (HFIR), the Advanced Test Reactor (ATR), and university research reactors.	Fuels based on a uranium-aluminide, uranium-silicide, or uranium-oxide particle phase dispersed in a continuous aluminum-alloy matrix.
10	Unknown		Small amount (0.2% of the total inventory of DSNF in MHTM) of uranium-nitride SNF and fuel with unknown matrices.
11	U-Zr-Hx	TRIGA test-reactor fuel comprise ~97% of the total Group 11 SNF inventory of ~1.6 MTHM.	Consists of a dispersion of fine particles of metallic uranium dispersed in a zirconium-hydride matrix (U-Zr-Hx). The loadings varied from approximately 8 to 45 wt%.
12	Immobilized Ceramic Pu		Cold-pressed, titania-based pyrochlore ceramic disks containing approximately 10.5 wt% embedded Pu as PuO2. These disks will be stacked in stainless-steel cans, which will be in turn embedded inside a canister filled with a vitrified borosilicate filler glass similar to the HLW glass waste form.

6.2.18 Internal Corrosion of Waste Container—YMP No. 2.1.03.06.00

YMP Primary FEP Description: Aggressive chemical conditions within the waste package for the different waste forms, including CSNF and DSNF, could contribute to corrosion from the inside out.

Screening Decision and Regulatory Basis: *Included* (the effects internal corrosion of high-sulfur steel within waste packages on the in-package chemistry after waste-package breach).

Excluded based on low consequence (internal corrosion of waste package prior to waste-package breach).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.03.01.0	Corrosion of waste containers

IRSR Issues: TSPAI3, ENF3, CLST1

Screening Argument: Internal corrosion of waste packages prior to waste-package breach can be excluded based on low consequence because only very minor corrosion can occur within the inert environment of an unbreached package. Therefore, omission of very minor internal corrosion of the waste package prior to breach will not significantly change the calculated expected annual dose. Radiolysis products could contribute to the interior corrosion, but they have been excluded based on low consequence. See FEP 2.1.13.01.00.

In general, no credit will be taken in the TSPA for DSNF inner canisters as a barrier to fuel degradation and radionuclide mobilization. They will be constructed of stainless steel, which degrades relatively quickly once the outer waste package fails. Inclusion of DSNF canisters as a barrier to fuel degradation and radionuclide mobilization in the TSPA-SR calculation would reduce interpreted doses. Therefore, omission of this FEP as a system wide feature from the performance assessment models will not significantly change the interpretation of expected dose.

Even though not included as a barrier, the effects on internal corrosion of high-sulfur steel within waste containers on the in-package chemistry after waste-package breach has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: The influence of internal corrosion of canisters after waste-package breach is included as part of In-Package Chemistry Component of the TSPA (CRWMS M&O 2000i; CRWMS M&O 2000m).

Internal corrosion of waste packages prior to waste-package breach and internal corrosion of DSNF inner canisters have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: The waste packages are planned to be filled with the inert gas, helium, prior to emplacement in the repository (DOE 1998b, Section 5.1.2.1). The inert helium environment will displace water and oxygen, thus not allowing or at least greatly reducing the likelihood that corrosive chemical reactions will take place. Additionally, limited water is allowed in the CSNF and HLW-glass waste packages. Some DSNF-containing waste packages, such as those containing N-reactor fuel, may have more water, but this water would be scavenged by the waste form (Gray and Einziger 1998, Section 4). Other DSNF forms would have low water content. Consequently, prior to the breach of the waste packages, there should be no or minimum corrosion in the CSNF, DSNF, and HLW waste packages from normal chemical and water-vapor mechanisms (CRWMS M&O 1999h).

Once the waste package has been breached and if water enters the package, the CSNF will dominate the degradation process due to its having the majority of the volume. The DSNF is not expected to affect the waste-package degradation phenomena identified in these FEPs. The waste packages for CSNF and DSNF will be essentially identical in design (DOE 1998b). Also, the DSNF will be placed in canisters that initially will isolate the DSNF from the waste package. These canisters will not degrade and fail until the waste package has failed, but no credit is taken for canister except for HICs.

After breach of the waste package, the corrosion of inner structural stainless steel is important to determining the in-package chemistry and, thus, will be included in the process model and as part of the uncertainty of the in-package chemistry.

In general, no credit will be taken in the TSPA for DSNF canisters (within the waste package) as a barrier to fuel degradation and radionuclide mobilization. This decision was made because the canisters will be constructed of stainless steel, which will degrade relatively quickly once the waste package fails. Because no credit will be taken for the canisters, mechanisms that might enhance canister degradation do not impact the predicted consequences from the DSNF (CRWMS M&O 1999h, Section 5.5.7, Figure 5-39).

6.2.19 Pathways for Unsaturated Flow and Transport in the Waste and EBS—YMP No. 2.1.08.07.00

YMP Primary FEP Description: Unsaturated flow and radionuclide transport may occur along preferential pathways in the waste form and EBS. Physical and chemical properties of the EBS and waste form, in both intact and degraded states, should be considered in evaluating pathways.

Screening Decision and Regulatory Basis: *Included* (pathways for unsaturated flow and transport in the waste and EBS).

Excluded based on low consequence (preferential pathways within the waste package).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs: None.

IRSR Issues: TSPAI3

Screening Argument: Pathways for unsaturated flow and transport in the waste and EBS are included in some aspects of the TSPA as discussed in the *TSPA Disposition* for this FEP.

Preferential pathways within the waste package are excluded based on low consequence. Inclusion of preferential pathways within the waste package would reduce interpreted doses by increasing travel times and adding diffusive processes. Therefore, omission of preferential pathways within the waste package as a system-wide feature from the performance assessment models will not significantly change the interpretation of the calculated expected annual dose.

The flow in the waste form and EBS is modeled as a one-dimensional pathway with advective and diffusive transport of radionuclides. The use of a one-dimensional model is a preferential pathway, in comparison to the complex two- or three-dimensional geometry of the waste form and the EBS. In addition, the reasonable conclusions for the one-dimensional model include a conservatism that minimizes travel times and/or flow resistance along the pathway. These conclusions result in the single-mixing-cell approximation for the waste form and instantaneous flow across the gap between the waste package and the invert. In particular the container is considered to be a mixing cell for In-Package Chemistry Component. Inclusion of the pathways would delay release of radionuclides and, thus, they are conservatively neglected.

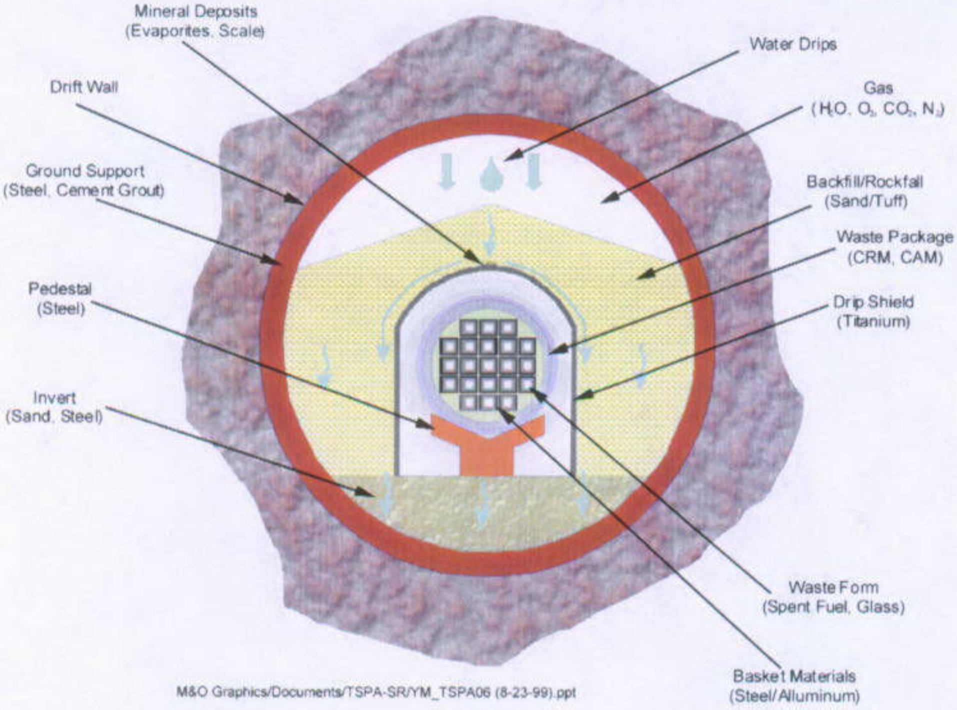
The drip shield and its surrounding backfill can form a preferential pathway if adjacent drip shields separate during the lifetime of the repository. The possibility of enhanced flow through a separation between adjacent drip shields is included in the computational models for corrosion of the drip shield and waste package and is included in the fluid pathways through the EBS.

A second element of the EBS, the invert, could have a preferential flowpath if the invert fractures. However, this possibility is excluded from the TSPA-SA and TSPA-LA models for two reasons: (1) the invert is filled with crushed tuff that is not expected to form and sustain a discrete fracture or pathway, and (2) the invert represents a minimal flow barrier in comparison to other elements of the system. Adding a preferential pathway to a minimal barrier will have negligible impact on total system performance.

TSPA Disposition: The waste form and EBS are represented as a one-dimensional pathway with diffusive and advective transport of radionuclides. This pathway is based on a single mixing cell for the waste form and a single cell that spans the depth of the invert. The invert is reasonably concluded to be a uniform porous medium with zero sorption and no fractures.

The EBS also includes the drip shield, the waste package, and the quartz sand backfill surrounding the drip shield. See Figure 4 in this AMR. The response of the sand backfill and drip shield is included in the WAPDEG calculations for corrosion of the drip shield and waste package as a function of the time-dependent seepage fluxes and groundwater chemistry (CRWMS M&O 2000x). The response of the sand backfill and drip shield is also included in the fluid-flow calculations; whereby, drip-shield separation or drip shield corrosion can open convective flowpaths onto the waste package. See the EBS flow and transport abstraction in CRWMS M&O (2000f).

Preferential pathways within the waste package and waste form have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.



M&O Graphics/Documents/TSPA-SR/YM_TSPA06 (8-23-99).ppt

NOTE: CRWMS M&O (1999h, Fig. 3.3-1)

Figure 4. Schematic of Drift Conditions for Engineered Barrier System (EBS)

Supplemental Discussion: The one-dimensional model for the waste form and EBS includes several conservative, reasonable conclusions that are equivalent to the enhanced flow associated with preferential pathways:

- The waste form and invert are represented by a one-dimensional flow-and-transport model. Potential two- or three-dimensional effects from the complex geometry of the waste form, waste package, gap, and invert are ignored by the one-dimensional model.

- The waste form is represented by a mixing cell. With a mixing cell, any seepage into the waste package has instantaneous access to and equilibrium with all of the available waste form. In effect, the mixing cell is a "bathtub" that provides zero flow resistance within the waste form.

- The presence of a gap between waste package and invert is ignored in the model. This approach is equivalent to instantaneous transport for fluid exiting the waste form and entering the invert. This approach also enhances transport out of the waste package and into the invert because the direct fluid contact between package and invert can support a diffusion process.

The drip shield and its surrounding quartz-sand backfill is a major flow barrier that prevents early contact of groundwater with the waste package. The drip shield is constructed from mailbox-shaped segments that will have an approximately 10-cm overlap. The adjacent segments may also be fastened or bolted together.

Separation of adjacent drip-shield segments after repository closure could provide a preferential pathway for groundwater to drip directly onto the waste package. Separation could occur because of seismic activity, thermal expansion, or rockfall. In the event the drip shield does separate, a new pathway is formed that can allow water to contact the waste package long before corrosion of the drip shield produces a through penetration. This pathway is represented in WAPDEG by the possibility that a drip-shield separation event can result in groundwater contacting one or several waste packages beneath the region of separation. This pathway is also represented in the EBS flow-and-transport module of the performance-assessment model as an increase in the cross-sectional area of the flow pathway after the separation event (CRWMS M&O 2000f).

The invert might also have a preferential pathway if it were to fracture during the lifetime of the repository. However, discrete fractures are not expected to form in the quartz-sand or crushed-tuff backfill that fills the invert. If a gap does form in the tuff, it is probable that the tuff will slump into the void, refilling the gap.

Even if a discrete fracture is formed and sustained in the invert, its impact on performance will be negligible because the invert is a minimal barrier to flow and transport. First, the invert has the relatively high permeability of a highly porous backfill. Second, the flow distance through the invert is on the order of 0.6 meters (CRWMS M&O 1999h). Third, the invert material is reasonably concluded to provide no sorption. Given these factors, the flow resistance of the intact invert will be minimal in comparison to other barriers, such as the unsaturated zone. It follows that adding a preferential pathway to a minimal flow barrier can be ignored because it will have negligible impact on total system performance.

6.2.20 Waste-Form and Backfill Consolidation—YMP No. 2.1.08.15.00

YMP Secondary FEP Description: Physical and chemical degradation of the drip shield, invert, backfill, waste form, and their containers will cause collapse and settlement within the repository. This consolidation may affect the development of the chemical environment and, therefore, the radionuclide transport out of the EBS.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Related Primary FEPs:

YMP No.	FEP Name
2.1.07.01.00	Rockfall(large block)

IRSR Issues: TSPAI3, CLST1, CLST4

Screening Argument: Waste-form and backfill consolidation are excluded based on low consequence. Omission of waste-form and backfill consolidation in the TSPA-SR calculation would reduce interpreted doses because consideration would tend to decrease the available reactive surface area (pore area) and permeability of the waste and would, therefore, reduce the calculated dose. Therefore, omission of waste form consolidation as a system-wide feature from the performance assessment models will not significantly change the interpretation of expected dose. The potential deleterious effect of maintaining water contact with the waste form is already conservatively bounded by allowing the waste package contents are fully accessible once the waste package has been breached.

Without the backfill, consolidation in the EBS, i.e., repository collapse, may occur at a slightly different rate depending on the structural response of the drift roof and walls due to different thermal behavior compared with the structural response expected with backfill being present.

TSPA Disposition: Waste-form and backfill consolidation have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Inconsequential gravity-induced settlement of a backfill is expected. Rockfall and collapse of the drift walls are expected to enlarge the drifts and fill them with relatively coarse rubble, covering and possibly damaging containers. Rockfall is at least partially self-supporting, so consolidation by transfer of some fraction of lithostatic pressure to the containers is self-limiting. Repeated seismic events are expected and will increase the degree of compaction and consolidation.

During the first 10,000 years, both the drip shield and the waste package itself will likely remain intact and structurally capable of withstanding any rockfalls without damage beyond inconsequential denting or deformation (CRWMS M&O 2000f, Section 6.5.3). Rockfalls are not expected to move or damage the drip shield enough even to create gaps that would allow backfill to flow through into the space between the drip shield and the waste package (CRWMS M&O 2000f). In the unlikely event that a prematurely weakened drip shield did collapse as a result of a rockfall, the waste package itself has sufficient structural strength to support the rockfall without breaching. The probability is remote, at best, that a large rockfall would occur on a prematurely weakened drip shield covering a waste package that has experienced such extensive juvenile failure that the container has lost structural integrity.

At later times, when the drip shield and waste package may have been breached as a result of corrosion, rockfalls and collapse of the drift walls may cause the waste packages and waste forms to collapse. However, because the waste package has already been breached, further collapse or structural changes within the waste package will only serve to continue processes, such as dissolution of radionuclides, that are already in progress. It is conservatively concluded that these processes occur unimpeded once the waste package is breached, so further changes to the physical condition of the waste package or waste form would have no additional impact.

Waste form collapse due to the corrosion of the containers and internal fuel supports, and degradation of the waste form itself, are anticipated processes. These are expected to be of no consequence because breaching of the waste package is already conservatively concluded to allow complete accessibility to the full contents of the waste package. In reality, collapse of the waste package or waste form and consolidation aided by seismic events may have some slight beneficial consequence by reducing the porosity and permeability of the waste, thereby impeding dissolution and transport of radionuclides. However, for this to occur, the containers and contents would need to have lost so much structural integrity that a large fraction of the contents must already have degraded and been lost to dissolution processes, so no credit is taken for this scenario.

For a repository design that includes a backfill surrounding a drip shield, the backfill will not come into contact with the containers or waste form until the drip shield fails due to corrosion from contact with the water. Both the use of backfill and any consolidation would tend to decrease the available reactive surface area and the area of the pore space in the backfill. The potential deleterious effect of maintaining water contact with the waste form is already conservatively bounded by concluding that the waste package is fully flooded with water (CRWMS M&O 2000m, Section 3.1). The amount of water available for in-package chemistry is determined from the influx of water through the upper part of the drifts and from wicking into the backfill (CRWMS M&O 2000f, Section 6.1.1) which must then pass through breaches in the waste package. A significant part of the drift walls is in contact with the backfill, and therefore a significant amount of the water in the EBS can come from wicking. Wicking is influenced by the porosity and saturation of the backfill and could, therefore, be affected by consolidation. However, the minor consolidation of the backfill is not expected to significantly influence the influx of water.

Without the backfill, consolidation in the EBS may occur at a slightly different rate, depending on the structural response of the drift roof and walls due to different thermal behavior compared with backfill being present. The drip shield will be slightly more susceptible to damage from rockfalls without the cushioning and support from backfill. However, it should still hold up to any anticipated rockfall (CRWMS M&O 2000f, Section 6.5.3) until it has been weakened by corrosion; it will take at least 50,000 years just for a substantial number of penetrations to occur in the drip shield (CRWMS M&O 2000f, Section 6.5.4), so structural failure is not expected until long after that. The fine-grained sand composing the backfill may accelerate consolidation by filling in pore space more readily. However, in the absence of backfill, rockfalls may contribute more to compaction. Thus, waste-form consolidation will be roughly the same with or without backfill as will radionuclide transport out of the EBS.

6.2.21 Induced Hydrological Changes in the Waste and EBS—YMP No. 2.1.08.08.00

YMP Primary FEP Description: Thermal, chemical, and mechanical processes may induce changes in the hydrologic properties and behavior of the waste and EBS.

Screening Decision and Regulatory Basis: *Included* (changes in flow areas for the waste package and drip shield, changes in flow volume into the waste package, and changes in exposed CSNF fuel-surface area available for seepage).

Excluded based on low consequence (changes to hydrological properties for the waste form and changes to hydrological properties for the invert).

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.01.00	DSNF degradation

2.1.02.02.00	CSNF degradation
2.1.02.03.00	HLW degradation

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: Changes in flow areas for the waste package and drip shield, changes in flow volume into the waste package, and changes in exposed CSNF fuel-surface area available for seepage have been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

The impact of thermal, chemical, and mechanical effects on hydrological properties of the waste form is excluded from the EBS model for the TSPA because the changes in hydrologic properties are small, relative to the overall uncertainty of the in-package chemistry and waste-form-degradation predictions. Therefore, changes to hydrological properties of the waste form have been excluded based on low consequence and omission of induced hydrological properties of the waste form that will not significantly change the calculated expected annual dose. For details see the CSNF, HLW and DSNF degradation AMRs (CRWMS M&O 2000c; 2000d; 2000e)

Changes to hydrological properties for the invert have been excluded based on low consequence. Omission of induced hydrological properties of the invert will not significantly change the calculated expected annual dose because the invert is anticipated to be a minimal barrier to flow and transport because (1) the backfill material will have relatively high permeability, (2) the flow distance through the invert is on the order of 0.5 meters, and (3) the invert backfill is reasonably determined to have no sorption for any radionuclide (CRWMS M&O 2000f).

Changes in exposed fuel-surface area as CSNF cladding fails is discussed in *Clad Degradation–Summary and Abstraction* (CRWMS M&O 2000w).

TSPA Disposition: Thermal, chemical, and mechanical processes are considered in modeling the response of the waste form, waste package, and drip shield. The specific processes that affect these elements of the EBS follow.

- Thermal effects can influence the physical state of the waste. The physical state of the waste, specifically the exposed surface area of fuel pellets and vitrified glass, is considered in determining dissolution rates for CSNF, HLW, and DSNF (CRWMS M&O 2000c; 2000d; 2000e).

- Thermal effects (temperature and relative humidity) are directly included in determining corrosion and associated areas for flow through the drip shield and waste package.

- Groundwater chemistry, including pH and concentration of specific species, is considered in determining dissolution rates for CSNF, HLW, and DSNF; in determining the solubility limits for radionuclides; and in determining the corrosion rates and associated areas for flow through the drip shield and waste package (CRWMS M&O 2000i; 2000m).

- Mechanical effects include rockfall and unzipping of cladding. The effects of rockfall are included in determining the time-dependent failure of waste packages and exposure of fuel pellets. Unzipping of cladding is also included as breaching mechanism that exposes the fuel waste (CRWMS M&O 2000w).

Induced hydrological changes to the waste form and invert have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Thermal, chemical, and mechanical processes are considered in modeling the response of the waste form, waste package, and drip shield. Thermal parameters (temperature and relative humidity) are directly included in predicting corrosion and associated flow areas for the drip shield and the waste package. However, thermal effects in the repository are not anticipated to further change the physical properties of the waste because spent fuel experiences much greater extremes of temperature in the reactor than in the repository. Also, vitrified waste is formed at much higher temperatures than will occur in the repository. Chemical effects are also directly included in predicting corrosion rates and associated flow areas for the drip shield and waste package. The mechanical effect of rockfall is included in determining the integrity of waste packages and fuel rods, hence the exposure of the waste form to groundwater seepage.

The effect of induced hydrological changes on the invert is excluded from the TSPA. In general, the invert is filled with a quartz-sand or crushed-tuff backfill that will be relatively inert in the repository environment. In addition, the impact of any changes induced in the invert will have a negligible impact on repository performance because the invert is a minor barrier in comparison to other elements of the repository system. The invert is anticipated to be a minimal barrier to flow and transport because (1) it will have the relatively high permeability of sand, (2) the flow distance through the invert is on the order of 0.5 meters, and (3) there will be no sorption for any radionuclide expected to be released.

Precipitation of secondary mineral phases into the pore spaces of the invert backfill material could alter its mechanical strength and permeability. However, the impact of these changes on total-system performance will be negligible because the invert is a minor barrier in comparison to other elements of the repository system. The invert is anticipated to be a minimal barrier to flow and transport because (1) the backfill material will have relatively high permeability, (2) the flow distance through the invert is on the order of 0.5 meters, and (3) the invert backfill is reasonably determined to have no sorption for any radionuclide (CRWMS M&O 2000f).

Mechanical response, specifically collapse of the supports for the waste package, could result in movement of the sand and a reduction in thickness of the invert. Again, the impact of this type of change on total system performance will be negligible because the invert is a minor barrier in comparison to other elements of the repository system, such as the unsaturated zone or the waste package.

Also note that any potential changes in invert permeability induced by the chemistry of the waste-package/waste-form leachate are likely to reduce its value from that for quartz sand, so the EBS model for the invert is conservative because it maximizes contaminant transport to the unsaturated zone.

6.2.22 Desaturation/Dewatering of the Repository—YMP No. 2.1.08.10.00

YMP Primary FEP Description: Decreases in the water content of the EBS occur because of ventilation and thermal affects.

Screening Decision: *Included* (desaturation/dewatering of the repository). Related Primary FEP: See Attachment V for the related secondary FEPs.

Related Primary FEPs:

YMP No.	FEP Name
2.1.08.11.00	Resaturation of Repository

IRSR Issues: TSPAI3

Screening Argument: Desaturation and dewatering of repository rock due to thermal effects and ventilation effects will affect two-phase flow between the host rock and the EBS. The flow of water between the host rock, drift, and EBS affects transport of any dissolved radionuclides through the unsaturated zone.

Desaturation/dewatering of the repository has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: Desaturation/dewatering of the repository rock due to thermal effects is included in the TSPA thermal hydrologic models. The water removed from the drift rock due to ventilation or construction is not included as an initial condition to the thermal-hydrologic simulations. However, it is considered to be bounding expected behavior, because in the simulation, water can return to the drift wall more quickly and can result in earlier corrosion of the waste package. Additionally, water can be present to transport any dissolved radionuclides through the unsaturated zone. For details see the models described in the following AMRs: *Mountain-Scale Coupled Processes (TH) Models* (CRWMS M&O 2000af), *Multiscale Thermo-Hydrologic Model* (CRWMS M&O 2000t), *Drift-Scale Coupled Processes (DST and THC Seepage) Models* (CRWMS M&O 2000u), and *Drift Scale Thermal Analysis* (CRWMS M&O 2000v).

Supplemental Discussion: The operational period of the repository will see a lowering of liquid saturation levels in the repository near field. Driven dominantly by ventilation, such dewatering will tend to delay the onset of water-induced waste degradation. Any such effect will be substantially overshadowed during the early thermal portion of the post-operational period. In response to waste heat, vapor movements will rapidly redistribute both water and vapor, thus removing relic effects due to dewatering.

6.2.23 Properties of the Potential Carrier Plume in the Waste and EBS—YMP No. 2.1.09.01.00

YMP Primary FEP Description: When flow in the drifts is reestablished following the peak thermal period, water may have chemical characteristics influenced by the near-field host rock and EBS. Specifically, the water chemistry pH and dissolved species in the groundwater may be affected by interactions with cementitious materials or steel used in the disposal region. These point source contaminated waters may coalesce to form a larger volume of contaminated water. This altered groundwater is referred to as the carrier plume because dissolution and transport will occur in this altered chemical environment as contaminants move through the waste, EBS, and down into the unsaturated zone. (Note: there is no defining limit as to what volume of contaminated water constitutes a plume.)

Screening Decision and Regulatory Basis: *Included* (potential effects of carrier plume on the water chemistry).

Excluded based on low consequence (the changing properties of incoming water, as evaluated by EBS).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.06.01.00	Degradation of cementitious material
2.2.08.12.00	J-13 well water as surrogate for incoming water
2.2.07.11.00	Return flow from condensation cap
2.2.08.02.00	Radionuclide transport occurs in a carrier plume

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Potential effects of the carrier plume on the water chemistry has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

The changing properties of incoming water, as evaluated by EBS, have been excluded from the TSPA based on low consequence. Omission of the changes in properties of incoming water will not significantly change the calculated expected annual dose because the range of uncertainty used for the in-package chemistry is not influenced by the changes of the initial chemical properties of the water entering the package. That is, the changes in the chemical properties of the water that occurs within the waste package are much larger than the variation in initial chemical properties of incoming water.

TSPA Disposition: Various corrosion rates of steel are used in the process model for in-package chemistry to establish an uncertainty band on the in-package chemistry. This calculated uncertainty band is then sampled in the In-Package Chemistry Component of the Waste Form Degradation Model (CRWMS M&O 2000n). Although the changing properties of the incoming water as evaluated by EBS are not coupled to the waste-form process calculations, slight changes in the incoming water, over time, are swamped by the dramatic changes that are predicted in the in-package chemistry because of the waste and internal parts of the waste package. That is, the range of uncertainty used for the In-Package Chemistry Component is not changed by the changes of the initial chemistry of the water entering the package, and so, because of the proximity of the structural steel of the waste package, only the waste-package steel is used to establish the uncertainty band. Therefore, water entering the waste package has the chemical properties of J-13 well water in the process models of the in-package chemistry (see FEP

2.2.08.12). Water chemistry is then altered based on steel corrosion rates of the structural steel of the waste package, waste type, waste degradation rate, cladding failure, and seepage flux.

Supplemental Discussion: The current design of the tunnel support system uses a steel liner. The waste package is also constructed of steel. The sulfur in the steel can influence the chemistry of water. Therefore, the In-Package Chemistry Component of the Waste Form Degradation Model includes the effect of steel corrosion on water chemistry.

The current design of the tunnel support system does not include concrete. Consequently, the amount of cementitious material that can influence water chemistry has been decreased by an order of magnitude from ~800 kg/m of tunnel length for the concrete liner to ~90 kg/m of tunnel length for grout. Therefore, the influence of cementitious material on water chemistry was not included in the PA models because of low consequence (that is, although the cementitious material raises the pH, the corrosion of steel lowers the pH; and the influence of the large amount of steel can swamp the influence of the small amount of cementitious material).

The effect of a carrier plume on the invert is excluded from the TSPA (CRWMS M&O 2000a). The invert will be filled with a fine quartz sand or crushed rock that is represented as a homogeneous, porous medium with no retardation of radionuclides for the TSPA. The plume will have no chemical effect on the invert because retardation is already at its minimum value and because the quartz sand or crushed rock will be relatively inert in any anticipated repository environment.

The plume is also expected to have no significant hydrological effects on the invert. Precipitation of secondary minerals into the pore spaces of the sand could alter its mechanical strength and permeability. However, the impact of these changes on total-system performance will be negligible because the invert is a minor barrier in comparison to other elements of the engineered or geologic barrier systems. The invert is anticipated to be a minimal barrier to flow and transport because (1) of the relatively high permeability of the invert material, (2) the flow distance through the invert is on the order of 0.5 meters, and (3) there is no sorption for any radionuclide.

Also note that any changes in invert permeability induced by the carrier plume are likely to make its permeabillity less than that of quartz sand, so the EBS model for the invert is conservative because it maximizes contaminant transport to the unsaturated zone.

6.2.24 Interaction with Corrosion Products—YMP No. 2.1.09.02.00

YMP Primary FEP Description: Corrosion products produced during degradation of the waste form and the metallic portions of the waste package may affect the mobilization and transport of radionuclides. Corrosion products may form a "rind" around the fuel that could (1) restrict the availability of water for dissolution of radionuclides or (2) inhibit advective or diffusive transport of water and radionuclides from the waste form to the EBS. Corrosion products also have the potential to retard the transport of radionuclides to the EBS. Finally, corrosion products may alter the local chemistry, possibly enhancing dissolution rates for specific waste forms or altering radionuclide solubility.

Screening Decision and Regulatory Basis: *Included* (the effect of the presence of a rind around the fuel on the availability of water for radionuclide dissolution, the interaction between the expanding rind in the sealing of the gap and the unzipping of the cladding, and selected chemical effects in the integrated source term for each waste form).

Excluded based on low consequence (the potential effects of corrosion products on advective/diffusive transport of water and radionuclides and the potential sorptive effects from corrosion products) (see the In-package Sorption FEP YMP No. 2.1.09.05.00 in this AMR).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.05.00	In-package sorption
2.1.09.03.00	Volume increase of corrosion products
2.1.02.23.00	Cladding unzipping

IRSR Issues: TSPAI3, ENF3

Screening Argument: The effect of the presence of a rind around the fuel on the availability of water for radionuclide dissolution, the interaction between the expanding rind in the sealing of the gap and the unzipping of the cladding, and selected chemical effects in the integrated source term for each waste form have been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

The potential effects of corrosion products on advective/diffusive transport of water and radionuclides has been excluded from the TSPA based on low consequence. The inclusion of the potential effects of corrosion products on advective/diffusive transport of water and radionuclides and the potential sorptive effects from corrosion products (see the in-package sorption, FEP YMP No. 2.1.09.05.00, in this AMR) in the TSPA-SR calculation would reduce interpreted doses. Therefore, omission of the potential of low-permeability corrosion products to limit transport of radionuclide species within the waste package and the potential sorptive effects from corrosion products, as a system-wide feature from the performance assessment models, will not significantly change the interpretation of calculated expected annual dose.

The potential sorptive effects from corrosion products has been excluded from the TSPA, based on low consequence (see the in-package sorption FEP YMP No. 2.1.09.05.00 in this AMR). No credit is taken for the potential retardation of radionuclide species on the corrosion products within the waste package. Therefore, omission of sorption on corrosion products will not significantly change the interpretation of calculated expected annual dose.

TSPA Disposition: The presence of a rind around the spent nuclear fuel is included in the evaluation of the In-Package Solubility Component of the TSPA in the calculation of the availability of water for radionuclide dissolution. See the chemistry abstraction AMR (CRWMS M&O 2000i) and source term AMR (CRWMS M&O 2000p).

The interaction between the expanding rind of alteration products in the sealing of the gap and the unzipping of the cladding are included in the fast-release fraction and wet unzipping model. See the cladding degradation abstraction AMR (CRWMS M&O 2000w).

Selected chemical effects in the integrated source term for each waste form are included in the chemistry-dependent solubility of the radionuclides, colloid concentrations, and waste-form corrosion models of the TSPA. See the concentration limit AMR (CRWMS M&O 2000y), colloid concentration limit AMR (CRWMS M&O 2000z), CSNF degradation AMR (CRWMS M&O 2000c), DSNF degradation AMR (CRWMS M&O 2000e), and HLW degradation AMR (CRWMS M&O 2000d).

The potential effects of corrosion products on advective/diffusive transport of water and radionuclides and the retardation of radionuclides within the waste package have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Chemical effects from corrosion products are included in the integrated source-term models for aqueous dissolution of CSNF, glass HLW, and DSNF. These chemical effects include temperature, pH, and concentration of selected species, as appropriate.

The expectation at Yucca Mountain is that once the package and container walls are breached, water or water vapor may enter the waste package. Exposed package materials and waste forms may react with water and air to produce secondary phases. These reactions may alter the chemical and hydrological environment within the waste package.

The minimum water that may interact with the waste and also provide for advective or diffusive transport, is enough water to support film or porous flow on the reacting waste-form surfaces. This water volume is calculated within the TSPA based on the volume, saturation and porosity of the waste-form secondary phases (see TSPA-VA [CRWMS M&O 1998c, p. T6-38]).

The expansion of fuel as the UO_2 reacts to form corrosion products is included in the clad unzipping model (see also FEP YMP No. 2.1.09.03.00 and 2.1.02.23.00). After rod perforation, the exposed gap surface area is modeled to react until the gap is filled with corrosion products. The fraction of fuel corroded before gap closure is modeled as the fast-release fraction. Continued reaction is then modeled to occur only in the vicinity of the cladding breach. This continued reaction exerts force on the cladding, which splits the cladding open at a rate that is faster than the forward dissolution rate. The interaction of the corrosion products with the cladding is covered in FEP YMP No. 2.1.09.03.00 and 2.1.02.23.00.

Except for within intact portions of fuel rods, no credit is taken for the potential of corrosion products to form low permeability solids that may reduce the flow of water and the transport of radionuclides (similar to the implementation in the TSPA-VA [CRWMS M&O 1998c, p. 6-132]). Instead, as water enters the waste package, it is instantly mixed with all resident water, which has full access to all exposed and altered waste. Similarly, no credit is taken for the retarding effects of clays, zeolites or any other corrosion products within the waste package on the transport of radionuclides from the waste-form surface. These are conservative conclusions for the TSPA-SA and TSPA-VA analyses because they maximize flow and transport from the waste package to the EBS.

Note that these conclusions are not conservative for criticality issues, where concentration of radionuclides within the waste form and waste package is a major concern. Analyses of near-field criticality issues therefore include appropriate partition coefficients to represent the potential retardation effects within the waste package. See YMP No. 2.1.14.11.00 for details.

The changes in the chemical environment are discussed in *Summary of In-Package Chemistry for Waste Forms* (CRWMS M&O 2000m). For example, corrosion of carbon steel packaging produces corrosion products and may decrease the pH. Corrosion and failure of the Zircaloy cladding and the subsequent oxidation of UO_2 fuel pellets will produce secondary uranium phases as discussed in *Secondary Uranium-Phase Paragenesis and Incorporation of Radionuclides into Secondary Phases* (CRWMS M&O 2000l). Reaction of HLW glass with water results in clays and zeolites (see *Defense High Level Waste Glass Degradation* [CRWMS M&O 2000d]). These later reactions tend to neutralize the acid produced by reaction of the carbon steel (CRWMS M&O 2000m).

The aqueous-dissolution models for wastes include the following chemical or physical parameters for the TSPA-SR analyses:

- The dissolution rate for CSNF is based on high flow-rate experimental data for commercial spent fuel and uranium dioxide (*CSNF Waste Form Degradation: Summary Abstraction* [CRWMS M&O 2000c]). The rate equation depends on temperature, pH, total carbonate ion concentration in solution, oxygen concentration in the gas, and surface area. The effective surface area is covered in *Clad Degradation–Summary and Abstraction* (CRWMS M&O 2000w).

- The dissolution-rate equation for HLW depends on temperature, the pH of the incoming water, surface area and a long-term dissolution-rate determined from test data (*Defense High Level Waste Glass Degradation* [CRWMS M&O 2000d]).

- The dissolution rate for DSNF is based on the dissolution rate of metallic fuel because this fuel comprises the majority (85%) of the DSNF on a MTHM basis (*DSNF and Other Waste Form Degradation Abstraction* [CRWMS M&O 2000e]).

Other TSPA-SR models are also dependent on in-waste-package chemistry:

- Solubility of Uranium, Neptunium, Americium are functions of pH and/or CO_2 (*Summary of Dissolved Concentration Limits* [CRWMS M&O 2000y]).

Colloid concentrations are functions of pH and ionic strength (*Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary* [CRWMS M&O 2000z]).

6.2.25 Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and EBS—YMP No. 2.1.09.04.00

YMP Primary FEP Description: Degradation of the waste form will mobilize radionuclides in the aqueous phase. Factors to be considered in this FEP include the initial radionuclide inventory, justification of the limited inventory included in evaluations of aqueous concentrations, and the solubility limits for those radionuclides. See related FEPs in this section for discussions of processes that influence solubility limits

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.01.00	DSNF degradation
2.1.02.02.00	CSNF degradation
2.1.02.03.00	HLW degradation
2.1.02.09.00	Void space
2.1.09.14.0	Colloid formation

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Solubility and speciation data are of primary importance to understand and predict radionuclide (RN) concentration limits and transport through the geosphere. Solubility in a natural system is defined as the total soluble RN concentration in solution under any given set of conditions of transporting pH, (ionic strength of the solution) and combined solid from; speciation refers to the nature in which the radionuclide occurs under a specific set of chemical conditions. The concentration of RNs in aqueous solutions (groundwater, pore fluids, etc.) will be limited by the solubility of RN-bearing solids formed by the interaction between RN-bearing waste (spent nuclear fuel) and the solution phase.

Radionuclide solubility, solubility limits, and speciation in the waste form have been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: As in previous TSPAs, the mass of radioisotopes released (based on the degradation rates of the CSNF, DSNF, or HLW matrix) will be compared to the maximum dissolved mass possible (based on the amount water flowing through the disposal container and specified concentration limits). If the maximum dissolved mass is less than the mass of radioisotopes liberated from the waste matrix, the mass released will be reasonably concluded to be the dissolved mass, and the difference will be reasonably concluded to precipitate out of solution and be available for transport at later times. The concentration limit usually is the solubility limit of pure phase species of the various radioelements, as discussed in the next section. However, experiments have shown much lower concentrations in solution for some critical radioisotopes such as Np-237 (CRWMS M&O 2000y, Section 6.4). As mechanistic base models are developed and experimentally confirmed, the pure-phase range may be extended to lower values to include mixed-phase effects (CRWMS M&O 1999h, Section 3.5.2.7).

Usually, the concentration limits for each radioisotope transported in the TSPA-SR will be expressed as a distribution of values (CRWMS M&O 2000y, Table 20). However, the concentration limit for uranium and neptunium, for which sufficient data are available, will probably be expressed as a function dependent on water chemistry (pH, Eh, and $[CO_3]_T$). Under equilibrium conditions, concentrations of radioisotopes in solution are limited by the solubility products of the solid phases that contain the radioisotopes (either solid phases with the radioisotope as the dominant element or solid phases with trace amounts as can occur with coprecipitation). The solid phases that form depend on the temperature, redox conditions, and species in solution in the groundwater. Uncertainty in the precise values for these variables in the waste package and emplacement drifts results in a wide distribution of possible concentration limits. For TSPA-VA, the distribution of each radioisotope transported was primarily based on an elicitation of experts both inside and outside the YMP, conducted in 1993 (Wilson et al. 1994, pp. 9-1 through 9-11; CRWMS M&O 1998c, Table 6-32). For TSPA-SR, a reevaluation of radioisotope solubility is planned. A distribution of concentration limits for important radioisotopes will first be based on a wide variety of chemical conditions. Although scientific judgment will be necessary to define the solid phases present and range of water chemistry, determination of the range of the distribution will be facilitated by using a chemical equilibrium code, based on either thermodynamic data available from respected sources, such as the database maintained by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development, or review of literature data (CRWMS M&O 1998e, p. 6-95). The distribution will later be refined and narrowed as information becomes available on the design of the engineered barrier, fluid flow rates, and thermal history (CRWMS M&O 1999h, Section 3.5.2.7.1).

Supplemental Discussion: Solubility and speciation data are of primary importance to understand and predict RN concentration limits and subsequent transport through the geosphere. Solubility in a natural system is defined as the total soluble RN concentration in solution under any given set of conditions of transporting pH (and strength of the solution), and combined solid form; speciation refers to the nature in which the radionuclide occurs under a specific set of chemical conditions. The concentration of RNs in aqueous solutions (groundwater, pore fluids, etc.) will be limited by the solubility of RN-bearing solids formed by the interaction between RN-bearing waste (spent nuclear fuel) and the solution phase. In addition, radionuclides may be sorbed on corrosion products or minerals or coprecipitated with secondary phases. Solubility constraints become primarily important for local dissolution and precipitation of waste-package materials. Thermodynamic constants, such as the solubility product of RN-bearing solid phases, formation constants of RN solution species, and potentials for redox couples, are key parameters that define the source term for RN transport from the spent nuclear fuel into the environment and are included in risk-assessment calculations.

Water is the main transport medium for RN migration in the environment. The chemistry of the RN in natural waters is governed by a variety of chemical reactions in parallel, such as complexation reactions, redox reactions, colloid formation, or mineral-surface reactions. Ambient water contains various constituents in micro- and macroconcentrations providing the basis for multicomponent reactions of soluble RN species. The key parameters of the solution phase affecting the RN solubility are redox potential, pH, pCO_2 or carbonate concentration, organic content, and ionic strength. The concentration of the water constituents may change with the contact of different geologic material, engineered waste-barrier material, or the RN-bearing waste. A change in the water composition may result in a different chemistry of the RN dissolved and may change the solubility and speciation. To predict these potential changes in RN migration behavior, the fundamental reactions and thermodynamic constants have to be known in order to allow solubility-limit calculations under a variety of water compositions.

Thus, to predict RN solubility limits accurately, one needs to know (1) the chemical composition of the aquifer in contact with the RN, (2) solubility products of RN-bearing solid phases, and (3) formation constants of RN solution species. Additional physicochemical phenomena, such as the dissolution kinetics of waste packages and solid phases or the formation of secondary solid phases, are interconnected but are not part of this FEP.

A list of elements/RNs (CRWMS M&O 2000j) has been selected to represent the inventory stored in Yucca Mountain for solubility calculations. Elements of highest priority are U, Np, Pu, Am, Pd, I, and Tc. Fourteen other elements of lower priority are C, Zr, Th, Nb, Ra, Sn, Ac, Cs, Pa, Cm, Pb, Se, and Cl. Uranium will dominate the elemental distribution as the matrix of spent nuclear fuel (CRWMS M&O 2000j). During the 10,000-year period of immediate regulatory concern for the potential Yucca Mountain Repository, ^{99}Tc and ^{129}I are estimated to contribute more than 95% of the potential dose. After this period ^{237}Np and $^{238/239}$Pu will contribute significantly. Some of these elements are very redox-sensitive (i.e., Pu, Np, Tc) and may exist in several oxidation states. The valence state of those redox-sensitive radionuclides primarily defines the geochemical reactions of these elements. Solubility-limited concentrations, complexation reactions in solutions, sorption onto minerals, and colloid formation differ considerably among the oxidation states. Generally, the RNs in lower oxidation states (+III, +IV) are unstable at neutral pH against hydrolysis and other complexation reactions resulting in low solubilities with concentrations below 10^{-6} M. As an example, neptunium (Np) ions may exist in the III, IV, V, VI, and even VII valence states, but only the IV, V, and VI states are relevant for natural environments. Neptunium speciation is dominated by the pentavalent cation, NpO_2^+, under a wide range of environmental conditions. Since Np(V) solid phases are highly soluble and Np(V) aqueous species do not easily sorb on common minerals, Np(V) is very mobile in the environment. Tetravalent neptunium exists under the reducing conditions that are expected at nuclear-waste disposal sites. Np(IV) solids are less soluble than those of Np(V), and the tendency for Np(IV) to form aqueous complexes produces strong interactions with the geomatrix. Np(VI) is not important under most environmental conditions but is stable in highly oxidizing solutions. In order to model the complex behaviors of Np and other RNs under environmental conditions, the knowledge of potential geochemical reactions is indispensable. Accurate thermodynamic data are key to reliable modeling of the geochemistry of RNs of concern and are usually obtained from experiments in well-defined laboratory systems. Empirical solubility data in natural waters provide additional information that can be used as test cases for the thermodynamic database. A good agreement between experiments and calculations may enhance confidence in the ability to model solubility distributions of RNs under repository conditions.

6.2.26 In-Package Sorption—YMP No. 2.1.09.05.00

YMP Primary FEP Description: Sorption of radionuclides within the waste package may affect the aqueous concentrations of radionuclides released to the EBS.

Screening Decision and Regulatory Basis: *Included* (sorption on mobile colloidal material).

Excluded based on low consequence (sorption within the waste form/waste package).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.02.00	Interaction with corrosion products
2.1.09.14.00	Sorption on mobile colloidal material
2.1.14.11.00	near field criticality

IRSR Issues: TSPAI3, ENF3

Screening Argument: Sorption on mobile colloidal material has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

Sorption within the waste form and waste package has been excluded from TSPA consideration based on low consequence. The inclusion of sorption within the waste form/waste package in the TSPA-SR calculation would reduce interpreted doses. Therefore, the omission of the beneficial retarding effects of alteration products within the waste package as a system wide feature from the performance assessment models will not significantly change the interpretation of calculated expected annual dose.

The waste package, spent fuel supports, and portions of the SNF assembly will oxidize to minerals that may provide substantial sorption for radionuclides in the waste (e.g., CSNF, HLW, DSNF). For example, iron in the waste package may eventually be converted to iron oxides that have large sorptive capacity for radionuclides, especially, actinides. Also HLW glass will probably be converted to clays and zeolites that strongly sorb radionuclides. No credit is taken for the retarding effects of these waste form and waste-package alteration products (i.e., clays, zeolites or iron oxides) on transport of radionuclides within the waste package. This reasonable and conservative conclusion for the TSPA-SR and TSPA-LA analyses because it maximizes transport to the EBS. However, zero retardation is considered to be nonconservative for near-field criticality, where retention of fissile radionuclides near the waste package may facilitate criticality (e.g., see FEP. 2.1.14.11.00). Sorption on mobile colloidal material is discussed in FEP 2.1.09.14.00 in CRWMS M&O 2000z.

TSPA Disposition: Sorption on mobile colloidal material is included in the colloid radioisotope concentration component model (CRWMS M&O 2000z).

The beneficial retarding effects of alteration products within the waste package have been excluded from consideration in the TSPA as discussed in the *Screening Argument* for this FEP.

Supplemental Discussion: A more complete discussion of the internal scenarios for fuel degradation is discussed in *Disposal Criticality Analysis Methodology Topical Report* (YMP 1998, Section 3.1). Both internal and external scenarios are discussed.

6.2.27 Reduction-Oxidation Potential in Waste and EBS—YMP No. 2.1.09.06.00

YMP Primary FEP Description: The redox potential in the waste and EBS influences the oxidation of barrier and waste-form materials and the solubility of radionuclide species. Local variations in the redox potential can occur.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.09.0	Reaction kinetics

IRSR Issues: TSPAI3, ENF3, CLST1, CLST3, CLST4

Screening Argument: The redox potential in the groundwater is taken into account in the equilibrium-model calculations that examine the probable range of in–package fluid chemistries likely to result from influx of ambient fluids. A reasonable conclusion in the model calculations is that atmospheric gases are in equilibrium with solutions in the waste package and will remain in equilibrium throughout any chemical reaction. Oxidizing conditions are set at the optimum value for the calculations, and, therefore, all calculations are conservative. Additionally, atmospheric carbon dioxide is also considered in equilibrium with the groundwater and is set at optimum values for the calculations. Therefore, the effect of carbonate on actinide solubility will always be conservative in the calculations.

Groundwater redox potential has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: The regression equation of the In-Package Chemistry Component bounds the redox potential of the groundwater seepage in the equilibrium model that examines the probable range of in-package pH. Specifically, the fluid is modeled as being equilibrated with the atmosphere to ensure maximum plausible oxygen and carbon dioxide conditions. These conditions are used in modeling CSNF matrix degradation (see CSNF FEP 2.1.02.02.00) and the dissolved radioisotope concentration (see Solubility FEP 2.1.09.04.00). For details see the in-package chemistry abstraction AMR (CRWMS M&O 2000i) and the summary of the in-package chemistry AMR (CRWMS M&O 2000m).

Supplemental Discussion: The redox potential in the groundwater is taken into account in the equilibrium-model calculations to examine the probable range of in–package fluid chemistries likely to result from influx of ambient fluids. A reasonable conclusion in the model calculations is that atmospheric gases are in equilibrium with solutions in the waste package and will remain in equilibrium throughout any chemical reaction (CRWMS M&O 2000m). Oxidizing conditions are set at the optimum value for the calculations, and, therefore, all calculations are conservative. Additionally, atmospheric carbon dioxide is also considered in equilibrium with the groundwater and is set at optimum values for the calculations. Therefore, the effect of carbonate on actinide solubility will always be conservative in the calculations.

Radiolysis close to the fuel can change the oxidation state of actinides, technetium, and other radionuclides. Radiolysis can result in the formation of oxidizing species such as perchlorate and hydrogen peroxide. These species can oxidize radionuclides to higher oxidation states (Vladimirova 1990; Pashalidis et al. 1993; Shoesmith and Sunder 1992; Cui and Eriksen 1996;

Finn et al. 1998; Farrell et al. 1999a). In the higher oxidation states actinides, technetium, and other radionuclides are more water soluble (Cui and Eriksen 1996; Farrell et al. 1999b; Felmy and Rai 1999).

However, once transported outside the waste package, corrosion of the outer package steel will produce a reducing environment in the vicinity of the waste package. It is well documented that iron corrosion reduces actinide, technetium, and other radionuclides to lower, less soluble oxidation states (Cui and Eriksen 1996; Grambow et al. 1996; Fiedor et al. 1998; Farrell et al. 1999b). Therefore, radionuclides will precipitate just outside the waste package.

If oxidizing groundwater enters the far-field and encounters a reducing environment, thus changing the redox chemistry of the contaminants, the effect will not be significant. Reducing conditions will result in reduction of radionuclides to lower oxidation states where they are less soluble (Cui and Eriksen 1996; Farrell et al. 1999b; Felmy and Rai 1999). Therefore, code solubility and transport calculations are always conservative (CRWMS M&O 2000m).

If the far field does maintain natural reducing conditions after repository closure, transport of actinides through the far-field will be minimized because radionuclides will be reduced to lower, less soluble oxidation states (Cui and Eriksen 1996; Farrell et al. 1999b; Felmy and Rai 1999). Therefore, code solubility and transport calculations are always conservative (CRWMS M&O 2000m).

6.2.28 Reaction Kinetics in Waste and EBS—YMP No. 2.1.09.07.00

YMP Primary FEP Description: Chemical reactions, such as radionuclide dissolution/ precipitation reactions and reactions controlling the reduction-oxidation state, may not be at equilibrium in the drift and waste environment.

Screening Decision and Regulatory Basis: *Included* (reaction kinetics in the in-package equilibrium model).

Excluded based on low consequence (impacts of transient disequilibrium states).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.06.00	Redox potential

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Redox equilibrium is rarely completely attained in near-surface natural systems, and, therefore, is not expected in the waste environment. Lack of redox equilibrium is not expected to be significant though the in-package chemistry model specifies that uniformly oxidizing conditions will prevail, thus stabilizing many radionuclides in their most soluble states.

Transient disequilibrium states, not otherwise dealt with by the reaction path approach or by using oxidizing conditions, may occur. For example, secondary phases predicted to form by equilibrium thermodynamics, may not occur because of unfavorable reaction kinetics. However, transient disequilibrium effects are excluded on the basis of low consequence. Their omission will not significantly change calculated expected annual dose because long reaction times and the explicit suppression of many disequilibrium phases is expected to limit the overall impact of transient disequilibria (CRWMS M&O 2000i; 2000m).

Reaction kinetics have been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: Included reaction kinetics through quasi-equilibrium analysis in evaluating the In-Package Chemistry Component in the Waste Form Degradation Model. Reaction kinetics is included through a reaction-path analysis that specifies local equilibrium between fluids and secondary phases but includes disequilibria between manufactured waste package components. The equilibrium model calculates the pH over time using as input the degradation rates of cladding, SNF matrix, HLW glass, and internal components of the disposal package which consist of stainless steel and aluminum (CRWMS M&O 2000i; 2000m).

Impacts of transient disequilibrium states have been excluded from consideration in the TSPA as discussed in the *Screening Argument* for this FEP.

Supplemental Discussion: Reaction kinetics are adequately discussed in the *Screening Argument.*

6.2.29 Chemical Gradients/Enhanced Diffusion in Waste and EBS—YMP No. 2.1.09.08.00

YMP Primary FEP Description: The existence of chemical gradients within the disposal system, induced naturally or resulting from repository material and waste emplacement, may influence the transport of contaminants of dissolved and colloidal species.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs: None.

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Possible chemical and physical gradients include redox, organic ligands, ionic strength, carbonate, pH, and temperature gradients. Redox has been discussed in YMP No. 2.1.09.06.00. Oxidizing conditions will not increase actinide solubility outside the waste package since there is no external source of actinides for further dissolution. Reducing conditions will result in reduction of actinides, technetium, and other radionuclides to lower less soluble oxidation states, therefore decreasing transport (Cui and Eriksen 1996; Farrell et al. 1999a; Felmy and Rai 1999). Therefore, omission of chemical gradients/enhanced diffusion in the waste will not significantly change the calculated expected annual dose, and chemical gradients/enhanced diffusion have been excluded from TSPA consideration based on low consequence.

TSPA Disposition: Chemical gradients/enhanced diffusion in the waste have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Possible chemical and physical gradients include redox, organic ligands, ionic strength, carbonate, pH, and temperature gradients. Redox has been discussed in YMP No. 2.1.09.06.00 Oxidizing conditions will not increase actinide solubility outside the waste package since there is no external source of actinides for further dissolution. Reducing conditions will result in reduction of actinides, technetium, and other radionuclides to lower, less soluble oxidation states, therefore decreasing transport (Cui and Eriksen 1996; Farrell et al. 1999a; Felmy and Rai 1999).

Gradients created by organic ligands are not considered since no ligands are in the waste for YMP, and natural background organics are low as well.

Ionic strength does have an influence on radionuclide solubility but no effect on oxidation state (Felmy and Rai 1999). Higher concentrations of salt are possible just outside the waste package where salt buildup can occur in the early post closure stage of the repository. Heat generated by the waste package can evaporate groundwater and concentrate salts near the waste package. However, once radionuclides are transported a short distance away from the waste package, ionic strength will decrease to groundwater levels and radionuclides will precipitate. Therefore, the possibility of a gradient to facilitate transport is very low.

Glass dissolution should cause waste-package pHs and CO_2 concentrations to increase. Carbonate complexes can solubilize actinides. Because glass dissolution will increase the solution pH inside the waste package, a pH gradient will be established going from high in the waste package to lower outside the waste package. Radionuclide concentrations tend to be higher at high pH. Therefore, this gradient will not facilitate actinide transport outside the waste package.

Temperature will be higher inside and near the waste package and decrease with distance from the waste package. Since actinides have higher solubility at elevated temperatures, a temperature gradient will not serve to facilitate transport away from the waste package. Actinides will precipitate as they move away from the waste package and temperature decreases.

For the no backfill repository design option, where the temperature gradient will be lower, the actinides will precipitate out closer to their source as they move away from the waste package.

The In-Package Chemistry Component (CRWMS M&O 2000m) of the TSPA specifies that the waste package is considered to be a mixing cell without chemical gradients. Provided the waste package is fully saturated, the rate of the flow of water into and out of the package is slow enough that no long-term gradient would be expected and equilibrium should occur. Furthermore, rather than include diffusive release of radionuclides out of the CSNF perforated cladding, DSNF perforated cladding, or cracked HLW glass, a more conservative conceptual model is developed that subsumes enhanced release from diffusion caused by chemical gradients by allowing cladding unzipping for CSNF; 100% failed cladding for DSNF; and very high reactive surface area for HLW glass.

6.2.30 Electrochemical Effects (Electrophoresis, Galvanic Coupling) in Waste and EBS—YMP No. 2.1.09.09.00

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

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.03.01.0	Corrosion of waste containers

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Electrochemical effects (electrophoresis, galvanic coupling) in the waste have been excluded based on low consequence. Omission of electrochemical effects (electrophoresis, galvanic coupling) in the waste will not significantly change the calculated expected annual dose because those effects are much smaller than the effects of modeling with a minimum flow rate through a failed container of 15 l/yr used in performing the equilibrium-model calculations (CRWMS M&O 2000m). This flow rate has a much greater effect than can be created by electrophoresis or electro-osmosis (Soderman and Jonsson 1996).

Possible effects of galvanic coupling between cladding and the waste package or other structures within the waste package are addressed in detail in FEP 2.1.02.22.03, Hydride Embrittlement from Galvanic Corrosion of WP Contacting Cladding (CRWMS M&O 2000ae, Clad Degradation–FEPs Screening Arguments) and are excluded from the TSPA on the basis that the zirconium itself is very chemically active and quickly becomes coated with zirconium oxides. If a galvanic reaction starts, it only continues for a very short time, until the surface repassivates. Galvanic coupling between cladding and the spent fuel within the fuel rods is not considered to be a credible process because fuel pellets and cladding are in intimate contact throughout most of the fuel cycle and galvanic reactions between the two material are not known to occur. Garde (1986, Section 4.7) examined the inner surface of high-burnup PWR cladding. He reports a small chemical-interaction layer from fissioning of plutonium on the pellet rim but does not report any observation of galvanic corrosion of the cladding. Einziger and Koli (1984, Figures 6 and 7) show micrographs of irradiated fuel after dry-storage tests. Again, they report no galvanic reaction between the fuel and cladding.

TSPA Disposition: Electrochemical effects (electrophoresis, galvanic coupling) in the waste are excluded based on low consequence as discussed in the *Screening Argument* and *Supplemental Discussion.*

Supplemental Discussion: Adequately discussed in the *Screening Argument* for this FEP.

6.2.31 Secondary Phase Effects on Dissolved Radionuclide Concentrations at the Waste Form—YMP No. 2.1.09.10.00

YMP Primary FEP Description: Inclusion of radionuclides in secondary uranium mineral phases, such as neptunium in schoepite and uranium silicates, could affect radionuclide concentrations in water in contact with the waste form. During radionuclide alteration, the radionuclides could be chemically bound to immobile compounds and result in a reduction of available radionuclides for mobilization.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.04.0	Radionuclide solubility

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Inclusion of secondary-mineral phases in the TSPA-SR would reduce calculated doses because these solids have the ability to sequester radionuclides (see *Supplemental Discussion*). However, no credit is allowed for this process in the TSPA. Omission of these secondary-phase effects on dissolved radionuclide concentrations will not significantly change the interpretation of calculated expected annual dose. Therefore, secondary-phase effects on dissolved radionuclide concentrations are excluded from consideration in the TSPA based on low consequence.

TSPA Disposition: Secondary-phase effects on dissolved-radionuclide concentrations have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: Incorporation of certain RN into corrosion products formed during the alteration of spent nuclear fuel may reduce RN concentrations in waters that have contacted fuel and its corrosion products. Whether a solid can sequester a particular RN depends, first, on whether that RN is compatible in the solid's structure.

During the 10,000-year period of immediate regulatory concern for the potential Yucca Mountain Repository, ^{99}Tc and ^{129}I are estimated to contribute more than 95% of the potential dose. Chemical behaviors of these two elements from a wide range of experiments indicate that their releases will not be significantly reduced by Tc- or I-bearing solids. Crystal chemistry suggests Tc(VII) and I will not enter structures of any known U(VI) solids, and all known Tc(VII) and I compounds are highly soluble in water. The four fission products, Tc, I, Se, and C, can be considered as being released from the fuel as it dissolves and as fuel-grain boundaries open and are exposed to water vapor or groundwater. These four elements are unlikely to precipitate under repository-relevant conditions, and the combination of reaction rate and water flow rate appear to control their releases in the ANL unsaturated experiments. In fact, Tc release may depend on the corrosion rate of e-Ru, rather than the fuel matrix. Only Se and C might become incorporated into corrosion products; however, no experimental evidence for this is available at this time (CRWMS M&O 2000l, Section 7.1, p. 39).

The dose contribution from ^{237}Np becomes significant only beyond 10,000 years as described in the Waste Form Inventory Abstraction Model (CRWMS M&O 2000j). As discussed in radionuclide secondary phase AMR (CRWMS M&O 2000l), crystal chemical similarities between oxysalts of Np(V) and U(VI) indicate that substantial substitution of Np(V) into some U(VI) corrosion products is possible; however, Np(V) can substitute for U(VI) only if (1) charge-balance mechanisms are available and (2) dissolved Np and U exist together in solution such that they can co-precipitate. Solution data from most studies on the oxidative dissolution of spent UO_2 fuels indicate that Np is released congruently from the UO_2 matrix. Unfortunately, few experiments report both secondary precipitates and Np concentrations in solution. If U(VI) solids have precipitated, apparent congruent release of U and Np is most readily explained if both elements are controlled by the same solid(s). Whether Np enters into U(VI) solids depends, at least in part, on the solid (which depends in turn on experimental conditions). Dehydrated schoepite (DS) formed during fuel corrosion in water vapor contains Np, and the Np/U ratio in DS is comparable to that of the fuel. In contrast, U(VI) solids from drip tests with high rates of groundwater injection do not contain Np in significant amounts. It remains uncertain whether the lack of substantial Np in solids formed in drip tests reflects incompatibility of Np(V) in solids (the uranyl silicate, Na-boltwoodite, and a Cs-Mo-uranate), or whether differences in the solutions contacting dissolving fuel prevent co-precipitation of U and Np. Nevertheless, solution behaviors of Np and U are closely similar in "drip" tests, suggesting some role of U solids in limiting Np release. Experiments at ANL injected Np-bearing groundwater onto unirradiated UO_2 that had already formed a suit of U(VI) corrosion products.

U(VI) solids formed during corrosion of both unirradiated UO_2 and spent fuel are similar to solids formed where natural UO_2 has been corroded by oxidizing groundwaters, suggesting that solids formed in fuel experiments are likely to form in the repository under similar conditions. In nature, many of these minerals persist for many thousands of years under some geochemical conditions. Thus, evidence from natural analogues suggests that U(VI) solids with RNs in stable structural sites, are potentially long-term RN hosts.

In order to quantify and model RN sequestration by U(VI) solids one needs to know (1) which solids can sequester Np and other RNs, (2) solubilities and stability ranges of relevant U(VI) solids, (3) RN partition coefficients between U(VI) solids and aqueous solutions of various compositions, and (4) precipitation and dissolution rates of relevant U(VI) solids. Strictly empirical evidence that RNs are incorporated into solids formed on corroded fuel under select experimental conditions provides only limited confidence that these same solids will limit RN releases over repository-relevant time scales.

Experimental evidence from the ANL unsaturated experiments suggests that although Np has been found in a uranyl oxyhydroxide from vapor-corroded fuel, Np may not be incorporated into uranyl silicates to a significant degree where fuel is fully exposed to Si-saturated waters. Fuel that is contacted by an effectively infinite supply of Si-saturated groundwater will probably be converted entirely to uranyl silicates. Modeling Np release as being only a function of the fuel-matrix dissolution rate appears to be a very conservative approach, because even in Si-saturated waters, Np apparently is not released congruently with other matrix elements. Np may well precipitate under these conditions, but the identity of a Np-bearing phase, if it exists, remains elusive. The apparent retention of Np in experiments that use Si-saturated waters may be due to the incomplete oxidation of Np(IV) in the fuel to Np(V) under most experimental conditions.

Without positively identifying the specific compounds that may contain Np, a conservative approach is recommended. Complete release of Np from the fraction of fuel contacted by Si-saturated groundwater is probably the most defensible approach. Because Np apparently exhibits different behaviors, depending on whether fuel is exposed to humid air or Si-saturated water, establishing the likely mode of water contact with exposed fuel is crucial for predicting the potential long-term release of Np from corroded fuel.

Experimental results from both the ANL unsaturated tests, as well as other experiments reviewed, indicate that releases of the actinides, Am, Pu, and Pa can be modeled as depending on the dissolution of the fuel matrix, with pure phases limiting their releases. As for Np, this appears to be a very conservative approach, because these elements are usually retained to a significant degree. This recommendation is based primarily on the fact that confidently modeling the dissolution behavior of the residual solid formed in the ANL unsaturated tests, for example, seems exceedingly difficult or impossible.

Even though the incorporation of many RNs into U(VI) corrosion products will probably occur in the YMP system, the uncertainties associated with this process are high, and the total contribution to a reduction in radionuclide mobilization is uncertain. The conservative approach is to not take credit for secondary phase effects until there is sufficient technical evidence to quantify the reduction of available radionuclides for mobilization (CRWMS M&O 2000l, Section 7.1, pp. 38–39).

DSNF and HLW have been evaluated with regard to their effect or contribution to the technical issues discussed in this FEP, and their particular or unique properties do not contribute any singular contribution not already accounted for in the analysis of commercial spent nuclear fuel.

6.2.32 Waste-Rock Contact—YMP No. 2.1.09.11.00

YMP Primary FEP Description: Waste and rock are placed in contact by mechanical failure of the drip shields and waste packages. Reactions between uranium, rock minerals, and water in contact with both precipitate uranium, leading spent fuel to dissolve more rapidly than if constrained by the equilibrium solubility of uranium.

Screening Decision and Regulatory Basis: *Included* (the indirect effects of waste to rock contact in the in-package chemistry model using J-13 well water chemistry for incoming water).

Excluded based on low consequence (chemical and mechanical effects due to direct waste to rock contact).

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.07.01.00	Rock fall
2.2.08.12.00	J-13 well water as a surrogate for incoming water
2.1.02.24.00	Mechanical failure of cladding

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: The indirect effects of waste to rock contact in the in-package chemistry through the use of J-13 well water (FEP 2.2.08.12.00) has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

Chemical and mechanical effects due to direct waste to rock contact have been excluded from consideration based on low consequence. Omission of the effects of direct waste to rock contact will not significantly change the calculated expected annual dose because rock is not expected to come directly in contact with the waste in the first 10,000 years after repository closure. This is because of the drip shield and long-lived waste disposal container. The drip shield is designed to provide protection from rockfall until the waste package and cladding have failed. (See FEPs on mechanical failure of cladding, YMP No. 2.1.02.24.00 and rockfall YMP No. 2.1.07.01.00.) Furthermore, even if some contact were to occur, the overall result would be little or no involvement of the rock minerals in chemical reactions due to their dissolution kinetics.

This FEP applies more specifically to emplacement of waste packages vertically in boreholes in the drift floor. Current design is horizontal in-drift emplacement using large containers. Eventual contact with rock is expected as a result of drift collapse (rockfall). However, the rock-water interactions and the Fe of the container is expected to be more controlling on radionuclide solubility than the interactions suggested in this FEP.

TSPA Disposition: The indirect influence of waste to rock contact on in-package chemistry is implemented in the TSPA through the presumption that the water entering a waste package has the chemical properties of J-13 well water. Also the range of water properties (pH, eH) within the waste package encompasses the chemical changes expected as a result of direct waste to rock contact. For details see the chemistry abstraction AMR (CRWMS M&O 2000i).

The perforation of cladding from shaking in a severe earthquake is considered in a disruptive event analysis (cladding FEP 2.1.02.24.00).

Chemical and mechanical effects due to direct waste to rock contact have been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: If the waste package is breached to the extent that the waste and associated fluid come into contact with the rock within the drift, changes to the chemical environment surrounding the waste may occur. The pH may increase from approximately 5-6 in the waste package to 7-8 in the groundwater in the drift (CRWMS M&O 2000m). Similarly, Eh may rise, ionic strength may decrease, and carbonate may increase slightly (CRWMS M&O 2000m). Note, the range of solution compositions in the in-package chemistry model (CRWMS

M&O 2000m) and, subsequently, used to estimate radionuclide solubility goes beyond the range of chemical shifts proposed to occur by this FEP. Consequently, the scenario envisioned in the FEP does not lead to higher uranium concentration than is otherwise calculated.

6.2.33 Rind (Altered Zone) Formation in Waste, EBS, and Adjacent Rock—YMP No. 2.1.09.12.00

YMP Primary FEP Description: Thermal-chemical processes involving precipitation, condensation and redissolution alter the properties of the waste, EBS, and adjacent rock. These alterations form a rind, or altered zone, with hydrologic, thermal, and mineralogical properties different from the original intact conditions.

Screening Decision and Regulatory Basis: *Included* (rind [altered zone] formation in the waste in evaluating cladding unzipping and water availability for radioisotope dissolution).

Excluded based on low consequence (rind formation in the adjacent host-rock).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.03.00	Volume increase of corrosion products
2.1.09.02.00	Interaction with corrosion products
2.1.02.09.0	Void space

IRSR Issues: TSPAI3, CLST3

Screening Argument: Rind (altered zone) formation in the waste is included in evaluating cladding unzipping and water availability for radioisotope dissolution as discussed in the *TSPA Disposition* for this FEP.

Rind formation in the adjacent host rock has been excluded from consideration in the TSPA based on low consequence. No credit is taken for rind formation in the adjacent host rock. Inclusion of rind formation in the adjacent host-rock in the TSPA-SR calculation would reduce calculated expected annual dose by reducing the permeability and increasing radionuclide retardation. Therefore, omission of a rind in the adjacent rock as a system wide feature from the performance assessment models will not significantly change the interpretation of expected dose.

Rind (altered zone) formation in the waste has been included in the TSPA as discussed in the *TSPA Disposition* for this FEP.

TSPA Disposition: The waste-form-degradation conceptual model uses a single mixing cell to represent the waste form within the waste package. This cell is scaled in size to the number of packages that have been breached within that waste-package sub-group. As more packages are breached, the size of the single waste-package cell increases to account for the volume and inventory of the newly breached package. Cells within the performance-assessment model are defined with a volume of water and mass of solid materials associated with the cell. The amount of water in the waste-form cell is reasonably concluded to be equal to the pore space of the rind of alteration products that forms as the UO_2 in the fuel is converted into secondary minerals. The volume of the rind is calculated to increase as a function of time, but is not allowed to exceed the original fuel-matrix volume (for details see the source term abstraction AMR [CRWMS M&O 2000p]).

Rind formation in adjacent rock has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion:

Waste—As individual waste rods degrade due to thermal-chemical interactions with water, a porous rind of secondary phases material is formed from the combined effect of condensation, dissolution, and precipitation processes (CRWMS M&O 2000w, Section 6.6). This rind can have different hydrologic, thermal, and mineralogical properties than the intact matrix, and these properties that subsequently will affect further waste degradation, and radionuclide dissolution and diffusion. Because the glass degradation rates are experimentally determined they implicitly include the effects of rind formation. The possible diffusion barrier effects, as well as all other rind effects, are (conservatively) excluded. In the dissolution model, radionuclides released from the waste are simulated as being dissolved, up to a solubility limit, in a fixed-volume reservoir. This volume is reasonably concluded to be the pore space in the rind.

EBS and Adjacent Rock—Rind formation in adjacent rock can conservatively be excluded. Any altered zone that does form would retard transport of nuclides to UZ through increased sorption and reduced permeability and porosity.

6.2.34 Complexation by Organics in Waste and EBS—YMP No. 2.1.09.13.00

YMP Primary FEP Description: The presence of organic complexants in water in the waste and EBS could affect radionuclide transport. Organic complexants may include materials found in natural groundwater such as humates and fulvates, or materials introduced with the waste or engineered materials.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs: None.

IRSR Issues: TSPAI3, CLST3

Screening Argument: The formation of stable aqueous complexes with organics found in natural groundwater, including high-molecular-weight humates, fulvates and methanic compounds, and with anthropogenic compounds, such as acetate, citrate, oxylate, and EDTA, could affect transport by changing buffer properties, reducing sorption and enhancing dissolved load.

While such organic complexes do not appear to be associated with Yucca Mountain waters, now or in the past, it is possible that organic complexing agents are introduced during the next pluvial cycle or by construction materials, depending on design. Also, small amounts of organics may be present in the waste packages. However, drift temperatures are expected to be sufficient to drive off volatile organics. Also, most common ligands are complexed more readily with multivalent non-radioactive metal cations. Organic complexation is not a FEP that will have significant effects on radionuclide releases. Therefore, omission of complexation by organics will not significantly change the calculated time and magnitude of the expected annual dose because there will not be reduced sorption, nor enhanced dissolved load. Complexation by organics can be excluded from consideration in the TSPA based on low consequence.

TSPA Disposition: Complexation by organics has been excluded from consideration in the TSPA as discussed in the *Screening Argument* and *Supplemental Discussion* for this FEP.

Supplemental Discussion: The formation of stable aqueous complexes with organics found in natural groundwater, including high-molecular-weight humates, fulvates and methanic compounds, and with anthropogenic compounds, such as acetate, citrate, oxylate, and EDTA could affect transport by changing buffer properties, reducing sorption and enhancing dissolved load

While such organic complexes do not appear to be associated with Yucca Mountain waters, now or in the past, it is possible that organic complexing agents are introduced during the next pluvial cycle or by construction materials, depending on design. Also, small amounts of organics may be present in the waste packages (DOE 1999c, Table ES-3). However, drift temperatures are expected to be sufficient to drive off volatile organics. Also, most common ligands are complexed more readily with multivalent non-radioactive metal cations (DOE 1996, Appendix SCR in Vol. 16, SCR.2.5.6 Organic Complexation pp. SCR-75 through SCR-76; Appendix SOTERM in Vol. 17, SOTERM.5 The Role of organic Ligands, pp. SOTERM-36 through SOTERM-41). These arguments suggest that complexation by organics is a minor consideration.

There are no unique aspects of DSNF/HLW that would significantly influence complexation by organics.

6.2.35 Heat Output/Temperature in Waste and EBS—YMP No. 2.1.11.01.00

YMP Primary FEP Description: Temperature in the waste and EBS will vary through time. Heat from radioactive decay will be the primary cause of temperature change, but other factors to be considered in determining the temperature history include the *in situ* geothermal gradient, thermal properties of the rock, EBS, and waste materials, hydrologic effects, and the possibility of reactions. Consideration of the heat generated by radioactive decay should take different properties of different waste types, including DSNF, into account.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.11.02.00	Non-uniform heat distribution/edge effects in repository
2.1.11.03.00	Exothermic reactions in waste and EBS
2.1.11.05.00	Differing thermal expansion of repository components
2.1.11.06.00	Thermal sensitization of waste packages increases fragility
2.1.11.07.00	Thermally induced stresses in waste and EBS
2.1.11.08.00	Thermal effects: chemical and microbial changes in the waste and EBS
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and EBS

IRSR Issues: TSPAI3, ENFE3

Screening Argument: Decay heat is a major issue in repository design, particularly at Yucca Mountain where high loading densities and high temperatures (>200°C) are intended to be part of the waste isolation scheme.

Temperature in the waste and EBS will vary through time. Heat from radioactive decay will be the primary cause of temperature change. The heat released from radioactive decay of the waste packages is accounted for in all of the process-level models that include repository heating. The heat-decay curves used in the models are documented in the AP-3.12Q calculation, *Heat Decay Data and Repository Footprint for Thermal-Hydrologic and Conduction-Only Models for TSPA-SR* (CRWMS M&O 2000aj). The heat from the radioactive decay will be the primary cause of temperature change, but other factors to be considered in determining the temperature history include the *in situ* geothermal gradient, and thermal properties of the rock, EBS, and waste materials, hydrologic effects, and the possibility of exothermic reactions. Consideration of the heat generated by radioactive decay will take different properties of different waste types, including DSNF, into account.

See discussion on DSNF and HLW in referenced FEPs.

TSPA Disposition: Included in thermo-hydrologic calculations.

Supplemental Discussion: Discussions can be found in the FEPs relating to non-uniform heat distribution/edge effects in repository (FEP 2.1.11.02.00); exothermic reactions in waste and EBS (FEP 2.1.11.03.00); differing thermal expansion of repository components (FEP 2.1.11.05.00); thermal sensitization of waste packages increases fragility (FEP 2.1.11.06.00); thermally induced stresses in waste and EBS (FEP 2.1.11.07.00); thermal effects on chemical and microbial changes in the waste and EBS (FEP 2.1.11.08.00); thermal effects on liquid or two-phase fluid flow in the waste and EBS (FEP 2.1.11.09.00); thermal effects on diffusion (Soret effect) in waste and EBS (FEP 2.1.11.10.00).

6.2.36 Exothermic Reactions and Other Thermal Effects in Waste and EBS—YMP No. 2.1.11.03.00

YMP Primary FEP Description: Exothermic reactions liberate heat and will alter the temperature of the waste and EBS. Oxidation of uranium metal fuels such as represented by N-Reactor fuels is one example of a possible exothermic reaction. Hydration of concrete used in the underground environment is an example of a possible exothermic reaction in the EBS.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.08.00	Pyrophoricity
2.1.11.01.00	Heat output/temperature in waste and EBS
2.1.11.04.00	Temperature effects/ coupled processes in waste and EBS
2.1.11.05.00	Differing thermal expansion of repository components
2.1.11.07.00	Thermally induced stresses in waste and EBS
2.1.11.08.00	Thermal effects: chemical and microbial changes in the waste and EBS
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS
2.1.11.10.0	Thermal effects on diffusion (Soret effect) in waste and EBS

IRSR Issues: TSPAI3, ENFE3, CLST1, CLST3, CLST4

Screening Argument: Exothermic reactions have been excluded from consideration in the TSPA based on low consequence. Omission of exothermic reactions will not significantly change the calculated expected annual dose because the possible temperature rise in a disposal container from an exothermic degradation of waste such as uranium metal in DSNF or in the EBS from the hydration of concrete is inconsequential in comparison to the heat generated by radioactive decay. Furthermore, the effects of pyrophoric reactions bound any of the effects of exothermic reactions and are also excluded (see FEP 2.1.02.08.00 Pyrophoricity). In addition, the fixed, conservative degradation rate for DSNF bound any thermal effects on waste degradation (see FEP 2.1.02.01.00–DSNF degradation, alteration, and dissolution).

TSPA Disposition: Exothermic reactions, and other thermal reactions, that liberate heat in the waste and EBS are excluded from the TSPA as described in the *Screening Argument.*

Supplemental Discussion: The amount of heat produced by radioactive decay is substantial. As shown in the TSPA-VA, the maximum rock temperature in the drift walls can reach as high as 200°C when the containers and tunnels are closely spaced (DOE 1998b, Section 5.1.3.2). The current EDA II design specifies temperatures of 96°C (CRWMS M&O 2000a; 1999c). The temperature changes caused by exothermic reactions suggested in this FEP are inconsequential by comparison. The heat of reaction of oxidizing all the uranium metal in N-Reactor fuel is about 1.3 x 10^{13} J or 1/100th of the heat energy produced in one year by all DSNF and HLW (where DSNF and HLW amount to ~7000 MTHM or one-tenth of the total repository mass of 70,000 MTHM) (DOE 1998d, Section ES.3.4). Furthermore, the effects of pyrophoric reactions bound any of the effects of exothermic reactions and are also excluded as discussed in FEP 2.1.02.08.00, Pyrophoricity. In addition, the fixed, conservative degradation rate bound the maximum degradation rates observed for N-Reactor by an order of magnitude (see FEP 2.1.02.01.00) and, thus, bound any thermal effects on waste degradation.

6.2.37 Temperature Effects/Coupled Processes in Waste and EBS—YMP No. 2.1.11.04.00

YMP Primary FEP Description: Repository temperatures, due to thermal loading, may reach elevated levels such that the dependence of temperature on processes such as hydrological, chemical, and mechanical behavior of the waste and EBS may affect repository performance.

Screening Decision and Regulatory Basis: *Included* (temperature effects and their associated coupled processes in the waste and EBS).

Excluded on low consequence (secondary temperature effects on in-package chemistry such as pH).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.11.01.00	Heat output/temperature in waste and EBS
2.1.11.02.00	Nonuniform heat distribution/edge effects in repository
2.1.11.03.00	Exothermic reactions in waste and EBS
2.1.11.05.00	Differing thermal expansion of repository components
2.1.11.06.00	Thermal sensitization of waste packages increases fragility
2.1.11.07.00	Thermally induced stresses in waste and EBS
2.1.11.08.00	Thermal effects: chemical and microbial changes in the waste and EBS
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and EBS

IRSR Issues: TSPAI3, ENFE3, CLST1, CLST3, CLST4

Screening Argument: The heat released by the waste will increase the temperature in the repository. These higher temperatures will affect the thermal, hydrological, chemical, and mechanical behavior of the waste and EBS. Temperature effects and their associated coupled processes in the waste and EBS, generally, are included in the TSPA as discussed in the *TSPA Disposition.*

Because the primary effects of temperature on waste-form degradation are included explicitly, the secondary effects of temperature on in-package chemistry, such as pH, are of low additional consequence. Therefore, omission of secondary temperature effects on in-package chemistry (such as pH) will not significantly change calculated expected annual dose.

TSPA Disposition: Temperature effects and their associated coupled processes in the waste and EBS, generally, are included in the TSPA. Temperature effects on in-package chemistry such as pH are not directly included since generally thermodynamic data as a function of temperature are lacking. Instead, temperature effects are included directly in the various components (e.g., cladding degradation, CSNF matrix degradation, HLW degradation, and solubility of uranium). For another example, see CRWMS M&O (2000a).

The potential effects of the coupling within the thermal-mechanical-hydro-chemical system are uncertain, but investigations are ongoing (both within the Yucca Mountain Project [YMP] and outside YMP) to delineate the first-order couplings that should be addressed and to define the magnitude of the effects of representing inherently coupled processes by uncoupled, or loosely coupled models. The coupling of the Near-Field Geochemical Environment models to other aspects within the Near-Field Geochemical Environment and other TSPA-VA components is done either in a single direction using output/input data at the process-model level, or by one-way, direct-connection links within the TSPA analyses (CRWMS M&O 1998a, Section 4.2.2). For an example of the coupling at the process level see waste-form component flow charts in the attachments section below.

More specifically for the following secondary FEPS, time dependence (YMP No. 2.1.11.04.05) and long-term transients (YMP No. 2.1.11.04.04) are modeled; heat from radioactive decay (YMP No. 2.1.11.04.03) is included, either calculated or provided as boundary conditions, in fuel assembly, waste package, drift and mountain scale numerical simulations; and unexpected localized temperature effects (YMP No. 2.1.11.04.02) are not included. Other thermally coupled processes are included on an item-by-item basis as discussed in other category 2.1.11 primary FEPs.

Supplemental Discussion: Essentially every physical or chemical process that is likely to occur in the repository can be considered as thermally coupled in some way. However, not all thermally coupled processes are relevant to performance. Depending on repository thermal loading, the near field may not reach temperatures sufficient to drive thermally coupled processes to produce significant effects. However, other types of coupled processes, such as the ambient temperature effects on alkalinity are likely to be significant. (CRWMS M&O 1998a, Section 4.2.2; Hardin 1998, p. 2-1)

The heat generation results primarily from radioactive decay (YMP No. 2.1.11.04.03). It will be dominated by commercial spent nuclear fuel (CSNF) over DSNF/HLW due to both its larger quantity and larger heat generating capacity (DOE 1998d). The waste will be evenly distributed in the repository resulting in a temperature distribution controlled by the CSNF. Some unique aspects of DSNF (inventory and packaging parameters for example) will be modeled but are not expected to significantly affect overall performance.

This FEP is a generic restatement of the 2.1.11 category, thermal processes and conditions in waste and EBS. Many FEPs listed as secondary to this FEP are generic in nature. For example, they are too general to address in detail such as thermal processes within the waste packages (YMP No. 2.1.11.04.01–Thermal (processes) and coupled processes, YMP No. 2.1.11.04.06–Coupled processes (in waste and EBS)). Detailed discussions of more focused temperature/ coupled effects are included in other 2.1.11 FEPs.

6.2.38 Differing Thermal Expansion of Repository Components—YMP No. 2.1.11.05.00

YMP Primary FEP Description: Thermally induced stresses could alter the performance of the waste or EBS. For example, thermal stresses could cause the waste form to develop cracks and create pathways for preferential fluid flow and, thereby, accelerate degradation of the waste. Also, thermal stresses could cause cracks to develop in the backfill or through the drip shield of the EBS

Screening Decision and Regulatory Basis: *Included* (thermally induced stresses in the CSNF waste form and cladding).

Excluded on low consequence (thermally induced stress changes for the near-field barriers and EBS).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.11.07.00	Thermally induced stress changes in waste (includes CSNF cladding) and EBS
2.1.11.01.0	Heat output/temperature in waste (specific to HLW) and EBS

IRSR Issues: ENFE3

Screening Argument: Temperature influences the rate that a material degrades. One potential process is by causing thermally induced stresses in the waste form that, in turn, cause the waste form to develop cracks, thereby, accelerating the degradation of the waste because the active surface area has been increased. The influence of temperature on the degradation rate of CSNF has been observed in experiments and, thus, is included in the CSNF Matrix Degradation Component of the Waste Form Degradation Model. The influence of temperature on the degradation of CSNF cladding and HLW is discussed in FEPs YMP No. 2.1.11.07.00, Thermally induced stress changes in waste and EBS and YMP No. 2.1.11.01.00, Heat Output/Temperature in Waste and EBS, respectively.

Omission of thermally induced stress changes for the near-field barriers and EBS will not significantly change the calculated expected annual dose for the following reasons.

The current drift design minimizes the thermal gradient, and temperatures where differential expansion occurs (due to differences in component/rock properties) will not be reached. To mitigate any possibility of thermal stresses as a result of differing thermal expansion coefficients of the waste-package materials, the waste-package barriers will be constructed with a gap up to 4 mm between the outer barrier (Alloy 22) and inner barrier (316 NG stainless steel).

Thermal expansion induced failure (separation) of near field barriers (i.e. the drip shields) has been screened out because the anticipated change in length is generally much less than the overlap between adjacent drip shields. Thermal expansion of other components, such as the waste package and pedestal, will affect repository performance because the separation between adjacent waste packages is adequate to accommodate this small amount of expansion.

Therefore, thermally induced stress changes for the near-field barriers and EBS have been excluded from consideration in the TSPA based on low consequence.

TSPA Disposition: Thermally induced stresses for the CSNF are not modeled directly. Rather for CSNF, temperature is included as a variable in the regression equation modeling the degradation of CSNF, as described in *CSNF Waste Form Degradation: Summary Abstraction* (CRWMS M&O 2000c). For HLW, temperature is also included since the degradation rate is expressed as an Arrhenius-type rate equation, as described in *Defense High Level Waste Glass Degradation* (CRWMS M&O 2000d).

Supplemental Discussion: Temperature influences the rate that a material degrades. One potential process is by causing thermally induced stresses in the waste form that, in turn, cause the waste form to develop cracks, thereby, accelerating the degradation of the waste because the active surface area has been increased. The influence of temperature on the degradation rate of CSNF and has been observed in experiments as described in the CSNF Waste Form Degradation AMR (CRWMS M&O 2000c). Thus, temperature is included in the CSNF Matrix Degradation Component of the Waste Form Degradation Model. The influence of temperature on the degradation of CSNF cladding and HLW is discussed in FEPs 2.1.11.07.00 (thermally induced stress changes in waste and EBS) and 2.1.11.01.00 (heat output/temperature in waste and EBS), respectively.

6.2.39 Thermally Induced Stress Changes in Waste and EBS—YMP No. 2.1.11.07.00

YMP Primary FEP Description: Repository heat at Yucca Mountain could result in thermally induced stress changes that would affect the mechanical and chemical evolution of the repository. These stress changes could affect both the waste and EBS, thus causing the formation of pathways for groundwater flow through the waste and EBS or altering and/or enhancing existing pathways.

Screening Decision and Regulatory Basis: *Included* (thermally induced stress changes are implicitly included in CSNF cladding failure modes such as creep, SCC, and wet unzipping).

Excluded based on low consequence (thermally induced stress changes in the HLW and DSNF fuel assemblies and packaging).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.07.01.00	Rockfall (large block)
2.1.07.02.00	Drift stability

2.2.01.02.00	Changes in physical properties of the disturbed rock zone around the repository
2.2.06.00.00	Changes in physical properties of the surrounding rock
2.1.02.21.0	Stress corrosion cracking of cladding

IRSR Issues: TSPAI3, ENFE3, CLST1, CLST3

Screening Argument: Repository heat at Yucca Mountain will drive the mechanical and chemical evolution of the repository, producing changes in both the waste and EBS. These changes include thermally induced stress changes that could result in pathways for groundwater flow through the waste and EBS, or it could alter and/or enhance existing pathways.

CSNF cladding, in particular, can fail, or failure zones can propagate due to thermally induced stresses changes within the cladding itself or within the CSNF. Thermal stresses within the CSNF can result in internally applied loads to the cladding.

Waste packages and the repository are designed in such a way that temperature increases are limited to acceptable levels in order to meet thermal design requirements and to maximize repository performance. Therefore, thermally induced stress changes in the waste and packaging are not explicitly considered

Thermal stresses are not of concern for DSNF or HLW. No credit is taken for DSNF cladding because it will probably be breached before it is placed in the repository. HLW experiences harsher temperature environments prior to being placed in the repository so additional thermal stress effects should be minimal. Therefore, omission of thermally induced stresses will not significantly change the calculated expected annual dose and can be excluded from consideration in the TSPA based on low consequence.

TSPA Disposition: Thermally induced stress changes in the HLW and DSNF fuel assemblies and packaging are not included in the TSPA. Thermally induced stresses are implicitly included in the CSNF for early-breach or juvenile-failure parameters because breach of waste packaging, if it occurs, is not likely to occur until after the repository has cooled considerably because of the choice of highly resistant package materials.

Temperature influences on creep perforations, SCC, and wet unzipping of CSNF cladding are included in the CSNF Cladding Degradation Component of the TSPA. During the first 1000 years, an analytical evaluation of the temperature differences in fuel assemblies in a waste-package cross section is used to determine the temperature influence. After 1000 years, when temperatures are much lower, they are expected to be uniform. Therefore, thermally induced stress is implicitly included in cladding failure modeling. See cladding FEPs 2.1.02.19.00, 2.1.02.21.00, and 2.1.02.23.00.

The effects of any thermal-mechanical interactions in the EBS are considered only to the extent that they are included in the near-field thermal-hydrology models.

Supplemental Discussion: Thermally induced changes in the *in situ* stress field that will affect the near-field environment are not relevant to waste-form degradation. See FEPs YMP No. 2.1.11.07.01, Changes in the *in situ* stress field, and 2.1.11.07.02, Stress field changes, settling, subsidence or caving. Waste packages and the repository are designed such that temperature increases are limited to acceptable levels in order to meet thermal design requirements and to maximize repository performance (DOE 1998b, Section 5.1.3.2). Also, static load, thermal stress, and internal pressurization were analyzed together in CRWMS M&O (1997a). Both the stresses in the barriers and the internal pressurization were very low. Even with the static load added in, the barrier stresses were still inconsequential. The waste-package internal basket components were sized not to be stressed by thermal loads (DOE 1998b, Section 5.1.3.3). Therefore, thermally induced stress changes in the waste and packaging are not considered.

6.2.40 Thermal Effects: Chemical and Microbiological Changes in the Waste and EBS—YMP No. 2.1.11.08.00

YMP Primary FEP Description: Temperature changes may affect chemical and microbial processes in the waste and EBS.

Screening Decision and Regulatory Basis: *Included* (thermal effects in various components of in-package chemistry modeling).

Excluded on low consequence (secondary effects such as temperature effects on in-package pH, and thermal effects on microbiological activity).

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.09.00.00	Chemical and geochemical processes and conditions in the waste and engineered barrier system.
2.1.10.01.00	Biological activity in waste and EBS
2.1.02.14.0	Microbiologically Influenced Corrosion of cladding

IRSR Issues: TSPAI3, ENFE3, CLST1, CLST3, CLST4

Screening Argument: The thermal load at Yucca Mountain results from a distributed, exponentially decaying heat source that is inserted over 30–50 years. The hydrogeological response and evolution of Yucca Mountain are driven by the thermal load. Thermal effects will alter the local environment inducing chemical changes. These changes are a function of temperature.

Temperature effects are included directly in various TSPA components such as cladding degradation rate (FEP 2.1.11.07.00–Thermally induced stress changes in waste and EBS), CSNF matrix degradation rate (FEP 2.1.02.02.00–CSNF alteration, dissolution, and radionuclide release), HLW degradation rate (FEP 2.1.02.03.00–Glass degradation, alteration, and dissolution), and solubility of uranium (FEP 2.1.09.04.00–Radionuclide solubility, solubility limits, and speciation in the waste form and EBS).

Chemical and microbiological changes due to temperature can be excluded based on low consequence. Omission of chemical and microbiological changes due to temperature will not significantly change the calculated expected annual dose for the following reasons.

Because primary effects of temperature on waste-form degradation are included directly, the secondary effects of temperature on in-package chemistry such as pH are of low additional consequence. Furthermore, the use of a cooler repository design, drip shield, and long-lived waste package implies that a waste-package breach will occur when temperatures are near ambient. Therefore, thermodynamic data as function of temperature is not necessary.

Preliminary analysis shows that sufficient quantities of microbes will not be available to beneficially affect colloid mobility or adversely accelerate corrosion rates significantly. See the FEP 2.1.10.01.00 for details.

There are no secondary FEPs related to temperature changes affecting chemical and microbial processes in the waste and EBS.

TSPA Disposition: Thermal effects are included in waste/repository chemistry models that determine water and gas composition, precipitation/dissolution, mineral stability, phase equilibrium, and reaction rates where data are available (CRWMS M&O 2000i). For the disposition of temperature effects that are included directly in various TSPA components, see the following FEPs: cladding degradation rate (FEP 2.1.11.07.00), CSNF matrix degradation rate (FEP 2.1.02.02.00), HLW degradation rate (FEP 2.1.02.03.00), and solubility of uranium (FEP 2.1.09.04.00). For details on strongly coupled near-field thermal-geochemical processes and conditions, see FEPs in category 2.1.09 and references (CRWMS M&O 1998a; 1998b).

Thermally driven single-phase flow within the waste is not included in the TSPA as discussed in the *Screening Argument.*

Supplemental Discussion: Two key thermal constraints on microbial growth are relative humidity and temperature thresholds that limit the start of microbial activity until the boiling period is over. Also, the temperature of the subsurface environment will greatly affect or limit the type of bacteria present (i.e., psychrophiles, facultative psychrophiles, mesophiles, thermophiles, and hyperthermophiles), based on the optimum growth band of the microbe. Although microbes in the near-field environment could be sterilized during the highest temperature period, because they are present in the Yucca Mountain water-rock system, they could return as water drips back into potential drifts (CRWMS M&O 1998a, Section 4.2.3.2.4).

Additionally, it is expected that the concrete components in Yucca Mountain will last only a few hundred years, regardless of the decay mechanism. Microbial growth would then be on rubble and could affect the pH of the carrier plume. However, current design does not include extensive use of concrete.

The evolution of microbial communities within the drift are not directly included in the TSPA. Instead, a bounding assessment on the masses of microbes produced will be made. It uses an idealized approach, similar to that of McKinley et al. (1997). This approach uses abiotic processes to determine the rate at which nutrients become available to microorganisms and then reasonably concludes the microorganisms convert those nutrients to their products instantaneously, using limiting guidelines of energy availability and the availability of all the required nutrients in the proper ratio (CRWMS M&O 1998a, Section 4.4.1; CRWMS M&O 1999h, Section 3.3.6).

6.2.41 Thermal Effects on Liquid or Two-Phase Fluid Flow in the Waste and EBS—YMP No. 2.1.11.09.00

YMP Primary FEP Description: High temperatures may result in two-phase flow within the waste and EBS that could influence in-package chemistry. Apart from this effect, temperature gradients may also result in convective flow in the waste and EBS.

Screening Decision and Regulatory Basis: *Included* (thermal effects on two-phase flow of water reaching the EBS and eventually the waste).

Excluded based on low consequence (two-phase flow within the waste and thermally driven single-phase flow within the waste).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.2.10.10.00	Two-phase buoyant flow / heat pipes
2.2.10.11.0	Natural air flow in unsaturated zone

IRSR Issues: TSPAI3, ENFE3

Screening Argument: During the thermal period, a repository in the unsaturated zone (UZ) in Yucca Mountain is expected to develop strong two-phase convective flow in the UZ and possibly a weaker single-phase convective flow in the SZ. Because of repository heat, these matters have received considerable scrutiny CRWMS M&O (2000t; 2000u; 2000v). Yucca Mountain FEPs 2.2.10.10.00, Two-phase bouyant flow/heat pipes, and 2.2.10.11.00, Natural airflow in unsaturated zone, are devoted to the complexities of two-phase flow. If water enters the waste packages during the thermal period, the size of the waste package is such that the temperature will be fairly uniform in the waste-package environment. Consequently, temperature variations that could occur inside the waste package prior to 1000 years (CRWMS M&O 2000w) are not significant enough to cause locally varying flow regimes to arise. Therefore, omission of thermally driven flow within the waste package will not significantly change the calculated expected annual dose and thermally driven flow within the waste package can be excluded form consideration in the TSPA based on low consequence.

DSNF and HLW were evaluated with regard to their effect or contribution to the technical issue(s) discussed in this FEP. There was not any unique or significant effect not already accounted for by CSNF.

TSPA Disposition: Thermal effects on fluid flow to the EBS are included to the extent that they influence the seepage fluxes into the drift. Specifically, the model for evaluating seepage into the drift uses the thermally driven fluxes five meters above the drift perimeter as further described in (CRWMS M&O 2000a, Section 6.1) *Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux.* (See also FEPs 2.2.10.10.00–Two-phase buoyant flow/heat pipes and 2.2.10.11.00–Natural air flow in unsaturated zone, for a discussion of thermally driven flow in the surrounding rocks of the unsaturated zone.)

Because the effects of two-phase flow and the influence of thermal gradients could be neglected, the in-package chemistry was modeled using a uniformly mixed cell; that is, the waste package was not discretized to account for differences in water chemistry caused by two-phase flow and local temperature differences.

Thermally driven single-phase flow within the waste is not included in the TSPA as discussed in the screening argument.

Supplemental Discussion: The amount of fluid flowing into the waste package has a strong influence on the in-package chemistry and, thereby, the alterations of the CSNF matrix, CSNF cladding, and HLW. The amount of fluid entering the waste package, in turn, is influenced by the temperature of the surrounding tuff, so temperature effects are included in those calculations (CRWMS M&O 2000m, Section 6.1). However, after the fluid enters the waste package, the size of the waste package is such that the temperature is fairly uniform in the waste package environment. The temperature variations that could occur inside the waste package prior to 1000 years (CRWMS M&O 2000w) are not significant enough to cause different regimes of water properties. The in-package water chemistry is also insensitive to these small temperature differences that might occur throughout the waste package (CRWMS M&O 2000m). Furthermore, breach of waste packages will most likely occur after 1000 yr (CRWMS M&O 2000x). The temperature gradients are too small to monitor even for cladding creep rupture or stress corrosion cracking (CRWMS M&O 2000w; FEPs 2.1.02.19, 2.1.02.21). Rather, temperatures are uniform and within a few degrees of the waste-package surface and host rock temperatures. Therefore, thermally driven convective fluid flow within the waste are explicitly excluded from the In-Package Chemistry Component of the Waste Form Degradation Model, based on low consequence. FEPs 2.1.11.10.00, Thermal Effects on Diffusion (Soret Effect), and 2.1.09.08.00, Chemical Gradients/Enhanced Diffusion in Waste and EBS, on thermal and chemical diffusion processes, are also excluded in modeling the In-Package Chemistry Component.

6.2.42 Thermal Effects on Diffusion (Soret Effect) in Waste and EBS—YMP No. 2.1.11.10.00

YMP Primary FEP Description: The Soret effect is a diffusion process caused by a thermal gradient. In liquids having both light and heavy molecules (or ions) and a temperature or thermal gradient, the heavier solute molecules tend to concentrate in the colder region. Temperature differences in the waste and EBS may result in a component of diffusive solute flux that is proportional to the temperature gradient.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs: **None**

IRSR Issues: TSPAI3, ENFE3

Screening Argument: The Soret effect is a diffusion process caused by a thermal gradient. In liquids having both light and heavy molecules (or ions) and a temperature or thermal gradient, the heavier solute molecules tend to concentrate in the colder region. Temperature differences in the waste and EBS may result in a component of diffusive solute flux that is proportional to the temperature gradient.

The Soret effect in the waste and EBS can be excluded from consideration in the TSPA based on low consequence. Omission of the Soret effect will not significantly change the calculated expected annual dose because the bounding argument present in the *Basis Screening Decision* concluded that the temperature gradients will be small enough that Soret diffusion effects will be insignificant.

DSNF and HLW were evaluated with regard to their effect or contribution to the technical issue(s) discussed in this FEP. There was not any unique or significant effect not already accounted for by commercial spent nuclear fuel (CSNF).

TSPA Disposition: Diffusion processes (Soret effect) caused by a thermal gradient is excluded from TSPA as described in the *Screening Argument.*

Supplemental Discussion: A detailed discussion of diffusion processes (Soret effect) can be found in *Supplemental Discussion* for this FEP in this AMR.

In addition to concentration gradients, mass diffusion can be affected by temperature gradients (Bird et al. 1960, p 564). Mathematical expressions for mass flux due to temperature gradients and concentration gradients, the following Equations 16 and 17, respectively, are given by Bird et al. (1960, pp. 574 and 575):

$$j_{Az}^{(T)} = -\frac{c^2}{\rho} M_A M_B D_{AB} \frac{k_T}{T} \frac{dT}{dz} \qquad \text{(Eq. 16)}$$

Here, $j_{Az}^{(T)}$ is the diffusive mass flux driven by temperature gradients, and $j_{Az}^{(x)}$ is the diffusive

$$j_{Az}^{(x)} = -\frac{c^2}{\rho} M_A M_B D_{AB} \frac{dx_A}{dz} \qquad \text{(Eq. 17)}$$

mass flux driven by concentration gradients. The remaining symbols are defined as follows:

c: total molar concentration
ρ: total mass density
M_A: molar mass of component A
M_B: molar mass of component B
D_{AB}: binary molecular diffusion coefficient
x_A: mole fraction of component A
z: distance
T: temperature
k_T: thermal-diffusion ratio
z: dimensions in direction of gradient

The thermal-diffusion ratio may be expressed as follows (Bird et al. 1960, p. 568):

$$k_T = \sigma\, x_A x_B T \qquad \text{(Eq. 18)}$$

where σ is called the Soret coefficient and x_B is the mole fraction of component B (Bird et al. 1960, p. 568).

Using Equations 16, 17, and 18, the following ratios can be derived:

$$\frac{j_{Az}^{(T)}}{j_{Az}^{(x)}} = \frac{\sigma\; x_A x_B \dfrac{dT}{dz}}{\dfrac{dx_A}{dz}} = \sigma\; x_A x_B \frac{dT}{dx_A} = \sigma\; x_A (1 - x_A) \frac{dT}{dx_A} \qquad \text{(Eq. 19)}$$

where, for a binary system, $x_B = 1 - x_A$.

Diffusion through the EBS is used as a bounding scenario. This is expected to be a bounding scenario because temperature gradients are largest in the EBS where Soret diffusion may have a more important role. Setting the ratio $j^{(T)}_{Az}/j^{(x)}_{Az} = 1$ and integrating Equation 19 between T_{WP}, x_{AWP}, and T_{DW}, x_{ADW} gives Equation 20:

$$\sigma\;(T_{DW} - T_{WP}) = \ln\left[\frac{x_{ADW}}{1 - x_{ADW}} \frac{(1 - x_{AWP})}{x_{AWP}}\right] \qquad \text{(Eq. 20)}$$

where,

T_{WP} = waste-package surface temperature
x_{AWP} = mole fraction of component A at waste-package surface
T_{DW} = drift wall temperature
x_{ADW} = mole fraction of component A at drift wall

Rearranging Equation 20 to solve for σ, in Equation 21 it is found that the Soret diffusion effect will be negligible when:

$$\sigma \ll \ln\left[\frac{x_{AWP}}{1 - x_{AWP}} \frac{(1 - x_{ADW})}{x_{ADW}}\right](T_{WP} - T_{DW})^{-1} \qquad \text{(Eq. 21)}$$

If component A is the solute mole fraction, then this will typically be very small (<0.01) for dissolved radionuclides (CRWMS M&O 1998c, Chapter 6, pp. T6-23, T6-24), so this result can be approximated by using Equation 22:

$$\sigma \ll \ln\left[\frac{x_{AWP}}{x_{ADW}}\right](T_{WP} - T_{DW})^{-1} \qquad \text{(Eq. 22)}$$

The value of σ for a selection of liquid binary mixtures may be computed from Table 18.4-1 in Bird et al. (1960, p. 569) and Equation 18. The values of σ range are from 0.0008 to 0.016. From CRWMS M&O (1998b, Figures 3-150 a and b), $T_{WP} - T_{DW}$ is no larger than 14°C. This maximum temperature difference is taken at the time when the waste-package surface temperature has returned to 100°C. Temperature differences are larger for earlier times, but water does not exist at the waste-package surface because its temperature exceeds the boiling temperature. Using the largest value for σ (0.016) and the largest temperature difference (14°C), Soret diffusion will be negligible when:

$$x_{AWP} >> 1.25 x_{ADW} \qquad \text{(Eq. 23)}$$

For the EBS diffusion model, x_{ADW} is reasonably concluded to be zero (CRWMS M&O 1998c, p. 6-138), so this criterion will always be met for the EBS diffusion-transport model. Elsewhere in the system (geosphere), temperature gradients are much smaller, so the effects of Soret diffusion are expected to be even less significant there.

6.2.43 Gas Generation—YMP No. 2.1.12.01.00

YMP Primary FEP Description: Gas may be generated in the repository by a variety of mechanisms. Gas may be generated in the fuel assemblies within the closed waste packages as a result of degradation prior to or following breaching of the waste packages or as a result of fuel decay or radiolysis. Degradation of the waste-package exteriors or other components (e.g., drip shield or gantry rails) may also result in gas generation. Additionally, any materials that are chemically sensitive to heating could outgas, and, if any organic chemicals are present, they could vaporize. Gas generation could lead to pressurization of the intact waste packages and the repository and could affect radionuclide transport. This FEP aggregates all types of gas generation into a single category.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.12.02.00	Gas generation (He) from fuel decay
2.1.12.03.00	Gas generation (H_2) from metal corrosion
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation
2.1.13.01.00	Radiolysis
2.1.02.08.04	Flammable gases generation from DSNF

IRSR Issues: TSPAI3

Screening Argument: Gas generation within the waste and repository has been excluded from consideration in the TSPA, based on low consequence. Omission of gas generation will not significantly change the calculated expected annual dose for the following reasons.

Gas generation inside waste-packages due to microbial activity or vaporization of organic substances will not be significant because waste-acceptance requirements prohibit the waste from containing detectable amounts of organic materials. In addition, the waste packages are required to be filled with helium, which displaces water and oxygen, creating an inert environment, thus greatly reducing the likelihood for chemical reactions.

Any gas that may be produced within the repository will not cause the repository to pressurize due to the repository's physical setting. The lithology and structure of the repository host rock and the physical properties of gas as it flows through these units, together mean any gas produced or released from the repository will eventually migrate to the land surface. Gas concentrations will become more dispersed (taking up more volume) and less concentrated as gas flows away from the repository.

TSPA Disposition: Gas generation within the waste and repository is excluded from the TSPA as described in the *Screening Argument*.

Supplemental Discussion: For the purposes of waste-form contribution to gas generation, gas may be generated in the waste in the waste packages prior to the waste-package breaching as well as after breaching, allowing any gas to escape to the repository. Gas generation might lead to pressurization of the waste packages prior to waste-package breaching. After the waste packages are breached, gas generation might lead to pressurization of the repository and affect radionuclide transport. This FEP presents the general argument for the exclusion of gas generation from consideration in the YMP models, based on low consequence.

Repository heating is expected to vaporize the water near the center of the repository, producing large amounts of steam in the early stages and driving out the available oxygen for at least the first 1000 years (DOE 1998d, Section 11.3.2, Table 11-4). This will effectively preclude corrosion in the repository until the fuel cools down sufficiently to allow the surrounding rock to cool below the boiling point. At that point, water will migrate back to the region of the disposed fuel waste packages and permit corrosion of waste-package exteriors, steel set-ground supports, steel inverts, gantry rails, and drip shields to resume. In order for the waste to corrode, water must be present inside the waste package, so the waste package will have to breach sufficiently to allow free water to enter the interior of the waste package.

One mechanism for gas production not related to corrosion is helium production due to fuel decay. Helium will accumulate inside the waste package from the time the waste package is sealed until the waste package breaches due to corrosion or mechanical effects such as rockfall. The pressure buildup inside waste packages due to helium production is not expected to be sufficient to cause structural failure (CRWMS M&O 1998c).

Another possible gas generation process is outgassing of all the materials available in the repository. This is not considered a problem because, even though the repository will heat up enough to boil off water in the vicinity of the waste packages, it will not be hot enough to produce any outgassing of the available materials such as container material and spent nuclear fuel.

There will not be any significant gas generation within waste packages from microbial degradation or from vaporization of organic substances due to the requirements from the Waste

Acceptance System Requirements Document that the "waste form shall not contain detectable amounts of organic materials" (DOE 1999a, p. 16). In addition, the waste packages are required to be filled with helium, which will displace water and oxygen and result in an inert environment, thus not allowing or at least greatly reducing the potential for chemical reactions (DOE 1998b, Section 5.1.2.1).

In the absence of water and organic materials, radiolysis will not produce gas inside waste packages until the waste packages are breached. Aside from juvenile failures, breaching of waste packages will generally occur only after thousands of years following emplacement when the potential for gas generation by radiolysis will be greatly reduced due to decay of the fuel.

Because the repository is an open system, the more mobile gases, such as hydrogen and helium, will diffuse out of the repository first, followed by denser gases such as CH_4 or H_2S (which have lower gas-diffusion rates). Thus, gas will diffuse out of the repository according to the gas density and relative gas-diffusion rate resulting in gas compositions within the repository being relatively homogeneous (CRWMS M&O 1999h, Section 3.3.4). Consequently, corrosive gases will not accumulate nor exacerbate degradation of waste packages and other components.

For the small amount of gas that may be produced (due to the above processes), gas is not expected to cause repository pressures to increase, given the repository's lithologic setting. The repository is situated in the Topopah Spring Welded Tuff, consisting of abundant fracture networks. Here, gas saturation in the matrix varies between 10-30 percent (CRWMS M&O 2000ag, Section 3.6); gas saturation in fractures is higher than matrix saturation. Overlying this unit are the Paintbrush non-welded Tuff (PTn) and the Tiva Canyon welded Tuff (TCw). With the exception of isolated perched zones, the units above the repository site are unsaturated. Each of these units is divided into several sub-units, their classification dependent on the degree of welding, vitrification, the degree and extent of primary and secondary porosity, and mineralogy. Welded tuffs tend to have a high concentration of fracture networks, equating to high fracture permeabilities ranging between $\sim 10^{-11}$ to $\sim 10^{-10}$ m^2, and relatively low matrix permeability, between $\sim 10^{-19}$ to $\sim 10^{-15}$ m^2. The non-welded tuffs have fewer fractures and lower fracture permeabilities than the welded tuffs, ranging from $\sim 10^{-13}$ to $\sim 10^{-11}$ m^2, and higher matrix permeabilities, ranging from $\sim 10^{-12}$ to $\sim 10^{-13}$ m^2 (CRWMS M&O 2000ag). In units where there are numerous fractures and fracture networks, fluid flow (both gaseous and liquid) tends to occur primarily in the fractures.

The relatively high gas saturation of the host rock within and above the repository horizon means that the host-rock gas permeability (i.e., the effective permeability) will be several orders of magnitude greater than the effective liquid permeability. An increase in liquid saturation will reduce gas permeability and increase liquid permeability. Gas flowpaths that encounter perched zones (where liquid saturation approaches 100%) will be deflected at the perched zone perimeter due to the increase in liquid saturation. At unit interfaces, especially between welded and non-welded units, the upward movement of gas will become laterally attenuated and more dispersed as it encounters the relatively larger matrix permeability and storage capacity of the non-welded units. In the TCw unit (at depths of ~100-150-m below the surface) atmospheric pressures will promote barometric pumping of gas to the surface (Ahlers et al. 1999). This pumping will cause mixing of repository-produced gases with atmospheric gases, thus diluting gas concentrations and promoting gas migration to the mountain surface.

Gas transport times will vary depending on repository temperatures. During the thermal period, the thermal gradient between the repository and the far field will promote gas to flow away from the repository relatively quickly, then, due to buoyancy effects, upward towards the surface. Once the repository cools to ambient temperatures, 23 °C (CRWMS M&O 2000af), buoyancy effects will be the dominant driving force, which promotes gas to flow away from the repository. Isotopic dating of gas components containing ^{14}C (Yang et al. 1996) indicates *in situ* gases were ~10,000 years old within the vicinity of the repository horizon. Therefore, once the post-closure thermal period ends, it is expected that the gas mixture(s) originating in the repository will take several thousands of years to reach the mountain surface, and due to dispersion, it will be diluted from its original composition.

6.2.44 Gas Generation (He) from Fuel Decay—YMP No. 2.1.12.02.00

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

Screening Decision and Regulatory Basis: *Excluded* based on low consequence (helium gas production affecting waste package and waste chemistry).

Included (helium gas production for cladding failure).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.12.07.00	Radioactive Gases in Waste and EBS
2.1.02.20.00	Pressurization from Helium production causes cladding failure.

IRSR Issues: TSPAI3

Screening Argument: Helium gas production effects on waste package and waste chemistry have been excluded from consideration in the TSPA based on low consequence. Omission of Helium gas production effects on waste package and waste chemistry will not significantly change the calculated expected annual dose for the following reasons.

Previous studies have demonstrated that an increase in internal pressures from He to be less than about 3.4 MPa over 10,000 years (3.44 MPa at 10,000 years). At early times (<1000 years), He partial pressures are less than about 1 MPa. During later times any helium gas will not affect the cladding failure rates because the clad is sufficiently cool by the time any significant amount of helium is produced.

TSPA Disposition: For the TSPA-SR calculations, the cladding failure rate is partially a function of internal pressures resulting from He production. See discussion given for FEP 2.1.02.20.00 in *Clad Degradation–FEPs Screening Arguments* (CRWMS M&O 2000ae).

Supplemental Discussion: Studies done for TSPA-VA documented in Chapter 6 of the Technical Basis Document (CRWMS M&O 1998c, Section 6.3.1.1.2.5 and Table 6-20) show increases in internal pressures from He to be less than about 3.4 MPa over 10,000 years (3.44 MPa at 10,000 years). At early times (<1000 years), when the repository is relatively hot, He partial pressures are less than about 1 MPa. For late times, CRWMS M&O (1998c, Section 6.3.1.1.5) notes He production will not contribute to waste-package failure.

Helium is an inert gas; therefore, it will not affect in-package chemistry. Pressure from He production in the waste package will not be sufficient enough to cause a waste-package breach.

The arguments presented above are not dependent on the fuel type and, thus, apply to DOE SNF/HLW as well as to Commercial PWR and BWR fuels.

6.2.45 Gas Generation (H_2) from Metal Corrosion—YMP No. 2.1.12.03.00

YMP Primary FEP Description: Gas generation could 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, 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 due to the thermal heating in the repository will produce steam around the waste package, which will exclude oxygen from the iron, thus inhibiting further corrosion for a limited amount of time in the early period of the repository.

Screening Decision and Regulatory Basis: *Excluded,* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.29.00	Flammable gases generation from DSNF
2.1.12.02.00	Gas generation (He) from fuel decay
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation
2.1.13.01.00	Radiolysis
2.1.02.13.00	General corrosion of cladding
2.1.03.01.00	Corrosion of waste containers
2.1.09.02.00	Interaction with corrosion products

IRSR Issues: TSPAI3, CLST3

Screening Argument: The design of the waste packages currently consists of Alloy 22 for the outer shell and an inner shell of 316SS. A major source of hydrogen could be from the corrosion of N-Reactor Metal part of DSNF (CRWMS M&O 2000am). For the waste package and other metals in the repository, the hydrogen that is produced will be an unquantifiable low value.

The effect of hydrogen on the waste package and the cladding is excluded for impact of hydride on cladding and waste-package materials. See FEP 2.1.02.22.00–Hydride Embrittlement of Cladding, which discusses this issue.

H_2 gas generation from metal corrosion has been excluded from consideration in the TSPA, based on low consequence. Omission of H_2 gas generation from metal corrosion will not significantly change the calculated expected annual dose for the following reasons. The hydrogen that could be produced from metal corrosion will be low since the repository is likely to be primarily oxic. Also, the lithology and structure of the repository host rock and the physical properties of gas as it flows through these units, together mean any gas produced or released from the repository will eventually migrate to the mountain surface. Gas will preferentially flow through the already unsaturated, stress-induced and *in situ* fracture networks. Gas will become more dispersed and less concentrated as it moves away from the repository. Therefore, any gas that may escape from the waste packages to the repository will not noticeably affect repository pressures, chemical conditions, and/or fluid flow.

TSPA Disposition: Generation of H_2 gas due to metal corrosion is excluded from the TSPA as described in the Screening Argument.

Supplemental Discussion: The design of the waste packages currently consists of Alloy 22 for the outer shell and an inner shell of 316SS. Alloy 22 was selected because of its enhanced resistance to pitting, crevice corrosion, and stress-corrosion cracking. Additionally, it has a very low (4.0% maximum) iron content. The majority of hydrogen will be generated from the corrosion of N-Reactor Metal, part of DSNF.

The effect of hydrogen on the waste package and the cladding is included on negative impact of hydride on cladding and waste-package materials. See "Hydride Embrittlement of Cladding" (FEP 2.1.02.22.00) which discusses this issue. This FEP is also the topic of AMR, *Hydride-Related Degradation of SNF Cladding Under Repository Conditions* (CRWMS M&O 2000g).

Gas produced by waste-form sources is expected to escape or at least be only temporarily confined beneath the condensate zone above the drifts (CRWMS M&O 1999h, Section 3.2.1). In the early stages of the repository, heating of the environment is expected to produce large amounts of steam, which will drive out hydrogen and the available O_2 between 0 to 2000 years, and as the repository is further heated all water will be driven off from 200 years and 1000 years (DOE 1998c, Section 3.3.3.1). Once the fuel cools down sufficiently to allow the surrounding rock to cool below the boiling point, water will migrate back to the region of the disposed fuel waste packages and permit corrosion to resume. Before the waste can corrode, water will be necessary, and the waste package will have to breach sufficiently to allow free H_2O to enter the interior of the waste package.

The lithology and structure of the repository host rock and the physical properties of gas as it flows through these units, together mean any gas produced or released from the repository will eventually migrate to the mountain surface. Gas will preferentially flow through the already unsaturated, stress-induced, and *in situ* fracture networks. Gas will become more dispersed and less concentrated as it moves away from the repository. Therefore, any gas that may escape from the waste packages to the repository will not noticeably affect repository pressures, chemical conditions, and/or fluid flow.

6.2.46 Gas Generation (CO_2, CH_4, H_2S) from Microbial Degradation—YMP No. 2.1.12.04.00

YMP Primary FEP Description: Microbes are known to produce inorganic acids, methane, organic byproducts, carbon dioxide, and other chemical species that could change the longevity of materials in the repository and the transport of radionuclides from the near field. The rate of microbial gas production will depend on the nature of the microbial populations established, the prevailing conditions (temperature, pressure, geochemical conditions), and the organic or inorganic substrates present. Initial analysis indicates the most important source of nutrient in the YMP repository will be metals. Minimal amounts of organics are mandated by regulation.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence, (effects of gas generation from microbial degradation on in-package waste chemistry).

Included (microbial degradation as affecting the localized corrosion rates in the cladding degradation model).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.14.0	Microbiologically Influenced Corrosion (MIC) of Cladding)

IRSR Issues: TSPAI3, ENFE3, CLST3

Screening Argument: Gas generation from microbial degradation has been eliminated on the basis of low consequence. Omission of the effects of gas generation from microbial degradation on in-package waste chemistry will not significantly change the calculated expected annual dose for the following reasons. Generally speaking, the YMP system is considered to be an open system, and any gas produced will not accumulate in significant amounts to affect the system. There may be some localized effects such as gas accumulation under biofilms. Also, biological activity inside the waste package has been excluded (colloid FEP 2.1.10.01.00) because no organic material will be allowed inside the waste package based on preliminary waste-acceptance-system requirements (DOE 1999c, Section 4.2.3.A.3).

The expected amount of microbes produced from the materials in the repository does not seem to be significant, especially when compared to the masses of materials that are to be used in ground support. Based on this small mass of microbes being generated, effects to the bulk chemistry in the drift are expected to be negligible. Although the bulk chemistry may not be affected by this level of microbial activity, there exists the potential to induce localized impacts to the near-field geochemistry. Microbes are known to produce inorganic acids, methane, organic byproducts, carbon dioxide, and other chemical species that could change the longevity of materials and the transport of radionuclides from the near field.

The sealed waste packages at emplacement will be filled with inert gas, probably helium, prior to emplacement in the Yucca Mountain repository. Due to the absence of oxygen and water, there is a very low probability of gas generation from microbial degradation in an inert gas environment within the package. Once the package is breached, it may be reasonably concluded that any gas produced will migrate to the outside of the package. The lithology and structure of the repository host rock and the physical properties of gas as it flows through these units, together mean any gas produced from microbial degradation will preferentially flow through stress induced fractures and *in situ* fracture networks, eventually reaching the mountain surface. Gas concentrations will become more dispersed and less concentrated as gas moves away from the repository; thus, gas concentrations become diluted before gas could accumulate in quantities of concern (see FEP 2.1.12.01.00).

TSPA Disposition: Gas generation from microbial degradation is excluded in the in-package chemistry model of the TSPA as described in the *Screening Argument*. The possibility of local depression of pH by microbes is included in the localized cladding corrosion rates.

Supplemental Discussion: The expected amount of microbes produced from the materials in the repository does not seem to be significant, especially when compared to the masses of materials that are to be used in ground support. Based on this small mass of microbes being generated, effects to the bulk chemistry in the drift are expected to be negligible. Although the bulk chemistry may not be affected by this level of microbial activity, there exists the potential to induce localized impacts to the near-field geochemistry. Microbes are known to produce inorganic acids, methane, organic byproducts, carbon dioxide, and other chemical species that could change the longevity of materials and the transport of radionuclides from the near field (CRWMS M&O 1998a, Section 4.6.2.3.3.4).

The sealed waste package at emplacement will be filled with inert gas, probably helium, prior to emplacement in the Yucca Mountain repository (DOE 1998b, Section 5.1.2.1). Due to the absence of oxygen and water, there is a very low probability of gas generation from microbial degradation in an inert gas environment within the package. There may be some localized effects such as gas accumulation under biofilms. Microbes will use the nutrients available in the drifts from chemical oxidation and reduction reactions. Microbial gas generation from bacteria (e.g., Fe-philic bacteria) in Yucca Mountain is expected to use waste packages as the predominant growth substrate (DOE 1998c, Volume 3, Section 3.3.1.3). Hydrogen gas (H_2) is the most likely gas that could be produced. The predominant source of material for microbes will be Fe in the waste packages. There will not be a large microbial community in the aggregate of the waste repository, but locally microbial degradation may be a factor. Locally under biofilms, there may be significant generation and accumulation of gas a result of microbial degradation. Overall, gas production from microbial degradation should not be a factor (CRWMS M&O 2000h). The possibility of local depression of pH by microbes is included in local corrosion rates for the CSNF cladding (see YMP No. 2.1.02.14.0, Microbiologically Influenced Corrosion (MIC) of Cladding).

Once the package is breached, it may be reasonably concluded that any gas produced will migrate to the outside of the package and, due to the air permeability of Yucca Mountain, will be diluted before it could accumulate in quantities of concern (DOE 1998b, Section 11.3.2). Due to differences in the properties of gases and water (viscosity, density, buoyancy, diffusion, and advection capability), the effective permeability will be significantly different. Even though the relative permeability for liquid will typically be much less than for gas in fractures (because of relatively low liquid saturation), the overall conductivity for water flow is still very large and is accounted for and is included in the (UZ) model

Organic material in the waste is another potential source of nutrients for microbes. Current waste acceptance requirements do not permit detectable amounts of organics, *Waste Acceptance System Requirements* (DOE 1999a, Section 4.2.3). Currently, there are no plans to dispose of organic materials at Yucca Mountain, thus eliminating a source of material for microbial degradation. The small amount of organics that may be present, such as oil films and incidental detrital organics that enter the repository, will be negligible.

DSNF and HLW were evaluated with regard to their effect or contribution to the technical issue(s) discussed in this FEP. There is not any unique or significant effect not already accounted for by CSNF.

6.2.47 Gas Transport in Waste and EBS—YMP No. 2.1.12.06.00

YMP Primary FEP Description: Gas in the waste and EBS could affect the long-term performance of the disposal system. Radionuclides may be transported as gases or in gases, gas bubbles may affect flowpaths, and two-phase flow conditions may be important.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.12.01.00	Gas generation
2.1.12.02.00	Gas generation (He) from fuel decay
2.1.12.03.00	Gas generation (H_2) from metal corrosion
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation
2.1.12.07.00	Radioactive Gases in Waste and EBS
2.1.12.08.00	Gas Explosions

IRSR Issues: CLST3, CLST4

Screening Argument: Transport of radionuclides in the gas phase has been shown to be insignificant in human dose calculations (see *Supplemental Discussion*) and, therefore, has been excluded from the TSPA on the basis of low consequence. Omission of transport of radionuclides in the gas phase will not significantly change the calculated expected annual dose for the following reasons. It is conservatively presumed that transport of any radionuclides that can be partitioned between both the aqueous and gaseous phase, such as ^{14}C, will have higher doses if transport is considered only in the aqueous phase. Therefore, transport of any gases containing ^{14}C as a constituent will be partitioned as ^{14}C (as $^{14}CO_2$) in the aqueous phase.

TSPA Disposition: Radionuclide transport in the gas phase or as a component of gas mixtures is excluded from the TSPA as described in the *Screening Argument*.

Supplemental Discussion: The only normally occurring gases of concern at Yucca Mountain are $^{14}CO_2$, various radioactive fission gases (including tritium), and radon (CRWMS M&O 2000j). Tritium in the vapor phase is screened out of future TSPA calculations based on a short half-life and low consequence. All radioactive gases have been screened out of the TSPA-SR based on low consequence except for carbon dioxide (CRWMS M&O 2000j). Carbon dioxide was considered as a special case where ^{14}C could be a constituent. Dose calculations were performed where $^{14}CO_2$ was transported in the gas phase. Results indicated ^{14}C doses (as $^{14}CO_2$), five orders of magnitude below the limit proposed in 40 CFR SS 197.4; releases for $^{14}CO_2$ partitioned in the aqueous phase were higher (see FEP 3.2.10.00.00 described in the TSPA System-Level FEPs AMR, CRWMS M&O 2000r, for details). This analysis presumed an instantaneous release of ^{14}C gas to the critical group (EPA 1999, Section 9.2.4) with mitigation prevented by geologic barriers. Consequently, transport of ^{14}C in the gaseous phase is screened out, based on low consequence. Mass is conserved by assigning any ^{14}C that would be in the gas phase to the aqueous phase CRWMS M&O (2000r). Given the above analysis, it appears that any short-circuit transport mechanisms through the waste or EBS, given the presence of bubbles or other two-phase flow mechanism, to ^{14}C gaseous components can also be screened out, based on low consequence. (Note: for the TSPA-SR calculations, it is conservatively expected that all ^{14}C will be partitioned in the aqueous phase.)

6.2.48 Radioactive Gases in Waste and EBS—YMP No. 2.1.12.07.00

YMP Primary FEP Description: Radioactive gases may exist or be produced in the repository. These gases may subsequently escape from the repository. Typical radioactive gases include ^{14}C (in $^{14}CO_2$ and $^{14}CH_4$) produced during microbial degradation, tritium, fission gases (Ar, Xe, Kr), and radon.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence (transport of radioactive gases).

Excluded based on low consequence (impact of radioactive gas partial pressures on the waste-package failure rates).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.12.01.00	Gas generation
2.1.12.02.00	Gas generation (He) from fuel decay
2.1.12.03.00	Gas generation (H_2) from metal corrosion
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation
2.1.12.07.00	Radioactive Gases in Waste and EBS
2.1.12.08.00	Gas Explosions

IRSR Issues: CLST3, CLST4

Screening Argument: At the time of waste emplacement, partial pressures of radioactive gases resident in the cladded fuel rods (and/or released from the fuel rods to the waste packages) will not be high enough to breach a waste package. During postclosure times, radioactive-gas production will not be sufficient to cause waste-package internal pressures to increase to the extent that a waste-package breach will occur. The potential human dose from inhalation of any resident radioactive gases that may reach the accessible environment falls below the inventory-screening criterion. Therefore, omission of radioactive-gas effects will not significantly change the calculated expected annual dose. Thus, the impacts of radioactive gases on waste-package integrity and/or inhalation doses are screened out of the TSPA-SR calculations, based on low consequence. Transport of $^{14}CO_2$ in the gas phase is also excluded in the TSPA-SR calculations, based on low consequence (see FEP argument 3.2.10.00.00).

Note: all $^{14}CO_2$ produced in the waste will be considered as dissolved and transported in solution (see CRWMS M&O 2000j; 2000ai; FEP argument 3.2.10.00.00).

TSPA Disposition: Transport of and partial pressures of radioactive gases has been excluded in the TSPA as described in the *Screening Argument.*

Supplemental Discussion: The major gas constituents trapped inside a waste package prior to disposal will be the result of fission-product and neutron-activated stable and radioactive noble gases (CRWMS M&O 1998c). Because of burn-up, these radioactive gases will be predominantly isotopes of Ar, Kr, and Xe. There are several arguments that support the exclusion of radioactive gases from the TSPA-SR calculations, based on low consequence to waste-package internal pressures and low consequence with respect to dose. These are given below.

1. Those radioactive gases resident in the fuel rods and waste packages prior to postclosure or produced after postclosure times will either decay rapidly or quickly become negligible in terms of mass and volume. Specifically:

   - Xenon (^{135}Xe), a fission yield product, is short lived (9.2 hours) and will not be produced during the regulatory time periods. It will undergo decay to its long-lived daughter ^{135}Cs by the time of waste emplacement.

   - Argon (^{39}Ar) has a low activity and is screened out due to low consequence. It will not be produced during the regulatory time period.

   - Radon gas (^{222}Rn and ^{219}Rn) is short lived (~3.8 days and ~4 seconds, respectively) as are its gaseous daughters (^{218}Po, half-life ~3.1 minutes and ^{215}Po, half-life ~microseconds) and will, thus, be in secular equilibrium with its parents, radium and thorium (^{229}Th). Radon has been screened out of the TSPA-SR due to low consequence (CRWMS M&O 2000j).

   - $^{14}CH_4$ and $^{14}CO_2$ generation from microbial degradation should not be produced because there are essentially no organics in the waste (DOE 1999a).

   - $^{14}CO_2$ will not be created as long as there is an anoxic environment because the waste package will be filled with helium. Once a waste package is breached, an oxidizing environment may be created within the waste package, thus enabling $^{14}CO_2$ to be formed. The limited amount of ^{14}C remaining in the inventory, coupled with the low package-failure rate, limits the potential creation of $^{14}CO_2$ to very small amounts. Furthermore, the amounts of ^{14}C transported to the accessible environment as a gas will be extremely small, so the risk consequence is negligible. See FEP 3.2.10.00.00, System Level FEPS (CRWMS M&O 2000r).

- Of the noble gases, ^{85}Kr has a significant initial inventory, but because of its short half-life (~10 yr), its concentration rapidly becomes insignificant.

2. Partial pressures from radioactive gas will not be great enough to compromise waste-package integrity that would result in a waste-package breach. This argument is supported by several numerical studies investigating the sensitivity of internal waste-package pressures presuming fuel rods rupture, thus causing a release of gases from the fuel matrix to the waste-package interior (CRWMS M&O 1997c; 2000aq; 2000ar). These studies conservatively estimate that the maximum pressures within the cladded fuel-rod assemblies would rise to 8.4 MPa, thus inducing clad failure. Once cladding ruptured, all radioactive and fission gases migrated from the fuel matrix to the waste-package voids. Waste-package interior pressures were determined for cases presuming 1, 10, and 100% of all fuel rods rupture, and rupture occurs at temperatures ranging from 20 to 600 °C. An additional case presumes all fuel rods rupture, and all fuel pellets have a 50% higher burn-up. Consequently, internal pressures would be greater than assemblies experiencing normal burn-up conditions by a factor of 1.5. CRWMS M&O (2000aq; 2000ar) took results from the fuel-rod-rupture studies, cited above (CRWMS M&O 1997c) and determined the maximum stress pressures imposed on the waste-package interior shells for given waste-package interior pressures.

   Results from these studies show interior waste-package pressures for normal burn-up rates range between 0.1-.067 MPa (see Table 7); in the high burn-up cases (using the 1.5 multiplier), waste-package pressures range between 0.39-1.41 MPa. The peak interior pressure of 1.41 MPa imposes a material stress pressure of 195 MPa on the shell (CRWMS M&O 2000aq). This stress pressure is much less than 384.7 MPa, the critical stress pressure required to cause the waste package outer shell to fail (CRWMS M&O 2000ar).

Table 7. Waste-Package Internal and Material Pressures [a],[b]as a Function of Temperature and Percent Breached Rods for Both 21 PWR and 44 BWR Fuel Assemblies

Temperature (°C)	Waste-Package Interior Pressure and Shell Stress Pressures (MPa) for Percentages of Rods Breached				
	Waste Package Shell Stress Pressures [b]	**1.5 x 100 % Rods Breached PWR/BWR**	**100 % Rods Breached PWR/BWR**	**10 % Rods Breached PWR/BWR**	**1 % Rods Breached PWR/BWR**
20[b] 25	63/*66*[b]	**0.38**[b] 0.39/*0.29*	0.26/*0.19*	0.12/*0.11*	0.10/*0.10*
50		0.42/*0.30*	0.28/*0.20*	0.13/*0.12*	0.11/*0.11*
100		0.50/*0.35*	0.33/*0.23*	0.15/*0.14*	0.13/*0.13*
200	102/*107*[b]	0.62/*0.45*	0.41/*0.30*	0.19/*0.17*	0.16/*0.16*
300		0.75/*0.54*	0.50/*0.36*	0.23/*0.21*	0.20/*0.20*
350 400[b]	134/*140*[b] 145/*151*[b]	0.81/*0.59* **0.88**[b]	0.54/*0.39*	0.25/*0.39*	0.22/*0.21*
500 600[b]	 187/*195*[b]	1.01/*0.72* **1.14**[b]	0.67/*0.48*	0.30/*0.28*	0.27/*0.27*

[a] *Waste Package Design Basis Events* (CRWMS M&O 1997c)

[b] *Internal Pressurization Due to Fuel Rod Rupture in Waste Packages* (CRWMS M&O 2000aq)

The above studies are upper-limit bounded because of the following:

- It is estimated that waste-package temperatures will not exceed 350 °C, well below the 500 °C used in the CRWMS M&O (1997c) analysis.
- Waste-package breach is expected to occur when temperatures are at, or below, 100 °C.
- Not all the fuel rods will rupture within a waste package within the regulatory time frame.
- Fission and radioactive-gas diffusion rates in the fuel matrix are very low, approximately 3 x 10 –31 m2/sec at 27° C (CRWMS M&O 2000ak). Therefore, fission gas released from the matrix will not all migrate to 'free' void space, especially at temperatures approaching 27 °C, where migration of fission gases, including radioactive gases, out of the matrix approaches 0.0 (CRWMS M&O 1997c). (For the fraction of fission gas release from the fuel matrix to waste-package interior adopted for the TSPA-SR, see CRWMS M&O 2000c.)

3. A third argument provides an upper-limit quantification of fission gas partial pressures most likely to be exerted on waste-package interiors. This analysis shows that the contribution of radioactive-gas partial pressures to the total waste-package pressure is small. For this analysis, an upper-limit estimate of radioactive-gas partial pressures in a waste package is determined by using as input the highest activity per CSNF fuel assembly provided in the Source Term Generation and Evaluation reports for both PWR (CRWMS M&O 1999k,

Attachment X) and BWR (CRWMS M&O 1999j, Attachment XV) assemblies. In CRWMS M&O (1999k), a conservative upper limit for potential fission and radioactive products was a PWR fuel matrix with a 5% enrichment and a burn-up rate of 75 Giga-Watt days per Metric Ton Uranium (GWd-MTU). In CRWMS M&O (1999j), a conservative upper limit was a BWR fuel matrix with a 4.5% enrichment and a burn-up of 70 GWD-MTU.

An additional consideration is whether PWR or BWR assemblies will produce the highest activity using these bounding values for radioactive gases, given in Table 8 as activity in Curies per assembly per second, and also as disintegrations per assembly second, DPS. From these activities, the number of radioactive gas moles per assembly can be calculated using the following equation:

$$\frac{moles}{assembly} = \underset{given}{\frac{curies}{assembly}} \bullet \underset{\text{conversion}}{\frac{DPS}{1 \cdot curie}} \bullet \underset{(DPS/gram)}{\frac{1}{Ni\lambda}} \bullet \underset{(grams/mole)}{\frac{1}{AtWt}} \qquad \text{(Eq. 24)}$$

Where

1 Curie = 3.7 x 10^{10} disintegrations per second (DPS)

The decay rate (N$i\lambda$) per gram element, or disintegration per second (DPS) per gram element, is defined in Equation 25 as:

$$\mathrm{N}i\lambda = \frac{\text{disintegrations}}{(\text{gram} \cdot {}_{\text{element}})(\text{second})} \qquad \text{(Eq. 25)}$$

Where

N = number of molecules per gram element (# nuclei/gram$_{\text{element i}}$)

$$= \frac{6.022x10^{23}\, molecules}{mole} \bullet \underset{AMUi}{\frac{mole}{grams}}$$

$\lambda_{\iota =}$ disintegrations (decay constant) per element i = $\dfrac{\ln 2}{\tau_{\cdot 1/2}}$

$\tau_{1/2}$ = half-life per element

Half-lives, decay constants, and decay rates for each element considered are given in Table 9.

Table 8. Activity of Radioactive Gases per CSNF Assembly

CPAS = Curies per assembly per second (taken from the Inventory Abstraction AMR–CRWMS M&O 2000j)								
DPAS = Disintegrations per assembly per second								
Activity	**CPAS**	**DPAS**	**CPAS**	**DPAS**	**CPAS**	**DPAS**	**CPAS**	**DPAS**
Time (years)	**100**		**500**		**1,000**		**10,000**	
Reactor Type **Isotope**	*PWR Average*							
PWR Bounding[1]								
^{39}Ar	***6.10E-05***	2.26E+06	***2.18E-05***	8.07E+05	***6.00E-06***	2.22E+05	***5.08E-16***	1.88E-05
^{14}C	***4.83E-01***	1.79E+10	***4.60E-01***	1.70E+10	***4.33E-01***	1.60E+10	***1.46E-01***	5.40E+09
^{3}H	***1.71E+00***	6.33E+10	***2.92E-10***	1.08E+01	***1.82E-22***	6.72E-12	***0.00E+00***	0.00E+00
^{85}Kr	***8.75E+00***	3.24E+11	***5.11E-11***	1.89E+00	***4.65E-25***	1.72E-14	***0.00E+00***	0.00E+00
^{219}Rn	***4.73E-05***	1.75E+06	***8.35E-05***	3.09E+06	***1.31E-04***	4.85E+06	***9.94E-04***	3.68E+07
^{220}Rn	***2.53E-02***	9.36E+08	***4.79E-04***	1.77E+07	***5.93E-06***	2.19E+05	***2.61E-06***	9.66E+04
Total Activity	**1.10E+01**	**4.06E+11**	**4.61E-01**	**1.70E+10**	**4.33E-01**	**1.60E+10**	**1.47E-011**	**5.44E+09**
BWR Bounding[2]								
^{39}Ar	***2.27E-05***	8.40E+05	***8.09E-06***	2.99E+05	***2.23E-06***	8.25E+04	***1.89E-16***	6.99E-06
^{14}C	***3.12E-01***	1.15E+10	***2.97E-01***	1.10E+10	***2.80E-01***	1.04E+10	***9.41E-02***	3.48E+09
^{3}H	***6.39E-01***	2.36E+10	***1.09E-10***	4.05E+00	***6.77E-23***	2.50E-12	***0.00E+00***	0.00E+00
^{85}Kr	***3.16E+00***	1.17E+11	***1.85E-11***	6.85E-01	***1.68E-25***	6.22E-15	***0.00E+00***	0.00E+00
^{220}Rn	***8.78E-03***	3.25E+08	***1.66E-04***	6.14E+06	***1.92E-06***	7.10E+04	***7.83E-07***	2.90E+04
Total Activity	**4.12E+00**	**1.52E+11**	**2.97E-01**	**1.10E+10**	**2.80E-01**	**1.04E+10**	**9.41E-02**	**3.48E+09**

[1] CRWMS M&O (1999k. Attachment X)

[2] CRWMS M&O (1999j. Attachment XV)

Table 9. Half-lives, Decay Constants (λ), and Activitiy (λ *N*) for the Radioactive Gases ^{39}Ar, ^{14}C, 3H, ^{85}Kr, ^{219}Rn, and ^{220}Rn

Element	Half-life (seconds)	λ ((disintegrations/second)	λN_I (disintegrations /second-gram)
^{39}Ar	8.48E+09	8.17E-11	1.26E+12
^{14}C	1.81E+11	3.84E-12	1.65E+11
^{3}H	3.89E+08	1.78E-09	3.58E+14
^{85}Kr	3.40E+08	2.04E-09	1.44E+13
^{219}Rn	3.96E+00	1.75E-01	4.81E+20
^{220}Rn	5.56E+01	1.25E-02	3.41E+19

The number of radioactive gas moles can be determined from this projected radioactive gas inventory. From moles, partial pressure for each radioactive gas is derived using the ideal gas law and by rearranging, solving for partial pressure for each radioactive gas constituent given in Equation 26 as follows:

$$P(\mathit{partial}\ \text{pressure}) = \frac{R \bullet T \bullet n}{V(\text{void volume waste package})} \qquad \text{(Eq. 26)}$$

Where

n= number of gas moles

R = Gas constant (0.08206 L Atm/ K mol)

T = Temperature in degrees Kelvin

and

1 Atm = 0.101325 MPa

Temperature Kelvin = Temperature Centigrade + 273.15

Void volume of BWR waste package = 4.43 m^3

Using the highest activity per assembly as input, an upper limit (which equates to largest concentration) for radioactive gas moles can be calculated. Bounding partial pressures for each gas can then be calculated presuming all waste packages are comprised of BWR fuel packages. This type of reactor is selected because the BWR packages contain the largest number of fuel assemblies (44) per waste package. Thus, it will have the largest radioactive gas concentrations and partial pressures.

Table 8 shows that the highest activities per assembly are from PWR fuels. Using these values as input, the upper limit for partial pressure for each radioactive gas is derived using Equations 2 and 3. An upper limit (and bounding values) for radioactive-gas partial pressures is derived by conservatively considering (1) all fuel assemblies have the same activity as PWR bounding set (given in Table 8, (2) all radioactive gases migrate out of the fuel matrix to waste-package interiors, (3) all fuel rods rupture within each assembly, and (4) there are 44 assemblies per waste package.

Table 10 shows that the partial pressure contribution for each radioactive and all radioactive gases taken together is negligible. Thus, radioactive gas partial pressures will not compromise waste-package integrity and are screened out of the TSPA-SR calculations based on low consequence. Defense and DOE SNF and HLW will have less radioactive fission gas generation than CSNF waste. Therefore, because CSNF generated radioactive fission gases is excluded from the TSPA-SR calculations, it logically follows that DSNF and DHLW radioactive gases can be excluded from the TSPA-SR calculations using the same arguments.

Note; for He, a non-radioactive gas, see FEP YMP No. 2.1.12.02.00–Gas Generation (He) from Fuel Decay.

Table 10. Estimated Partial Pressure for Radioactive Gases at Selected Times after Repository Closure for Selected Repository Temperatures

100 years After Closure–Repository Temperatures Estimated to be 120 °C		
Isotope	Atm	MPa
^{39}A	1.47E-08	1.49E-09
^{14}C	2.48E-03	2.51E-04
^{3}H	1.89E-05	1.91E-06
^{85}Kr	8.45E-05	8.56E-06
^{219}Rn	5.32E-18	5.39E-19
^{220}Rn	4.00E-14	4.05E-15
Total Partial Pressures from All Radioactive Gases	2.58E-03	2.62E-04
500 years After Closure–Repository Temperatures Estimated to be 110 °C		
Isotope	Atm	MPa
^{39}A	1.38E-19	1.40E-20
^{14}C	6.22E-14	6.30E-15
^{3}H	8.50E-26	8.61E-27
^{85}Kr	1.30E-26	1.32E-27
^{219}Rn	2.47E-28	2.51E-29
^{220}Rn	1.99E-26	2.02E-27
Total Partial Pressures from All Radioactive Gases	6.22E-14	6.30E-15
10,000 years After Closure–Repository Temperatures Estimated to be 60 °C		
Isotope	Atm	MPa
^{39}A	1.04E-30	1.06E-31
^{14}C	1.11E-14	1.12E-15
^{3}H	0.00E+00	0.00E+00
^{85}Kr	0.00E+00	0.00E+00
^{219}Rn	2.02E-30	2.04E-31
^{220}Rn	0.00E+00	0.00E+00
Total Partial Pressures from All Radioactive Gases	1.11E-14	1.12E-15

NOTE: *Internal Pressurization Due to Fuel Rod Rupture in Waste Packages* (CRWMS M&O 2000aq); *Design Analysis for UCF and Waste Packages* (CRWMS M&O 2000ar)

6.2.49 Gas Explosions—YMP No. 2.1.12.08.00

YMP Primary FEP Description: Explosive gas mixtures could collect in the sealed repository. An explosion in the repository could have radiological consequences if the structure of the repository were damaged or near-field processes enhanced or inhibited.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No. FEP Name

2.1.12.01.00	Gas generation
2.1.12.03.00	Gas generation (H_2) from metal corrosion
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation
2.1.12.07.00	Radioactive Gases in Waste and EBS
2.1.12.08.00	Gas Explosions

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: Explosive gases may be produced in the waste. Gas explosions could result if sufficient gases are produced and sufficient oxygen is available for an explosive environment. However, the probability of an ignition source being present is low. The availability of oxygen would be limited for combustion and greatly reduce the corrosion of containers, thereby reducing the number of containers potentially producing flammable/explosive gases. For these reasons, post-closure explosions in the drifts resulting from ignition of flammable gases are excluded from the TSPA-SR/LA based on low consequence. Omission of explosive gases and their possible effects will not significantly affect the time or magnitude of the calculated annual dose. Furthermore, oxygen for detonation would not be available prior to breach of the disposal container.

Examples of gases that may be present are hydrogen (YMP No. 2.1.12.03.00), methane (YMP No. 2.1.12.04.00), and acetylene (YMP No. 2.1.02.08.). Hydrogen would be produced from the radiolysis of water H2O as described in YMP No. 2.1.13.01.00. However, there is virtually no H2O in waste form until canister breach. Methane could be produced from the microbial action on organics, if they were available, or the metal containers. Acetylene would be produced when moisture comes in contact with the carbide component of the uranium carbide DOE fuel. The volume of DSNF is a small percent of the total waste inventory and is dispersed among the CSNF. Therefore, the possibility of gas explosions are low and can be excluded from consideration in the TSPA based on low consequence. Details can be found in the Supplemental Discussion. In brief, it is unlikely that any significant amount of explosive gases will form. If they do, they will be diluted, diffused, and/or dispersed before they could reach explosive concentrations.

TSPA Disposition: Gas explosions have been excluded from consideration in the TSPA as discussed in *Screening Argument* and *Supplemental Discussion.*

Supplemental Discussion: Explosive gases may be produced in waste. Examples of gases that may be present are hydrogen (H_2) (YMP No. 2.1.12.03.00), methane (CH_4) (YMP No. 2.1.12.04.00), and acetylene (C_2H_2) (YMP No. 2.1.02.29.00). Hydrogen would be produced from the radiolysis of water (H_2O) (YMP No. 2.1.13.01.00). Methane could be produced from the microbial action on organics or the metal containers. Acetylene would be produced when moisture comes in contact with the carbide component of the uranium carbide DOE fuel. Because the volume of DSNF is a small percent of the total waste inventory and is dispersed among the CSNF, the dominant or controlling factor is the commercial SNF (DOE 1998b, ES.3). Acetylene from breached DSNF packages will be diluted, and its concentration likely will be below threshold for combustion.

The sealed waste package will be filled with inert gas, helium, prior to emplacement in the Yucca Mountain repository (DOE 1998a, Section 5.1.2.1, pp. 5-6). Because of the absence of oxygen and water, there is a very low probability of explosion in an inert gas environment within the waste package until the package is breached.

After the waste package is breached (DOE 1998a, Section 5.1.4, pp. 5-40), it may be reasonably concluded that any gas produced will migrate to the outside of the canister. Generally, permeability of Yucca Mountain to air will provide an adequate condition for the flammable/explosive gases (e.g., H_2, CH_4, and C_2H_2) to be diluted, diffused, and/or dispersed before they could reach explosive concentrations. Due to differences in the properties of gases and water (viscosity, density, buoyancy, diffusive, and advective capability) the effective permeability will be significantly different. Although the relative permeability for liquid will typically be much less than that for gas in fractures (because of relatively low liquid saturation), the overall conductivity for water flow will still be very large and is accounted for and is included in the (UZ) model.

Possibly gases, as well as water, could accumulate within the drift area, if there was a condensation cap or reduced permeability (CRWMS M&O 1999h, Section 3.2.3). Changes to the fluid-flow characteristics of the flow system in the mountain could be produced by thermally driven mechanical and chemical processes (CRWMS M&O 1999h, Section 3.2.1). However, the probability of an ignition source being present is low. The possibility of reduced permeability to gas would also limit the availability of oxygen for combustion and greatly reduce the corrosion of containers, thereby reducing the number of containers potentially producing flammable/explosive gases.

For these reasons, post-closure explosions in the drifts resulting from ignition of flammable gases are excluded from the TSPA-SR/LA.

6.2.50 Radiolysis—YMP No. 2.1.13.01.00

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

Screening Decision and Regulatory Basis: *Excluded* on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.12.03.00	Gas generation (H_2) from metal corrosion
2.1.13.02.00	Radiation Damage in Waste and EBS
3.1.01.01.00	Radioactive Decay and In-growth
2.1.02.15.00	Acid corrosion of cladding from radiolysis

IRSR Issues: TSPAI3, CLST3

Screening Argument: The effects of radiolysis for gas production and waste-form chemistry can be excluded from consideration in the TSPA based on low consequence. Omission of H_2 gas generation from metal corrosion will not significantly change the calculated expected annual dose. Alpha, beta, gamma, and neutron irradiation of water leads to formation of highly reactive excited and ionized species which in turn can undergo various reactions. In pure water, the final products are hydrogen and oxidants. In addition, the oxidants formed may react with dissolved iron (+2) which will decrease the net yield of oxidants. However, water is not expected to affect the fuel until all, except potentially alpha radiolysis and perhaps gamma, have become negligible. Shielding calculations can show that intact cladding will stop most alphas so that alpha radiolysis will not occur during the early periods of highest alpha activity. Additionally, the rate of corrosion effects of used UO_2 fuel due to alpha radiolysis, taking no credit for cladding, can be predicted (based on semi-empirical methods) to have minor consequence. Details provided in *Supplemental Discussion.*

TSPA Disposition: Radiolysis of the waste form will not be modeled in the TSPA calculations as described in the *Screening Argument.*

Supplemental Discussion: Interior to the waste package, water will not intrude (i.e., the waste container will not fail) until gamma and beta emitters have decayed to very low concentrations (Sunder and Shoesmith 1991). According to Sunder and Shoesmith, "strong gamma and beta fields associated with the used fuel will decrease by a factor $>10^3$ in the first few hundred years after disposal...groundwater reaching the fuel after this period will be subjected mainly to alpha radiolysis."

The use of presently available used fuels to study the effects of alpha radiolysis on fuel dissolution is inappropriate because they have strong beta and gamma fields associated with them. However, Sunder et al. (1997) describe an experimental strategy for determining fuel dissolution rates as a function of alpha-source strength, and they show how the evolution of corrosion behavior can be predicted as a function of the age of the fuel. Sunder et al. (1997) conclude that "predictions presented...suggest the effects of alpha radiolysis on fuel corrosion (dissolution) will be transitory and will become minor as alpha dose rates decrease."

During the periods of highest alpha activity, it is expected that, for commercial fuel, most of the clad will remain intact and should substantially reduce alpha dose rates to groundwater (Kaplan 1963, p. 307). This can be shown in a simple shielding calculation for alpha radiation:

$$R = \int_0 dE/S(E) \qquad \text{(Eq. 27)}$$

where E is the kinetic energy, E_0 is the initial kinetic energy, and S(E) is the stopping power as a function of the kinetic energy gives the range, R, of a charged particle in a given material (Kaplan 1963, p. 314). The stopping power function depends on the material. Instead of using the stopping power, it may be useful to use a relative stopping power (to air) since the range of an alpha particle in air is only a few centimeters. The stopping power of metals is at least three orders of magnitude greater than air; thus clad of thickness of a few microns would stop alphas.

Additionally, YMP FEP 2.1.13.02.00 had these two net findings:

1. The α-radiolysis corrosion rate is three orders of magnitude less than the dissolution rate.

2. The α-radiolysis-enhanced corrosion rates for HLW metallic carbide, and ceramic spent nuclear fuels are much lower than their dissolution rates.

For DSNF, this argument becomes a non-issue since no credit is being taken for the cladding. However, the argument by Sunder et al. (1997) does not include the protection of clad against dissolution.

6.2.51 Radiation Damage in Waste and EBS—YMP No. 2.1.13.02.00

YMP Primary FEP 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 spent nuclear fuel. If these radicals migrate (diffuse) to the surface of the fuel, they may then enhance the degradation/corrosion rate of the fuel (UO_2). This effect would increase the dissolution rate for radionuclides from the fuel material (fuel meat) into the groundwater flow.

Screening Decision and Regulatory Basis: *Excluded* based on low consequence.

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.12.03.00	Gas generation (H_2) from metal corrosion
2.1.13.01.00	Radiolysis
3.1.01.01.00	Radioactive Decay and In-growth
2.1.02.15.00	Acid corrosion of cladding from radiolysis

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: The effects of radiolysis-enhanced dissolution of spent nuclear fuel has been excluded from consideration in the TSPA based on low consequence because there will not be any significant effect on the time or magnitude of the expected annual dose for the following reasons. The nuclear radiation from spent nuclear fuel (alpha, beta, and gamma-ray and to a lesser magnitude, neutrons) can increase the quantity of chemically reactive products in water (various oxidants including H_2O_2 and OH, and hydrogen) and in humid air (HNO_3 and HNO_2) as a result of radiolysis. These oxidizing and acidic products can enhance the rate of spent fuel degradation. However, the extent to which they enhance the degradation is not significant, as discussed in the *Supplemental Discussion*. Omission of H_2 gas generation from metal corrosion will not significantly change the calculated expected annual dose. Nonetheless, for evaluation of the solubility of the Pu, Pa, and Cm, where the controlling solid is unknown, radiation damage to crystal lattice was presumed and an amorphous solid conservatively used (CRWMS M&O 2000y), thereby, indirectly providing radiation damage.

DSNF and HLW are being evaluated with regard to their effect or contribution to the technical issues discussed in the *Screening Argument.*

TSPA Disposition: The effects of radiolysis impacting radiation damage in waste and EBS will not be modeled in the TSPA as described in the *Screening Argument.*

Supplemental Discussion: The basis of the screening plan is to argue that the effect of radiolytically produced reactive products do not need to be included in the chemical model to be used for spent-nuclear-fuel dissolution and that the model encompasses any anticipated effects. The screening plan considers both the effects of low linear-energy transfer (LET) (i.e., beta and gamma-rays) and high-LET (i.e., alpha) radiation.

1. For low-LET radiation, the nuclear radiation effects are small (in comparison to high-LET effects) because the radiation fields decrease dramatically over the first 1,000 years after the fuel is out-of-reactor (Sunder et al. 1997, p. 67). Furthermore, the effect of low-LET radiation on fuel degradation will depend on the time at which cladding integrity is lost for a given fuel rod (the low-LET radiation would not be able to penetrate past the cladding material while intact cladding material is present). After cladding material is lost, the radiolysis rate will be dominated by high LET radiation. The effect of low-LET radiation fields after this time will be assessed along with those from high-LET fields, and arguments will be developed to show that the spent-nuclear-fuel dissolution model conservatively encompasses the anticipated effects.

2. The high-LET alpha radiation fields, on the other hand, will persist for tens of thousands of years. Experimental and theoretical evidence of the radiolytic effects due to alpha radiation on spent-nuclear-fuel dissolution will be used to determine whether the model of spent-nuclear-fuel dissolution bounds the effects of radiolytic products from alpha radiation. This will be demonstrated by direct comparison to the existing YMP dissolution rates for varies SNFs.

Radiolysis due to radioactive decay is a mechanism for increasing the transfer of radionuclides from degraded waste forms into the near-vicinity groundwater. This contaminated groundwater may then be released to the unsaturated zone and ultimately transported to the saturated zone to be transported and released to the biosphere. The transport of the radionuclides through natural geologic media is dependent upon many site-specific factors such as: mineralogy, geometry, and microstructure of the rocks, as well as the geometric constraints on the type of groundwater flow (e.g., rock-matrix or fracture flow). Radioactive decay of radionuclides after they have entered the groundwater is not of concern since they will not, to any measurable quantity, increase the release of radionuclides from the waste forms into the groundwater. The decay within the groundwater will only transmute the specific radionuclide inventory already being transported by the groundwater (i.e., due to colloids, dissolution, etc.), and the subsequent decay chains from the transported radionuclides should be modeled within the radionuclide-transport computational codes.

Of the various modes of radioactive decay (i.e., alpha [α] decay, beta [β] decay, gamma [γ] decay, spontaneous fission [SF], isomeric transition [IT], etc.), the most important for fissile materials (Attachment IV) is α-decay. (Radionuclides that decay by the other modes have correspondingly relatively short [i.e., tens to hundreds of years] half-lives; thus, they would decay away prior to the estimated time to corrosion break-through of the fuel cladding [Table IV-1 lists half-lives of radionuclides].) The dominate decay mode for heavy radionuclides is α-decay. All the heavy nuclides above ^{209}Bi are radioactive because they are trying to achieve a more stable nucleus configuration (i.e., atomic masses less than 200 AMUs). Because these heavy radionuclides need to lose significant quantities of mass in order to become more stable, in general they will decay by the mode that results in the largest mass loss. Thus, the decay mode with the highest probability will be α-decay, which has the largest rest mass and associated kinetic energy (contributor to the linear-energy transfer [LET]). While many of the heavy radionuclides emit alpha particles with energies greater than 4.0 MeV (Parrington et al. 1996), there are no gamma releases or beta particles emitted with energies greater than 4.0 MeV and only a very few with energies greater than 1.0 MeV. Thus, α-decay LET values will bound the effects due to beta particles and gamma-rays. Also, as can be identified from Table IV-1, the majority of the long-term radionuclides for YMP are alpha emitters. Other special decay modes such as IT and SF decay have probabilities of occurrence that are orders of magnitude less than that for alpha decay. Of the various radionuclides inventoried for YMP, IT occurs for ^{108m}Ag, ^{242m}Am, ^{93m}Nb, and ^{121m}Sn, none of which are significant in terms of mass contribution. Also, information from Lederer and Shirley 1978 indicates that the SF half-lives are several orders of magnitude longer than that for other decay modes and are thus insignificant.

To be conservative, this FEP screening argument reasonably concludes the following:

1. All disintegrations give off alpha particles, which have the largest values for LET and, thus, also contribute the most to radiolysis effects.

2. Groundwater comes into contact with fuel rods relative short periods of time (after several hundred to several thousands of years). The radioactive inventory will, thus, further decay to radioactive levels less than those used in this FEP.

After applying the above conservative (bounding) conclusions, estimation of corrosion rates of UO_2 will be made by using experimental data obtained for alpha radiolysis and illustrated in Figure 5. This data was obtained for oxidizing conditions from a micro-scale experiment. Conversion of the data to conventional units yielded Figures 4 and 5. Analysis of these data yields an expression that can be used for estimating the corrosion rates. The equation chosen to fit the data was an Arhenius function identified by Equation 28. This equation is superimposed upon the experimental data in Figures 5, 6, and 7.

$$CR = C_1 A^{C_2} e^{-C_3/A} \qquad \text{(Eq. 28)}$$

where

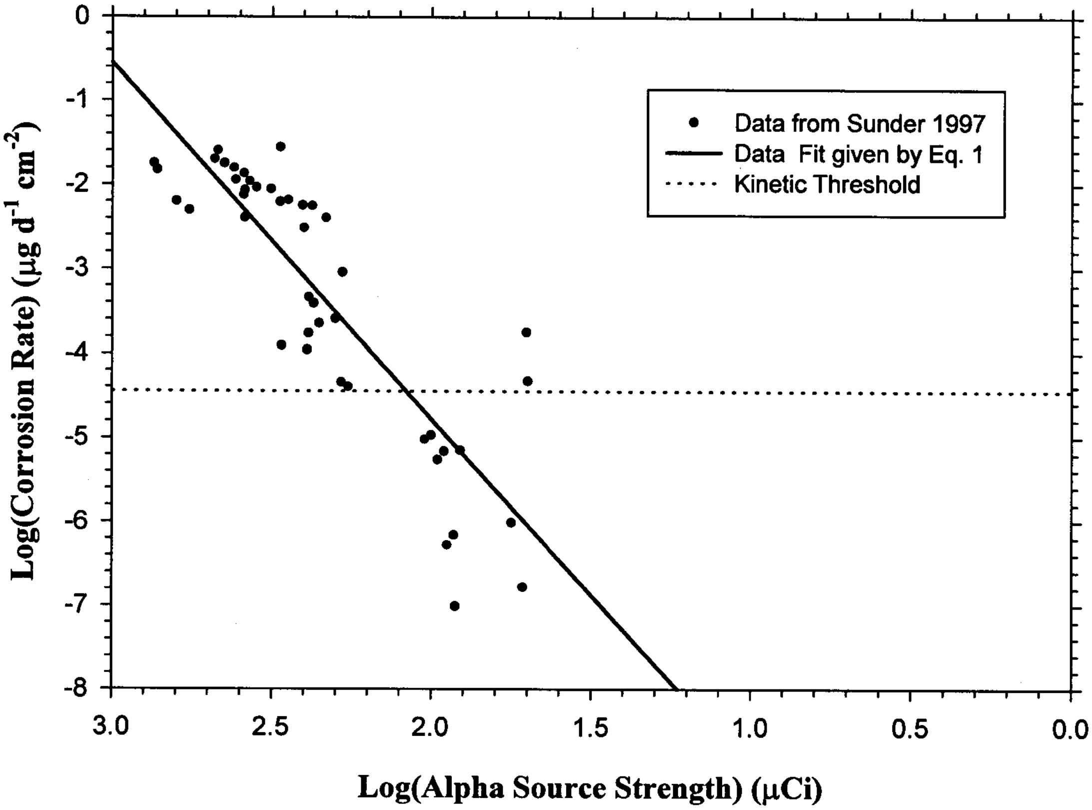
CR = corrosion rate due to α radiolysis (gm/yr-m^2)
A = alpha source strength (Ci/m^2)
$C_1 = 4.0168303\text{x}10^{-1}$
$C_2 = 4.2341650\text{x}10^{0}$
$C_3 = -2.8349524\text{x}10^{-3}$

Tables 11 and 12 were generated to apply Equation 28. Table 11 identifies typical dimensions for worst-case spent-nuclear-fuel types (commercial pressurized-water reactor and boiling-water reactor). Fuel-rod diameter and length data from this table are used to produce Table 12, which estimates the associated alpha source strengths for the major spent nuclear fuel types and the corresponding radiolysis-induced corrosion rate. It should be noted that results are presented only for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). These commercial reactor fuels have fuel burnups and inventories of radionuclides that are much larger than that resulting from the U.S. Department of Energy (DOE) spent nuclear fuels because the DOE fuels are not subjected to long reactor burn times, which are the standard in commercial reactors.

The last column in Table 12 identifies conservative estimates for bounding corrosion rates. Visual comparison of these rates can be made with dissolution rates predicted by Yucca Mountain Project (YMP) models for high-level, metallic, carbide, and ceramic spent nuclear fuel as shown in Figure 8 (CRWMS M&O 1998c). From this comparison, it is evident that corrosion rates due to radiolysis effects (alpha, beta, etc.) are negligible in comparison to the expected non-radiolysis corrosion rates, and this FEP can be considered excluded. Specific findings are listed below:

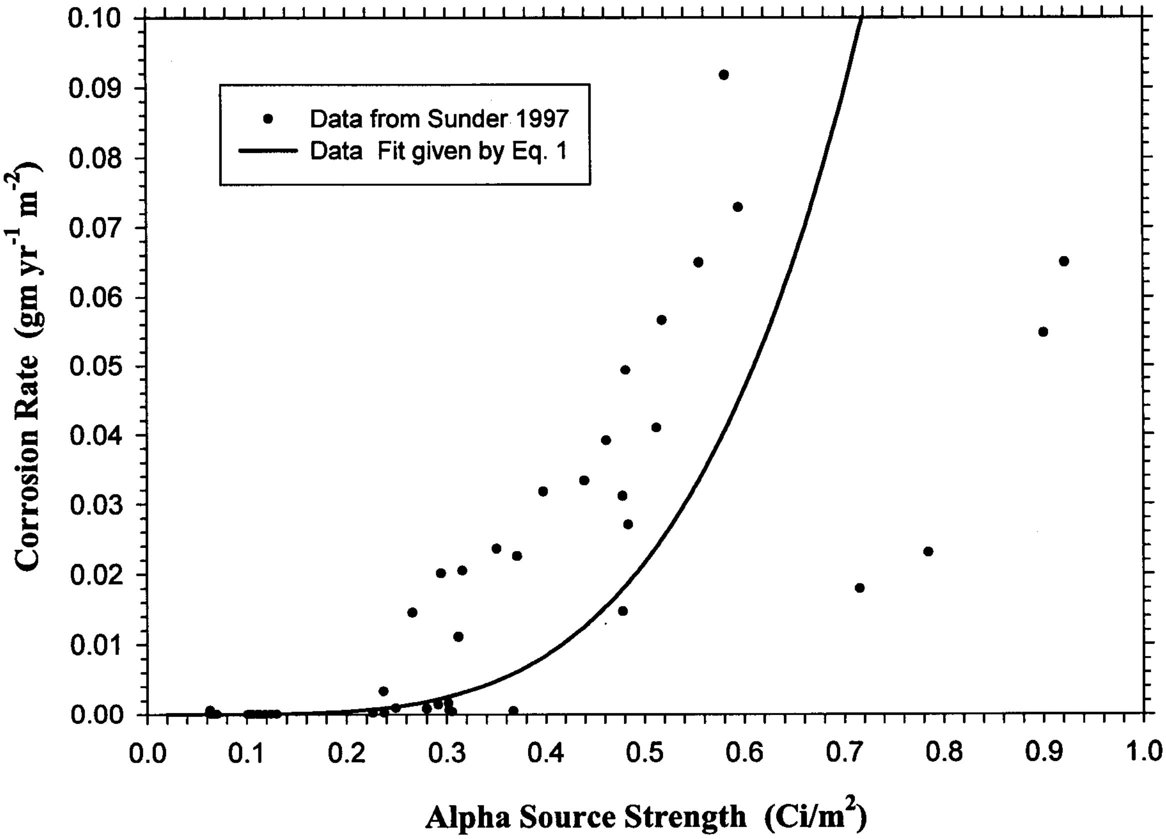
1. The bounding α-radiolysis corrosion rate for uranium oxide fuels is 1.9×10^{-3} gm/yr-m^3. The expected dissolution rate from YMP models for uranium oxide fuels is about 2×10^{0} gm/yr-m^3. Thus, the α-radiolysis corrosion rate is three orders-of-magnitude less than the dissolution rate.

2. While high-level waste (HLW), metallic carbide, and ceramic spent nuclear fuels exhibit lower dissolution rates than that due to metal and oxide spent fuels, these DOE spent fuels/waste have corresponding burnups that are typically one-twentieth or less than that from commercial fuels. This can be identified by inspection of DOE spent fuel/waste radionuclide inventories from CRWMS M&O (1998c). Thus the α-radiolysis-enhanced corrosion rates for HLW, metallic carbide, and ceramic spent nuclear fuels are much lower than their dissolution rates.



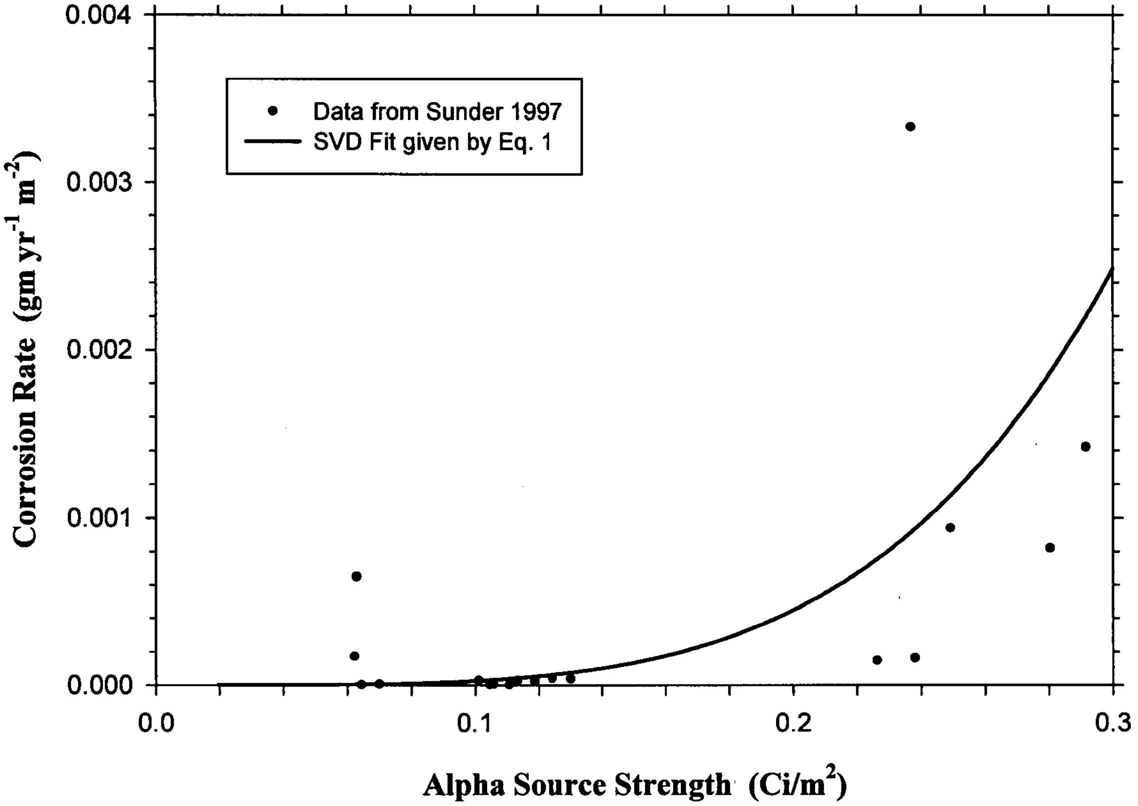
Note: The horizontal dashed line corresponds to the threshold above which kinetically controlled oxidative dissolution (corrosion) of UO_2 fuel occurs (after Sunder et al. 1997).

Figure 5. Corrosion Rates of UO_2 as a Function of Alpha Source Strength in Solutions Undergoing Alpha Radiolysis (0.1 mol L^{-1} $NaClO_4$, pH=9.5, alpha source disks of diameter 1.6 cm)



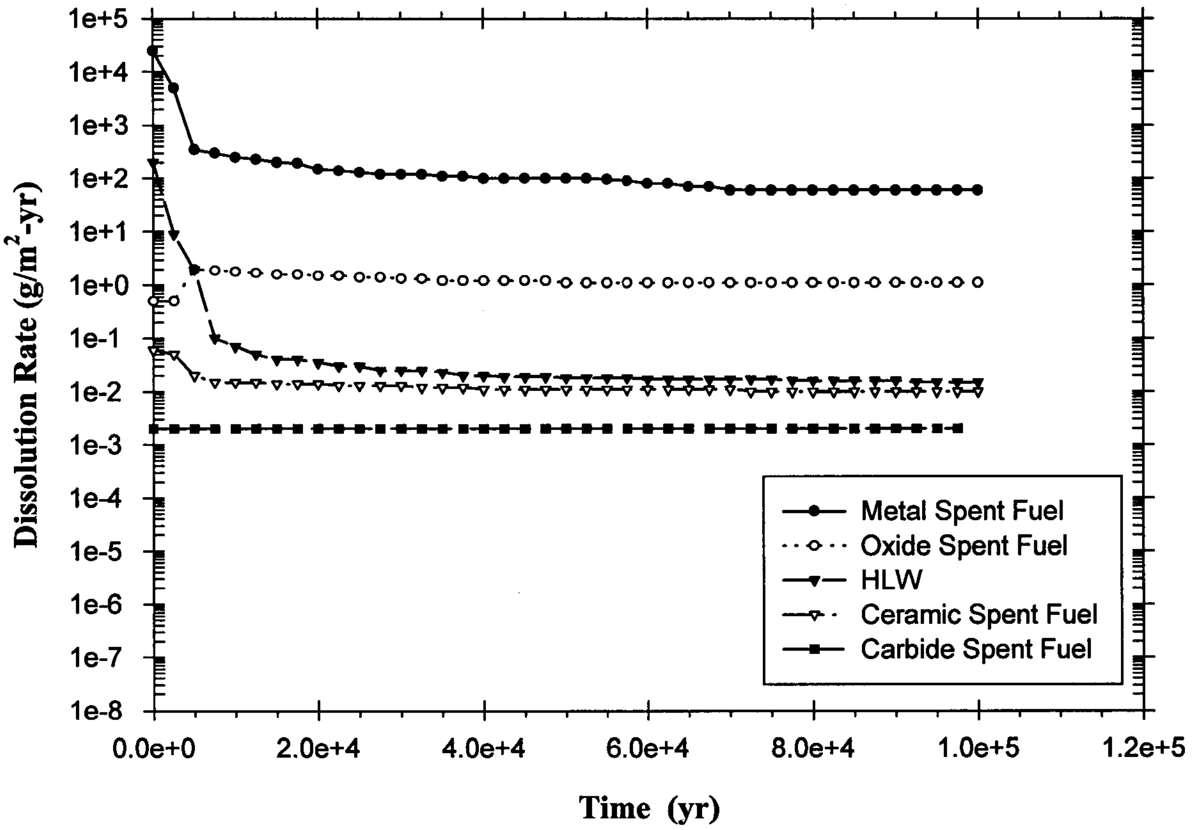
Note: Data taken from Figure 5, converted to conventional units, and plotted in linear-linear coordinates (after modifying data from Sunder et al. 1997).

Figure 6. Data Fit for Corrosion Rates of UO_2 as a Function of Alpha Source Strength in Solutions Undergoing Alpha Radiolysis



NOTE: Original data from Sunder et al. (1997)

Figure 7. Expanded Range (alpha-source strength ≤ 0.3 Ci/m^2) of Corrosion Rates for Uranium Oxide using Alpha Radiolysis Rates from Figure 6



NOTE: Reproduced from CRWMS M&O (1998c, p. A6-29, Figure A6-1)

Figure 8. Dissolution Rates for High-Level Waste, Metallic Carbide, and Ceramic Spent Nuclear Fuel

Table 11. General Dimensions for Spent Nuclear Fuels [(a)]

ID	Fuel Rod Diameter (in) {cm}	Fuel Assembly		
		Array	Lattice Pitch	Length
PWR	~0.5 in {1.27 cm}	14 x 14, 15 x 15	N/A	144 in. {366 cm} [Foster and Wright 1973, p. 393]
BWR	~0.5 in {1.27 cm} [Foster and Wright 1973, p. 395]	7 x7 [Foster and Wright 1973, p. 395]	Pellet Dia + 0.10 in.→ Pellet Dia + 0.20 in	92 in {234 cm} [Foster and Wright 1973, p. 426]
LMFBR	0.27 in {0.685 cm} [Foster and Wright 1973, p. 427]	91 Hex-array [Foster and Wright 1973, p. 426] 271 Hex-array [Foster and Wright 1973, p. 427]	N/A	70 in {178 cm} [Foster and Wright 1973, p. 427]

NOTES:

BWR	Boiling-water reactor
LMFBR	Liquid-metal fast-breeder reactor
N/A	Not available at present time (data not necessary for analysis in this FEP argument).
PWR	Pressurized-water reactor
(a)	References (these references are used to identify dimensional data; this data can also be found in many readily available references: [Foster and Wright 1973]

Table 12. General Properties for Spent Nuclear Fuels

ID	Total Specific Activity (a)	Fraction of α-Curies Near Surface (b)	Surface Area of Fuel Rod (c)	Alpha Source Strength per Surface Area (d)	Estimated Corrosion Rate(e)
	(Ci/rod)	(no units)	(m^2)	(Ci/m^2)	(gm/yr m^2)
PWR (14x14)	~9.5 {@ 525 yr} ~5.6 {@ 1,025 yr} ~2.0 {@ 5,025 yr} ~1.5 {@ 10,025 yr}	0.00236	0.146	0.15 { 525 yr} 0.091 { 1,025 yr} 0.032 { 5,025 yr} 0.024 { 10,025 yr}	1.3E-04 1.6E-05 2.1E-07 6.3E-08
BWR (7x7)	~11. {@ 525 yr} ~6.8 {@ 1,025 yr} ~2.6 {@ 5,025 yr} ~1.9 {@ 10,025 yr}	0.00236	0.0934	0.28 { 525 yr} 0.17 { 1,025 yr} 0.066 { 5,025 yr} 0.048 { 10,025 yr}	1.9E-03 2.3E-04 4.2E-06 1.1E-06
LMFBR(f)	N/A	0.00437	N/A	N/A	N/A

NOTES:

BWR	Boiling-water reactor
LMFBR	Liquid metal fast breeder reactor
N/A	Not Available at present time (data not necessary for analysis in this FEP argument).
PWR	Pressurized-water reactor

(a) Specific activity values calculated from radionuclide inventory data identified in Attachment IV. Specific activity values are equal to the total radioactivity inventory per fuel assembly at a selected time, divided by the number of fuel rods per fuel assemblies (i.e., 196 for PWR and 49 for BWR). These values are approximate in nature and are used only to identify a range of expected values. Their accuracy is not of importance for this FEPs screening argument. The largest total specific activity = BWR(@ 525yr) = 5.49E+02/49≅11 Ci/rod.

(b) The fraction of the alpha particles generated within the spent nuclear fuel rods that reaches the rod surface is estimated by determining the amount of fuel that is within one alpha range of the surface. From Lamarsh 1983, page 91 the range of alpha particles in a material can be found from its range in air by noting its relative stopping power. The range in a material = range(air)/relative stopping power. Where the stopping power = 3100 x density / square root of the atomic weight. For uranium oxide fuels, the density is 10.97 gm/cc (Foster and Wright 1973, p. 367), the atomic weight is approximated 235+2(16)=267 AMUs (Parrington et al. 1996), and corresponding relative stopping power is thus = 3100 * 10.97 / SQRT(267) = 2,081. From Foster and Wright 1973, p. 96 it can be seen that range in air for select alpha energies are: Range (2 MeV)= 1cm, Range(3.5 MeV)=2cm, and Range(5.3 MeV)=3.7cm (very few alphas have energies above 5.3 MeV). The equivalent range of these alpha energies in the uranium oxide fuels (which have a relative stopping power of 2,081) are thus: Range(2 MeV)= 4.8µm, Range(3.5 MeV)= 9.6µm, and Range(5.3 MeV)= 18.µm. Since most alphas are emitted with magnitudes on the order of ~4 MeV or less the resulting UO_2 stopping range is estimated to have a range of about 11 → 15 µm. The fraction of α-curies within one alpha range of the rod surface is the ratio of the rod volume made up of the hollow cylinder (within the outer diameter and a thickness of 15 µm) to the total rod volume.

(c) Surface area calculated using rod diameter and length values from Table 11.

(d) Alpha source strength is equal to the product of the data from Columns 2 and 3 divided by the data value in Column 4.

(e) The estimated corrosion rate is obtained by applying Equation 28on the data from Column 5. (Estimated corrosion rate values can be visualized from Figure 5.) Comparison of these data values with expected dissolution rates for high-level waste, metallic carbide, and ceramic spent nuclear fuel (see Figure 6) indicates that the corrosion rate due to radiolysis (even if the radioactivity is reasonably determined to be 100% alpha emitting) is negligible. These estimated values are approximate in nature and are used only to identify a range of expected values. Their accuracy is not of importance for this FEPs screening argument. DSNFs burnup is less than 1/20 that of commercial fuels

(f) LMFBR fuels are not analyzed, their burnup along with other DSNF is much less than that of commercial fuels.

6.2.52 Use of J-13 Well Water as a Surrogate for Water Flowing into the EBS and Waste—YMP No. 2.2.08.12.00

YMP Primary FEP Description: J-13 water chemistry may be used in analysis or modeling that requires initial water chemistry.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: No Secondary FEPs associated with this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.2.08.01.00	Groundwater chemistry / composition in UZ and SZ Groundwater Chemistry FEPs
2.2.08.02.00	Radionuclide transport occurs in a carrier plume in geosphere SZ—Radionuclide Transport in a Carrier Plume
2.1.09.01.00	Properties of the potential carrier plume in the waste and EBS
2.1.09.04.00	Radionuclide solubility, solubility limits, and speciation in the waste form and EBS

IRSR Issues: TSPAI3, ENFE3, CLST1, CLST3, CLST4

Screening Argument: J-13 well water will have equilibrated with minerals in Yucca Mountain and is, therefore, expected to be representative of groundwater passing through the repository. Hence, J-13 well water is used to set initial water chemistry inside the waste package.

There are no secondary FEPs with respect to chemical composition of J-13 well water being used as the initial water chemistry in the waste form.

TSPA Disposition: J-13 well water is used to set initial water chemistry inside the waste package in the In-Package Chemistry Component (CRWMS M&O 2000i).

Supplemental Discussion: Water chemistries vastly different from J-13 (and compositions measured at the site) are required, in order to have a material impact on EBS and waste-form interactions. To begin with, the behavior of the EBS is reasonably determined to be independent of the chemistry of fluids that enter the EBS but not the waste form. Once inside the waste form, fluids will be quite different in composition (pH, ionic strength, etc.) from the original state because of dissolution of waste-form components. The only way for waste-form degradation to be materially different from that predicted using J-13 well water as input is if the input fluids cause accelerated degradation of cladding. Water chemistry vastly different than J-13 is not seen in the vicinity of Yucca Mountain (McKinley et al. 1991).

There is the possibility that fluids slightly different compositionally than J-13 will flow into the EBS and waste form after the peak thermal period. Indeed such waters can be collected from selected horizons at the site. The composition of waters at the site reflects a balance between addition of atmospheric gases (primarily O_2 and CO_2) and dissolved mineral components (e.g., Na, Si, etc.) to an original rainfall composition and possibly some degree of evaporative concentration. Because atmospheric and mineral additions are similar throughout the site, analyzed fluid compositions tend to give a reasonably consistent picture of ambient groundwater chemistries. Namely, fluids are dilute, mildly alkaline solutions close to equilibrium with atmospheric O_2 and CO_2. Fluids that enter breached waste forms are expected to rapidly react with waste-form components, causing the chemistry of the interacting fluids to change drastically in composition from the initial state. Whereas the initial fluids are likely to be dilute, interaction with waste packages causes the resulting fluids to be more concentrated and materially different from the original. In other words, effluent chemistry will largely be more dependent on waste-form interaction than its starting composition.

6.2.53 Radioactive Decay and Ingrowth—YMP No. 3.1.01.01.00

YMP Primary FEP Description: Radioactivity is the spontaneous disintegration of an unstable atomic nucleus that results in the emission of subatomic particles. Radioactive isotopes are known as radionuclides. Ingrowth occurs when a parent radionuclide decays to a daughter nuclide so that the population of the daughter nuclide increases. Over a 10,000-year performance period, these processes will produce daughter products that need to be considered in order to adequately evaluate the release and transport of radionuclides to the accessible environment.

Screening Decision and Regulatory Basis: *Included.*

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.01.01.00	Waste Inventory

IRSR Issues: TSPAI3, CLST3, CLST4

Screening Argument: Radioactivity is the spontaneous disintegration of an unstable atomic nucleus that results in the emission of subatomic particles. Radioactive isotopes are known as radionuclides. Ingrowth occurs when a parent radionuclide decays to a daughter nuclide so that the population of the daughter nuclide increases. Because the licensing criteria include a 10,000-year performance period, these processes must be accounted for in order to adequately evaluate the release and transport or radionuclides to the accessible environment and through the biosphere to humans.

TSPA Disposition: Generically included in computer-modeling capabilities and as described in AMRs *Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA* (CRWMS M&O 2000q) and *Inventory Abstraction* (CRWMS M&O 2000j). Radioactive decay and ingrowth (i.e., further production of a radioisotope as its parent decays) is either incorporated in the computer codes used for the TSPA-SR total-system model, or the waste inventory of a daughter is conservatively increased when generating the waste inventory in cases where decay and ingrowth are not explicitly modeled.

Supplemental Discussion: Most radionuclides in the inventory have been screened from consideration for performance by CRWMS M&O (2000j), "Inventory Abstraction." For those radionuclides which were not screened out, radioactive decay and ingrowth is either already fully incorporated into the computer codes used for performance-assessment modeling or daughter-product waste inventories are conservatively increased in cases where decay and ingrowth are not explicitly modeled. Thus, the actual implementation of this process as performed by analysts may not strictly reflect the radionuclide decay-chain structure. This approach (CRWMS M&O 2000c) is intended to minimize computational requirements. For more detail about the actual implementation of decay chain phenomena, refer to CRWMS M&O (2000j; 2000k).

6.2.54 Isotopic Dilution—YMP No. 3.2.07.01.00

YMP Primary FEP Description: Mixing or dilution of the radioactive species from the waste with species of the same element from other sources (i.e., stable and/or naturally occurring isotopes of the same element) could lead to a reduction of the radiological consequences.

Screening Decision and Regulatory Basis: *Included* (mixing or dilution of specific radionuclides due to isotopes of the same element within waste package).

Excluded based on low consequence (mixing or dilution of specific radionuclides due to isotopes of the same element within waste package).

Treatment of Secondary FEPs: See Attachment V for the Secondary FEPs related to this Primary FEP.

Related Primary FEPs:

YMP No.	FEP Name
2.1.02.02.00	CSNF alteration, dissolution, and radionuclide release
3.1.01.01.00	Radioactive Decay and Ingrowth

Radionuclide solubility, solubility limits

IRSR Issues: TSPAI3, ENF3, CLST3, CLST4

Screening Argument: Mixing or dilution of specific radioisotopes of the same element within the waste package is accomplished by reasonably concluding that the dissolved isotopic mass at a given elemental concentration is proportional to the isotopic inventory at that time.

Inclusion of isotopic dilution during transport away from the waste package in the TSPA-SR would reduce calculated doses by reducing concentrations of critical nuclides. Therefore, no credit is allowed for the beneficial effects of isotopic dilution outside the waste package. Omission of isotopic dilution during transport away from the waste package as a system-wide feature from the performance assessment models will not significantly change the interpretation of expected dose and can be excluded from consideration in the TSPA based on low consequence.

TSPA Disposition: Within the waste-form-dissolution model, if the solubility-limited value for a given radionuclide is lower than its concentration derived from waste-form dissolution, then the aqueous concentration is set to the solubility-limited value, and the difference in mass is calculated to precipitate out of solution. These solubility-limited values place constraints on the aqueous concentration of the particular radionuclide element considered with each isotope of that element present in proportion to its isotopic abundance (CRWMS M&O 1998a, Section 6.4.1.2).

No credit is taken for isotopic dilution outside of the waste package.

Supplemental Discussion: Isotopic dilution, which refers to the mixing of radionuclides derived from the waste with less-radioactive or stable isotopes of the same element, either could have no effect on the concentrations of radionuclides in groundwater, or, if the element in question has reached its elemental solubility limit, could result in a reduction of the aqueous concentration of the specific radionuclide. Because the expected annual dose received by the critical group is directly dependent on the aqueous concentrations of radionuclides reaching the location of the critical group, this process (excluding ingrowth) has no potential to result in increases in the expected annual dose.

For the discussion in this FEP, isotopic dilution could result from:

- Mixing of radionuclides of concern either with other isotopes contained within the waste (e.g., the mixing of ^{234}U, which has a potential to contribute to doses to the critical group, with the less-radioactive ^{238}U),

- Or with isotopes that occur within the natural barrier system. Because isotopes of the same element will display the same chemical behavior during transport, and will be partitioned in the aqueous phase under equilibrium conditions proportionally to their total abundance in the system.

At that time, isotopic dilution has a potential to lower concentrations of radionuclides of concern for any element that reaches its elemental solubility limit. Isotopic dilution (excluding ingrowth) cannot raise radionuclide concentrations, and the reasonable conclusion used throughout the TSPA that radionuclide concentrations are not affected by the presence of other isotopes of the same element, provides a conservative bound to the effects of this process.

Strictly speaking, isotopic dilution means diluting emplaced radionuclides (in the waste inventory) with naturally occurring isotopes in the groundwater or minerals. Examples included diluting ^{129}I from the waste with naturally occurring ^{127}I in the saturated zone (SZ) groundwater, or diluting ^{14}C in the waste with ^{12}C in naturally occurring calcite. For ^{238}U the naturally occurring concentration in the groundwater at Yucca Mountain ranges from .01 to 1.9 ppb (mg/m^3) (CRWMS M&O 1998b, Section 6.2.6.7). The mean solubility limit used in the WF/EBS modeling is about 7 ppm (g/m^3) (CRWMS M&O 1998a, Table 6-32); therefore, the natural ^{238}U would have very little effect because of its extremely low concentration.

Ingrowth generated by the decay of a parent radionuclide has the potential to change the isotopic abundance at a given location, and time, and therefore, the relative isotopic concentrations. Ingrowth can increase the amount of critical isotopes both in the waste form (source term) and during transport. Particles of a parent nuclide could be released from a waste package and transported through the EBS before they decay into a daughter product. Proper accounting for all isotopic forms of an element is critical to the analyses. In the TSPA, in growth and radionuclide decay chains are not explicitly modeled because of computational limitation and efficiency considerations. Their effects are included by appropriately adjusting the initial quantities of the radionuclides that are being tracked through the system. See YMP No. 2.1.02.02.00, Waste Inventory and YMP No. 3.1.01.01.00, Radioactive Decay and Ingrowth for details.

7. CONCLUSIONS

7.1 MISCELLANEOUS WASTE-FORM CONCLUSIONS

The following conclusions may be drawn:

- This AMR contributes to Yucca Mountain scenario development by screening 54 Miscellaneous WF FEPs.
- For 54 of these FEPs, identified as "Miscellaneous Waste-Form FEPs," this report develops screening arguments and TSPA-disposition statements. This AMR provides both information for the YMP FEP Database and guidance to TSPA analyses, which is appropriate for both site-requirement and license-application documents.
- By providing references to appropriate other AMRs, this report provides a valuable link between WF issues and the research directed at their resolution.
- Finally, this AMR correlates DOE's WF FEPs with three of NRC's key technical issues (CLST, ENFE, and TSPAI).

Tables 13 through 20 provide a summary of the FEPs screening decisions for Miscellaneous Waste Form FEPs, Cladding FEPs, and Colloid FEPs and the bases for *Excluded* decisions. Gray shading indicates Miscellaneous WF FEPs in this AMR. For the purposes of this AMR, secondary FEPs identified as important enough to be considered primary are also included, as well as new technical subjects proposed as "new" FEPs.

7.2 NRC ISSUE RESOLUTION

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

7.2.1 Staff-Level Issue Resolution

To structure staff-level interactions, the NRC has focused on topics most critical to post-closure performance of the potential geologic repository (Sagar 1997). At present, NRC staff developed 10 Key Technical Issues (KTIs), nine of which relate to post-closure performance assessment (Table 21).

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

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

7.2.2 Relation of Waste Form FEPs to NRC KTIs

YMP's FEP and NRC's KTI processes arise from similar objectives. One evolves from the features, events, and processes relevant to performance, and the other evolves from the issues most critical to performance. Thus, the strong correlation that exists between FEPs and KTIs is to be expected. Although waste-form FEPs relate to a variety of KTIs, they relate dominantly to the first, second, and third issues listed in Table 21. Tables 15, 16, and 17 associate waste-form FEPs with TSPAI (NRC 1998b), CLST (NRC 1998a), and ENFE (NRC 1997) sub-structures. The associations demonstrate a partial, if not complete, satisfaction of these issues.

Table 13. FEPs Related to the Radioisotope Inventory Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.01.01.00	Waste inventory	Included Excluded	WF	ANL-WIS-MD-000009 ANL-WIS-MD-000006	Included: Average and bounding inventories for 27 radioisotopes for 3 waste allocation categories (CSNF, DSNF, and HLW) have been developed; 24 isotopes from screening arguments based on human dose, 2 isotopes mandated by the Groundwater Protection Requirement of the proposed 40 CFR 197 and 1 (^{235}U) to properly perform radioactive decay; different subsets of the isotopes are used for the various scenarios: volcanic release (16), nominal groundwater release (21), and human intrusion (23). This FEP is the topic of AMR, *Inventory Abstraction* (CRWMS M&O 2000j). Excluded based on low consequence, the effects of hazardous waste

Table 13. FEPs Related to the Radioisotope Inventory Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.01.03.00	Heterogeneity of waste forms	Included	WF	ANL-WIS-MD-000009	Heterogeneity of waste forms is included by handling CSNF, DSNF, and HLW separately. Heterogeneity is included within these three categories only to the extent that it is used to determine the average or bounding inventory.
2.1.12.07.00	Radioactive gases in waste and EBS	Excluded	WF, EBS	ANL-WIS-MD-000009	Excluded based on low consequence. The potential human dose from inhalation of radioactive gases was low relative to other doses and, thus they were screened out except for dissolved C-14. Excluded based on low consequence, impact of radioactive gas partial pressures on the waste package failure rates., because gas production rates have been shown by analyes to be insufficient. Other influences of gases (radioactive or otherwise) are also excluded as discussed in other FEPs: Biological activity in the waste is screen out (colloid FEP 2.1.10.01.00) and so generation of microbial gases such as CO_2, CH_4, H_2S are also screened out (in-package chemistry FEP 2.1.12.04.00). The chemical influence of H_2 is conservatively neglected (in-package chemistry FEP 2.1.12.03.00). Generally, the physical aspects of gas generation such as displacement of oxygen and pressurization of the disposal drifts by He and H_2 (in-package chemistry FEPs 2.1.12.02.00 and 2.1.12.03.00, respectively) have been excluded since gas would readily dissipate from the repository over geologic time (i.e., because fractures and fault zones are unsaturated, the relative permeability to gas is large). (Note: the relative permeability of the fractures and fault zones to water is lower than for air and is included in the unsaturated flow model—see FEPs related to unsaturated flow model.) See cladding FEP 2.1.02.20.00 where the pressurization of CSNF cladding is included.

Table 13. FEPs Related to the Radioisotope Inventory Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
3.1.01.01.00	Radioactive decay and ingrowth	Included	WF, UZ, SZ	ANL-WIS-MD-000009	Included: Radioactive decay and ingrowth (i.e., further production of a radioisotope as its parent decays) is either incorporated in the computer codes used for the TSPA-SR total system model, or the inventory of a daughter is conservatively increased when generating the inventory in cases where decay and ingrowth is not explicitly modeled.
3.2.07.01.00	Isotopic dilution	Included/ Excluded	WF, SZ	ANL-WIS-MD-000009	Included: Mixing or dilution of specific radionuclides of the same element within the WP is accomplished by assuming that the dissolved isotopic mass at a given elemental concentration is proportional to the isotopic inventory at that time. Excluded outside waste package based on low consequence. No credit is taken for isotopic dilution during transport away from the WP due to stable and/or naturally occurring isotopes of the same element; dilution only decreases adverse radiological effects.

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.01.02.00	Co-disposal/ collocation of waste	Included/ Excluded	WF	ANL-WIS-MD-000009	Included: The chemical interaction between DSNF and HLW with in a co-disposal package is explicitly included in the In-Package Chemistry Component. Co-disposal of DSNF and HLW is also included in the TSPA-SR by separately modeling waste form degradation of and radionuclide release from both categories within a co-disposal waste package (see DSNF FEP 2.1.02.01.00 and HLW FEP 2.1.02.03.00). Excluded, low consequence: chemical interactions between waste packages; the effects DSNF cladding (except for Naval SNF); the effects of DSNF and glass-pour canisters as barriers to DSNF degradation, to HLW-glass dissolution, or to radionuclide release; DSNF geometry and surface area dependence; and preferential condensation. Excluded, low probability: the possible effects of collocation with low- and intermediate-level nuclear waste, of toxic chemical wastes, and wastes disposed by future human activity in a nearby facility. Some of these are also addressed in other FEPs: spatial heterogeneity caused by cooler initial temperatures of the packages and thus preferential condensation when the repository cools (in-package chemistry FEP 2.1.01.04.00); DSNF cladding and time dependence of surface area for degradation and radioisotope release (DSNF FEP 2.1.02.25.00); DSNF standard or high integrity canister (DSNF FEP 2.1.02.01.00); HLW standard canister (HLW FEP 2.1.02.03.00).

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.01.04.00	Spatial heterogeneity of emplaced waste	Excluded/ Included	WF	ANL-WIS-MD-000009	Excluded based on low consequence: The time scale of convective transport and diffusion of aqueous species in the fully flooded waste package is rapid in relation to alteration of the contents (waste for, cladding, and support material); consequently, the contents are assumed to be uniformly mixed and spatial heterogeneity in the waste to evaluate in-package chemistry is excluded. Spatial heterogeneity is of the two types of waste packages modeled is also excluded based on low consequence. WPs of CSNF and co-disposed DSNF and HLW are to be closely packed together so that variations in temperature (which can influence aspects of waste form degradation and in-package chemistry) of individual WPs will not vary significantly. Included: Spatial variations of temperature within a waste package are evaluated to determine perforation through creep rupture [cladding FEP 2.1.02.19.00] and stress corrosion cracking [cladding FEP 2.1.02.21.00]
2.1.02.09.00	Void space (in waste package)	Included/ Excluded	WF	ANL-WIS-MD-000009	Included: For determining the radioisotope concentration, the Dissolved Radioisotope Concentration Component uses a time varying void volume of the altered waste form (see Solubility FEP 2.1.09.12.00). In addition, in the process model for evaluating the In-Package Chemistry Component, the mixing cell volume is equal to the fixed void volume in the waste package; hence, void space is indirectly included in the In-Package Chemistry Component. Excluded based on low consequence. The potentially time varying void space available and its influence on determining the in-package chemistry was excluded based on low consequence. Rather, the uncertainty in the in-package chemistry that might be caused by different void volumes was surmised to be swamped by the use of various water inflow rates into the waste package that were used to define the various simulations and, thereby, the uncertainty used in the In-Package Chemistry Component.

Table 14. FEPs Related to the In-Package Chemistry Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.10.00	Cellulosic degradation	Excluded	WF	ANL-WIS-MD-000009	Excluded based on low probability (not credible). According to preliminary waste acceptance criteria, no cellulosic materials will be included as part of the waste in the potential Yucca Mountain repository.
2.1.03.06.00	Internal corrosion of waste container	Included/ Excluded	WF, WP	ANL-WIS-MD-000009	Included: After breach of the WP, the corrosion of the inner structural stainless steel is important to determining the in-package chemistry (provided a high sulfur steel is used) and is, thus, included in the process model and as part of the uncertainty of the In-Package Chemistry Component (see also in-package chemistry FEP 2.1.09.02.00). Excluded based on low consequence: Prior to WP breach, internal corrosion will not be significant. The current design specifies placement of inert gases in the package and limited water allowed in the CSNF (see cladding FEP 2.1.02.11.00) and HLW glass packages. Some canisters containing DSNF such as those containing N-Reactor fuel, may have more water, but this water would be scavenged by the waste form itself.
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and EBS	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	Included through the use of a series of linked one-dimensional flow paths and mixing cells through the EBS, drip shield, waste package, and into the invert. Excluded based on low consequence No credit is taken for transport pathways inside container; rather, container assumed to be a mixing cell for In-Package Chemistry Component. Inclusion of the pathways would delay release of radionuclides and thus they are conservatively neglected.
2.1.08.08.00	Induced hydrological changes in the waste and EBS	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	Included: Changes in flow area into the WP as it degrades are included. Changes in the exposed fuel area as CSNF cladding fails are also included (see cladding unzipping FEP 2.1.02.23.00). Excluded based on low consequence: Changes to hydrologic properties of the waste form and invert are excluded since the slight changes in hydrologic properties are small in relation to the overall uncertainty of the in-package chemistry predictions.

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.08.10.00	Desaturation/ dewatering of the repository	Included	WF, NFE	ANL-WIS-MD-000009	Included through unsaturated flow modeling in the TSPA thermal hydrologic models which, in turn, affects the volume of water entering the disposal WPs.
2.1.09.01.00	Properties of the potential carrier plume in the waste and EBS	Included/ Excluded	WF	ANL-WIS-MD-000009	Included: Water entering the package is assumed to have chemical properties of J-13 well water in the process models of the in-package chemistry. Water chemistry was then altered based on waste type, cladding failure, seepage flux, and waste degradation rate. Furthermore, corrosion of steel is directly used to establish an uncertainty range for the water chemistry. Excluded based on low consequence: the changing properties of incoming water, as evaluated by EBS. Although the changing properties of the incoming water as evaluated by EBS are not coupled to these process calculations, slight changes in the incoming water over time are swamped by the dramatic changes that are predicted in the in-package chemistry because of the waste and internal parts of the waste package. That is, the range of uncertainty used for the In-Package Chemistry Component is not changed by the minor changes of the initial chemistry of the water entering the package. This fact is especially true now that a concrete tunnel liner is not used in the current design.

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.02.00	Interaction with corrosion products	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	Included: The chemical effects of the corrosion of steel (e.g., release of sulfur) on the in-package water chemistry have been included (see also in-package chemistry FEP 2.1.03.06.00). Several other interactions are also included and discussed in other FEPs: the presence of corrosion product is in included in determining the availability of water for radioisotope dissolution (solubility FEP 2.1.09.12.00); the volume increase of the corrosion products and the unzipping of the cladding is included (cladding FEP 2.1.09.03.00); and the sorption on mobile colloids is included (colloid FEP 2.1.09.05.00). Coupling of related processes was also included (see in-package chemistry coupled processes FEP 2.1.11.04.00) Excluded based on low consequence: No credit is taken for the potential beneficial effects of corrosion products, which Included sorption on immobile corrosion products (colloid FEP 2.1.09.05.00), and the decreased ability for advective and diffusive transport of radioisotopes.
2.1.09.06.00	Reduction-oxidation potential in waste and EBS	Included	WF, EBS	ANL-WIS-MD-000009	Included: In developing the regression equation of pH for the In-Package Chemistry Component, which is based on the EQ3/6 equilibrium process model, the fluid is assumed equilibrated with the atmosphere to ensure maximum plausible O2 and CO2 conditions. Furthermore, to match these assumptions, the In-Package Chemistry Component sets the O2 and CO2 conditions at atmospheric conditions. These conditions are then used by the CSNF Matrix Degradation Component (see CSNF FEP 2.1.02.02.00) and Dissolved Radioisotope Concentration Component (see Solubility FEP 2.1.09.04.00).

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.07.00	Reaction kinetics in waste and EBS	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	Included reaction kinetics through quasi-equilibrium analysis. The equilibrium model evaluates the pH over time as a function of several rates of degradation of the cladding, SNF matrix, HLW glass, and internal components of the disposal package (stainless steel and aluminum); the resulting variance in the pH is then used to set bounds on the uncertainty of the in-package water chemistry. Excluded based on low consequence the impacts of transient disequilibrium states. The in-package chemistry process model, EQ3/6, assumes instantaneous equilibrium between changes in amounts of corrosion products available..
2.1.09.08.00	Chemical gradients / enhanced diffusion in waste and EBS	Excluded	WF, EBS	ANL-WIS-MD-000009	Excluded based on low consequence. The WP is assumed to be a mixing cell without chemical gradients in the In-Package Chemistry Component. Provided the WP is fully saturated, the rate of the flow of water into and out of the package is slow enough that no long-term gradient would be present and equilibrium would occur. Furthermore, rather than include diffusive release of radionuclides out of the CSNF perforated cladding, DSNF perforated cladding, or cracked HLW glass, a more conservative conceptual model is developed that subsumes enhanced release from diffusion caused by chemical gradients cladding unzipping for CSNF (cladding unzipping FEP 2.1.02.23.00), 100% failed cladding for DSNF cladding FEP 2.1.02.25.00), and very high reactive surface area for HLW glass (HLW surface area FEP 2.1.02.05.00).
2.1.09.09.00	Electrochemical effects (electrophoresis, galvanic coupling) in waste and EBS	Excluded	WF	ANL-WIS-MD-000009 ANL-EBS-PA-000002	Excluded based on low consequence. The influence of fluid flow through a failed container on the in-package chemistry is much greater than any effect on the degradation of the SNF or HLW matrix that can be created by electrophoresis or electro-osmosis.

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.11.00	Waste-rock contact	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	Included: The indirect influence of waste-rock contact on in-package chemistry is included through the use of the J-13 well water (in-package chemistry FEP 2.2.08.12.00). The perforation of cladding from shaking in a severe earthquake is included as a disruptive event (cladding FEP 2.1.02.14.00). Excluded based on low consequence: Because of the drip shield and the long-lived waste package, rock is not expected to come directly in contact with the waste in the first 10,000 yr. Furthermore, even if some contact were to occur, the overall result would be little or no involvement of the rock minerals in chemical reaction due to their dissolution kinetics. The perforation of CSNF cladding from rock fall is also excluded in the first 10,000 yr (cladding FEP 2.1.07.01.00)

Table 14. FEPs Related to the In-Package Chemistry Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)					
FEP Number	**FEP Title**	**Include/ Exclude**	**Process Model Report (PMR)**	**AMR Document Identifier (DI)**	**Reason for Include/Exclude Decision**
2.1.11.04.00	Temperature effects / coupled processes in waste and EBS	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	Included: Many of the processes of the Waste Form Degradation model are coupled. CSNF matrix degradation (CSNF FEP 2.1.02.02.00), HLW degradation (HLW FEP 2.1.11.01.00), and Dissolved Radioisotope Concentration (solubility FEP 2.1.09.04.00) is coupled to the chemical conditions calculated by the In-Package Chemistry Component (this FEP). Cladding unzipping (Cladding FEP 2.1.02.23.00) is coupled to CSNF matrix degradation (see also in-package chemistry FEP 2.1.09.02.00). In turn, weak feedback to the In-Package Chemistry Component occurs from HLW degradation and cladding degradation. Temperature is also coupled to CSNF matrix degradation (CSNF FEP 2.1.11.05.00), CSNF cladding degradation (cladding FEP 2.1.11.07.00), and HLW degradation (HLW FEP 2.1.11.01.00), and solubility of uranium (solubility FEP 2.1.09.04.00). Excluded based on low consequence: because the primary effects of temperature on waste form degradation are included directly, the secondary effects of temperature on in-package chemistry such as pH are of low additional consequence (also see in-package chemistry FEP 2.1.11.08.00)..

Table 14. FEPs Related to the In-Package Chemistry Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and EBS	Included/ Excluded	WF, NFE, EBS	ANL-WIS-MD-000009	Included: temperature effects are included directly in various components (e.g., cladding degradation rate [cladding FEP 2.1.11.07.00], CSNF matrix degradation rate [CSNF FEP 2.1.11.05.00], HLW degradation rate [HLW FEP 2.1.11.01.00], and solubility of uranium [solubility FEP 2.1.09.04.00]) Excluded based on low consequence: because the primary effects of temperature on waste form degradation are included directly, the secondary effects of temperature on in-package chemistry such as pH are of low additional consequence. Furthermore, the use of a cooler repository design, drip shield, and long-lived WP implies WP breach occurs when temperatures are near ambient temperature where thermodynamic data has been collected. Thus, the collection of thermodynamic data as a function of temperature is not necessary. Thermal effects on microbiological activity excluded since microbiological FEP screened out as a process that effects waste form degradation (colloid FEP 2.1.10.01.00)
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS	Included/ Excluded	WF, EBS, NFE	ANL-WIS-MD-000009	Included: thermal effects on fluid flow are included indirectly to the extent that they influence the seepage flux into the drift. Excluded based on low consequence. Thermal effects on flow within the waste form are excluded because temperature within and outside the package will be fairly uniform by the time WP fail and thus thermal gradients will not significantly influence flow. Furthermore, temperature will be too low to promote two phase flow (see in-package chemistry FEPs 2.1.11.10.00 and 2.1.09.08.00 on thermal and chemical diffusion processes, which are also excluded).
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and EBS	Excluded	WF, EBS, NFE	ANL-WIS-MD-000009	Excluded based on low consequence. Bounding analyses indicate diffusion due to temperature gradients within the waste form is insignificant (see cladding gap release FEP 2.1.02.07.00).

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.12.01.00	Gas generation	Excluded	WF, EBS, UZ	ANL-WIS-MD-000009	Excluded based on low consequence: Prior to breach of a WP, gas generation WP will be insignificant (see in-package chemistry FEP 2.1.03.06.00 on internal corrosion of waste container). After breach of a WP, the chemical influence of gas generation of H_2 has been conservatively excluded since it moderates and suppresses the low pH values encountered from sulfur release from steel (see following in-package FEP 2.1.12.03.00). Generation of radioactive gases in waste are also excluded since the inhalation dose is low relative to ingestion doses (inventory FEP 2.1.12.07.00). In addition, most physical aspects of gas generation such as displacement of oxygen and pressurization of the disposal drifts by He and H_2 (in-package chemistry FEPs 2.1.12.02.00 and 2.1.12.03.00, respectively) have been excluded since gas would readily dissipate from the repository over geologic time (i.e., because fractures and fault zones are unsaturated, the relative permeability to gas is large). (Note: the relative permeability of the fractures and fault zones to water is lower than air and included in the unsaturated flow model—see FEPs related to unsaturated flow model.) Only the physical aspect of pressurization of CSNF cladding is indirectly included (cladding FEP 2.1.02.20.00).
2.1.12.02.00	Gas generation (He) from fuel decay	Included/ Excluded	WF, EBS	ANL-WIS-MD-000009	The impact of He production and pressure build up inside DSNF cladding is included in evaluating cladding failure due to creep and stress corrosion cracks. Excluded based on low consequence. Radioactive alpha decay of actinides such as Pu, U, Th, etc., could result in the creation of additional He gas within the waste form; however, studies done for TSPA-VA shows increases in internal pressures from He to be small relative to fission gas pressures. Internal gas pressure inside of CSNF cladding is assumed to be an energy source of cladding creep nature and stress corrosion cracking and included in cladding FEP 2.1.01.20.00.

Table 14. FEPs Related to the In-Package Chemistry Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.12.03.00	Gas generation (H_2) from metal corrosion	Excluded	WF, EBS, WP	ANL-WIS-MD-000009	Excluded based on low consequence. The hydrogen that could be produced from metal corrosion will be low since the repository is likely to be primarily oxic since it is well connected to the surface. Also, because the repository is connected to the surface, any gas produced is expected to escape. The influence of H_2 gas on in-package chemistry is negligible. It is conservative to exclude H_2 evolution from steel corrosion as this limits pH suppression. See cladding FEP 2.1.02.22.00 for effect of H_2 on cladding degradation.
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation	Excluded/ Included	WF, EBS	ANL-WIS-MD-000009	Excluded based on low consequence., because YMP is considered to be and open system. Also, biological activity inside the WP is excluded (colloid FEP 2.1.10.01.00) since no organic material will be allowed inside the WP based on preliminary waste acceptance criteria. Therefore, the effects of biological gases on the in-package chemistry are also expected to be negligible. Microbial degradation is included as affecting the localized corrosion rates of the cladding
2.1.12.06.00	Gas transport in waste and EBS	Excluded	WF, EBS	ANL-WIS-MD-000009, ANL-WIS-MD-000006	Excluded based on low consequence: Screening studies conclude that if most gaseous radioisotopes, except C-14, were to be released through an atmospheric pathway, the potential dose relative to the ingestion dose would be low thus most gaseous radioisotopes were screened out (waste inventory FEP 2.1.01.01.00). In TSPA-SR, rather than transport C-14 as a gas (which would normally be incorporated in CO_2), C-14 was transported to a receptor entirely in the aqueous phase. This approach is conservative since less dilution occurs in aqueous transport).
2.2.08.12.00	Use of J-13 Well water as a surrogate for water flowing Into the EBS and waste	Included	WF	ANL-WIS-MD-000009	J-13 well water is assumed to have equilibrated with minerals in Yucca Mountain and is, therefore, assumed to be representative of ground water passing through the repository. Hence, J-13 well water is used to set initial water chemistry inside the WP.

Table 15. FEPs Related to the CSNF Matrix Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
1.2.04.04.00	Magma interacts with waste	Included	WF, Tec, WP	ANL-WIS-MD-000009	Included: Direct volcanic disruption of the repository is included as a process in TSPA-SR. Hence, magmatic interaction with the waste is plausible. This FEP sets plausible particle sizes for the fragmented waste. Direct volcanic effects (i.e., radioisotopes carried by ash plumes from volcanic eruptions) are modeled as described in the Disruptive Events Report.
2.1.02.02.00	CSNF alteration, dissolution, and radionuclide release	Included	WF	ANL-WIS-MD-000009	CSNF degradation is included in the TSPA-SR analysis as a source term for the mobilization of contaminants. A kinetic rate equation, assuming saturated conditions, is used where terms of the coefficients have been evaluated through regression analysis on experimental data obtained over a range of temperatures and water chemistry.
2.1.02.04.00	Alpha recoil enhances dissolution	Excluded	WF	ANL-WIS-MD-000009	Excluded based on low consequence. The radioactive decay processes that directly increase waste matrix corrosion are bounded by alpha-recoil rates. Even when assuming that all radioactive decay result in an alpha-recoil, analyses shows that it will not cause significant increases to the degradation rate of the different waste forms (CSNF, DSNF, or HLW).
2.1.08.15.00	Waste-form and backfill consolidation	Excluded	WF	ANL-WIS-MD-000009	Excluded based on consequence No credit is taken for backfill consolidation since it would tend to decrease the available reactive surface area and decrease degradation rates. The potential deleterious effect of maintaining water contact with the waste form is already conservatively bounded (with or without backfill) by assuming the waste package is fully flooded with water.

Table 15. FEPs Related to the CSNF Matrix Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.11.05.00	Differing thermal expansion of repository component	Included/ Excluded	WF	ANL-WIS-MD-00009; ANL-EBS-MD-000015	Degradation of CSNF is a function of temperature in the CSNF Matrix Degradation Component. This FEP is the topic of CSNF Waste Form Degradation: Summary and Abstraction (CRWMS M&O 2000c); influence of temperature is also discussed in FEPs 2.11.11.07.00 on cladding and 2.1.11.01.00 on HLW. Excluded based on low consequence, thermally induced stress changes for the near-field barriers and EBS.
2.1.13.01.00	Radiolysis	Excluded	WF, EBS, WP	ANL-WIS-MD-000009	Excluded based on low consequence. Fluid is not expected to contact the waste (CSNF, DSNF, or HLW) until gamma radiolysis has become negligible. Specifically for CSNF and HLW, alpha radiolysis is included since irradiated fuel and active HLW was used for some of the experiments from which regression equations were developed. It is, therefore, not necessary to include radiolysis explicitly in the models.
2.1.13.02.00	Radiation damage in waste and EBS	Excluded	WF, WP, EBS	ANL-WIS-MD-000009, ANL-EBS-MD-000015; ANL-WIS-MD-000010	Excluded based on low consequence. Experimental data show little influence of burnup on degradation rate of CSNF as described in AMR, *CSNF Waste Form Degradation: Summary Abstraction* (CRWMS M&O 2000c). However, radiation damage was conservatively assumed for evaluating the solubility of Pu, Pa, and Cm when the controlling solid was unknown; that is, an amorphous controlling solid was used; thereby, indirectly assuming radiation damage to the crystal lattice of the unknown controlling solid.

Table 15. FEPs Related to the CSNF Matrix Degradation Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.07.00	Gap and grain release of radionuclides after cladding perforation	Included/ Excluded	WF	ANL-WIS-MD-000008	Included: The inventory of radionuclides located at gap and grain boundaries is included. This inventory is released either when the WP fails for cladding that is already perforated or when the cladding is first perforated. Excluded based on low consequence: Generation of additional gap and grain-boundary inventory *after* emplacement in the repository since temperatures are not great enough to promote sufficient diffusion. Interaction with impurities in the waste matrix which mitigate the release of the gap and grain boundary inventory is also excluded both because the reactions are thought to be limited and because it is conservative to do so.
2.1.02.11.00	Waterlogged rods	Excluded	WF	ANL-WIS-MD-000008	Excluded based on low consequence. The moisture content of waste will be controlled by Waste Acceptance Criteria. The moisture in a dried fuel rod using standard drying processes is insufficient to cause further degradation of spent fuel cladding or structural stainless steel of container.

Table 16. FEPs Related to the CSNF Cladding Degradation Component (Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.12.00	Cladding degradation before YMP receives it	Included	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000048	The CSNF Cladding Degradation Component includes a distribution with a mean 0.1 percent cladding of fuel perforated during reactor operations. The minor additional failures during less harsh storage and transportation conditions are also included but do not substantially change the distribution of perforations. This FEP is the topic of AMR, *Initial Cladding Condition* (CRWMS M&O 2000ak).
2.1.02.13.00	General corrosion of cladding	Excluded	WF	ANL-WIS-MD-000008	Excluded based on low consequence. Generalized corrosion of Zircaloy cladding is very slow and will not be important in the first 10,000 yrs. or even 100,000 yrs.
2.1.02.14.00	Microbial corrosion (MIC) of cladding	Excluded	WF	ANL-WIS-MD-000008	Excluded based on low probability (not credible). Microbial activity has been screened out as a process at YMP (colloid FEP 2.1.10.01.00). Even if some microbial activity occurred, there is no experimental evidence of enhanced microbial corrosion of zirconium metal or alloys.
2.1.02.15.00	Acid corrosion of cladding from radiolysis	Excluded	WF	ANL-WIS-MD-000008	Excluded based on low consequence. Zirconium is used for fuel cladding because of its high resistance to corrosion in highly acidic environments including those local environments in high radiation fields. Studies show that zirconium has excellent corrosion resistance to nitric acids and concentrated hydrogen peroxide.
2.1.02.16.00	Localized corrosion of cladding through pitting	Excluded	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000012	Excluded based on low consequence: The conditions for pitting corrosion of the cladding (very low pH and high negative ion concentration) are not present inside WP. For example, the low pH and high chloride concentrations necessary for $FeCl_3$ enhanced corrosion of the cladding are not present inside the WP. Thus in general, pitting corrosion is excluded except for localized corrosion from fluorides, which is included as discussed in cladding FEP 2.1.02.27.00. This FEP is the topic of AMR, *Clad Degradation--Local Corrosion of Zirconium and its Alloys Under Repository Conditions* (CRWMS M&O 2000ao).

Table 16. FEPs Related to the CSNF Cladding Degradation Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding	Excluded	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000012	Excluded based on low probability (not credible). The CSNF Cladding Degradation Component excludes a component that accounts for localized corrosion of the cladding through crevice corrosion because zirconium does not corrode in this manner. This FEP is the topic of AMR, *Clad Degradation–Local Corrosion of Zirconium and its Alloys Under Repository Conditions* (CRWMS M&O 2000ao).
2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding	Excluded	WF	ANL-WIS-MD-000008	Excluded based on low consequence. Silica is stable and is not corrosive to most materials, including Zircaloy. The corrosion of Zircaloy in the presence of fluoride contamination in silica saturated water is part of the localized corrosion by fluoride (cladding FEP 2.1.02.27.00).
2.1.02.19.00	Creep rupture of cladding	Included	WF	ANL-WIS-MD-000008; ANL-WIS-MD-000007	Perforation of Zircaloy cladding from creep rupture at high temperature is included in the CSNF Cladding Degradation Component as described in detail in AMR, *Clad Degradation–Summary and Abstraction* (CRWMS M&O 2000ao).
2.1.02.20.00	Pressurization from He production causes cladding failure	Included	WF	ANL-WIS-MD-000008	Creep perforation is driven by the cladding stress caused by the internal gas (including initial fill gas, fission product gases, and helium gas from alpha decay) and, thus indirectly, pressurization is included in the CSNF Cladding Degradation Component.
2.1.02.21.00	Stress corrosion cracking (SCC) of cladding	Included	WF	ANL-WIS-MD-000008; ANL-WIS-MD-000007	Stress corrosion cracking is modeled as a perforation mechanism for the rods with the high stresses. The SCC is primarily relevant to iodine since SCC from other halogens such as chlorides is generally not observed. However, reactive fluorine is considered independently as a localized corrosion mechanism (cladding FEP 2.1.02.27.00).
2.1.02.22.00	Hydride embrittlement of cladding	Excluded	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000011	Excluded based on low consequence. Available experimental data indicate that the in-package environment and cladding stresses are not conducive to hydride cracking and embrittlement. This FEP is the subject of AMR, *Hydride-Related Degradation of SNF Cladding under Repository Conditions* (CRWMS M&O 2000g).

Table 16. FEPs Related to the CSNF Cladding Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.23.00	Cladding unzipping	Included/ Excluded	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000013; ANL-EBS-MD-000014	Included: Unzipping of cladding in an aqueous environment is included as the key element in exposing and dissolving fuel after perforation of the cladding. This mechanism conservatively bounds the slow diffusive release of radionuclides through pinholes of the fuel cladding, *Clad Degradation–Summary and Abstraction* (CRWMS M&O 2000w). Excluded: Unzipping of cladding in a dry environment is excluded based on low consequence. In the environment of the repository only a very small fraction of the fuel would undergo dry unzipping, and then only if the disposal container was breached in the first few hundred years, and the cladding was already perforated such that the UO_2 would be rapidly oxidized to U_3O_8. Dry unzipping is the topic of AMR, *Clad Degradation–Dry Unzipping* (CRWMS M&O 2000ap).
2.1.02.24.00	Mechanical failure of cladding	Included	WF	ANL-WIS-MD-000008	The CSNF Cladding Degradation Component includes perforation of cladding from severe earthquakes (once per million years).
2.1.02.26.00	Diffusion-controlled cavity growth (DCCG) concerns	Excluded	WF	ANL-WIS-MD-000008, ANL-WIS-MD-000007	Excluded based on low probability. Diffusion-controlled cavity growth as a mechanism to fail Zircaloy cladding has not been observed experimentally. Excluded based on low probability (not credible). NRC once required license applicants for dry storage of CSNF to assume DCCG to evaluate dry storage designs. Recent literature does not support the use of this specific creep mechanism for zirconium materials since it has not been observed experimentally (voids and cavities are rarely seen in irradiated Zircaloy). Current NRC Interim Staff Guidance permits license applicants to use other creep models. Although DCCG is excluded as a specific type of creep rupture mechanism, creep rupture, as a general cladding perforation process, is included (see cladding FEP 2.1.02.19.00).

Table 16. FEPs Related to the CSNF Cladding Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.27.00	Localized corrosion perforation of cladding by fluoride	Included	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000012	The CSNF Cladding Degradation Component includes a component that accounts for localized corrosion of the cladding through corrosion by fluorides flowing into the WP. This FEP is the topic of AMR, *Clad Degradation–Local Corrosion of Zirconium and its Alloys Under Repository Conditions* (CRWMS M&O 2000ao).
2.1.07.01.00	Rockfall (large block)	Excluded/ Included	WF	ANL-WIS-MD-000008	Excluded based on low consequence. Cladding perforation from the collapse of the WP with a large block rockfall is not considered since cladding perforation from other modes occurs before the WP can collapse Included in determining cladding failure during seismic events as part of disruptive events analyses(see also FEP 2.1.02.24.00).
2.1.09.03.00	Volume increase of corrosion products	Included/ Excluded	WF, WP	ANL-WIS-MD-000008	Included: The underlying driving mechanism assumed for wet unzipping is the volume increase as the UO_2 forms secondary minerals, and thus, this FEP is included in the CSNF Cladding Degradation Component. Excluded-based on low consequence. Dry oxidation of CSNF is not expected to be a significant contributor to the results since it can only occur when both the WP and cladding fail in the first 300 years (see cladding FEP 2.1.02.23.00).
2.1.11.07.00	Thermally induced stress changes in waste and EBS	Included/ Excluded	EBS, WF	ANL-WIS-MD-000009; ANL-WIS-MD-000007	Included: Thermally induced stress changes in the CSNF cladding influences creep rupture and SCC perforations in the CSNF Cladding Degradation Component. Furthermore, temperature differences on different fuel assemblies in the cross-section of the WP are considered in the first 1000 yr. Degradation of the waste matrix of CSNF and HLW is also a function of temperature (CSNF matrix FEP 2.1.11.05.00 and HLW FEP 2.1.11.01.00). Excluded based on low consequence: thermally induced stress changes in fuel assemblies and packaging. Current degradation models bound effects due to thermally induced stress changes.

Table 17. FEPs Related to the DSNF Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.01.00	DSNF degradation, alteration, and dissolution	Included	WF	ANL-WIS-MD-000009	Included: DSNF and its degradation is included in TSPA-SR. The largest component of the DSNF is the uranium metal, N-Reactor fuel, and since the corrosion characteristics of uranium metal conservatively bound the degradation of all other DSNF, it is used as a surrogate for DSNF in TSPA-SR. (The inventory, however, is the weighted mass average of all the ~250 types of DSNF, Waste Inventory FEP 2.1.01.01.00).
2.1.02.08.00	Pyrophoricity	Excluded	WF	ANL-WIS-MD-000009	Excluded based on low consequence. Based on scoping analysis, the heat produced by a pyrophoric event is not sufficient to impact the performance of an adjacent WP. The performance on the DSNF waste form itself is not affected since no credit for cladding is assumed and the degradation rate of the matrix is already conservatively bounded.
2.1.02.29.00	Flammable gas generation from DSNF	Excluded	WF	ANL-WIS-MD-000009	Excluded based on low consequence. The chemical reaction of U/Th carbide with water could produce CO_2 and hydrocarbon gases. The major constituent of the gas is methane; minor constituents are ethane, ethylene, and acetylene. Only DSNF from the Peach-Bottom Core 1 has uncoated U/Th carbide pellets and thus the potential to generate flammable gases over a short period of time and, thereby, consequences from gas explosion. However, the Peach Bottom core 1 is in only ~100 WPs out of~3900 DSNF WPs and of ~11,800 WPs. Also, relatively good connection with the surface through fractures and fault zones would eventually disperse any flammable gas in the repository before explosive concentrations could be obtained (see in-package chemistry gas generation FEP 2.1.12.01.00).

Table 17. FEPs Related to the DSNF Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.25.00	DSNF cladding degradation	Excluded	WF	ANL-WIS-MD-000009	Excluded based on low consequence: No protective performance credit is taken for DSNF cladding (except for Naval SNF cladding). First more than 80% of the DSNF is from the N-Reactor and is in poor condition; up to 50% may already be perforated; hence, all the cladding is conservatively assumed to be perforated. Second, unzipping of the cladding is conservatively neglected since the bounding alteration rate of the DSNF fuel matrix is so high and the radioisotope inventory is so small. The fact the DSNF fuel rapidly degrades influences the in-package chemistry pH in the first 1000 yr (in-package chemistry co-disposal FEP 2.1.01.02.00)
2.1.11.03.00	Exothermic reactions and other thermal effects in waste and EBS	Excluded	EBS, WF	ANL-WIS-MD-000009	Excluded based on low consequence. The possible temperature rise in a disposal container from exothermic degradation of DSNF is inconsequential in comparison to heat generated by radioactive decay. The fixed, conservative rate bounds by at least an order of magnitude the maximum degradation rate observed and thus also bounds any thermal effects on waste degradation.
2.1.12.08.00	Gas explosions	Excluded	EBS WF	ANL-WIS-MD-000009	Excluded based on low consequence. Because of the absence of O_2 and water, it is very unlikely that an explosion in an inert gas environment within the canister will occur until the waste package and DSNF canister is breached. After the canister is breached, generation of flammable gases have been excluded (see DSNF FEP 2.1.02.29.00). Furthermore, any gas produced will disperse into the drift. The permeability of Yucca Mountain provides adequate connection to the surfaces such that flammable/explosive gases would be diluted and/or dispersed before they could reach explosive concentrations (see in-package chemistry gas generation FEP 2.1.12.01.00)

Table 18. FEPs Related to the HLW Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.02.03.00	Glass degradation, alteration, and dissolution	Included/ Excluded	WF	ANL-WIS-MD-000009	Included: HLW degradation included in the TSPA-SR analysis as a source term for the congruent mobilization of contaminants. Degradation rate is dependent on pH of in-package water chemistry. The effects of minor phase separation are subsumed in the degradation model since parameters for the model are derived from dissolution experiments on samples where minor phase separation might have occurred. Excluded based on low consequence: Extensive phase separation ,precipitation of silicates and other minerals and selective leaching. Extensive phase separation is excluded because it is controlled during production. Removal of silicates by precipitation can be ignored because it is over compensated by the rapid dissolution of the glass. Also, no credit is taken for retarding effects of the precipitated silicates. Selective leaching is conservatively excluded; however, the rate of degradation used for the glass matrix is an upper bound on the actual rate of release and, therefore, bounds any additional consequence due to selective leaching.
2.1.02.05.00	Glass cracking and surface area	Included	WF	ANL-WIS-MD-000009	The thermal stress during cooling increases the surface area accessible to water through cracking beyond the geometric surface area; a robust container and cool temperatures, relative to the transition temperature of glass, prevent extensive increases in this initial surface area during disposal. None the less, a very conservative cracking surface area-enhancement factor of 20 is used to establish the surface area accessible to water for degradation.
2.1.02.06.00	Glass recrystallization	Excluded	WF	ANL-WIS-MD-000009	Extensive recrystallization excluded based on low consequence. Controls are to be implemented as part of the waste production to ensure that extensive glass recrystallization and phase separation will not occur.

Table 18. FEPs Related to the HLW Degradation Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.11.01.00	Heat output / temperature in waste and EBS	Included	WF, EBS, NFE	ANL-WIS-MD-000009	Included: Decay heat is a major issue in repository design, where high loading densities and high temperatures (~96°C) are intended to be part of the waste isolation scheme. In particular for the waste form, degradation of HLW is a function of temperature. See also CSNF FEP 2.1.11.05.00 and Cladding FEP 2.1.11.07.00.

Table 19. FEPs Related to the Dissolved Radioisotope Concentration Component
(Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.04.00	Radionuclide solubility, solubility limits, and speciation in the waste form and EBS	Included	WF	ANL-WIS-MD-000009	A controlling phase is used to develop a dissolved concentration limit for each of the transported radioactive elements. These limits are expressed as an empirical equation, distribution, or a bounding constant depending on the element and available data. The concentration limit is used to bound the amount of a radionuclide element that can be mobilized by the Dissolved Radioisotope Concentration Component.
2.1.09.10.00	Secondary phase effects on dissolved radionuclide concentrations at the waste form	Excluded	WF	ANL-WIS-MD-000009	Excluded based on low consequences. No credit is taken for secondary phase effects radionuclide concentration. The solubility of many radionuclides depend only on pure phases; however, a few radionuclides such as Np may be incorporated into the structure of phases of other minerals (primarily uranium) that form during degradation of the waste. These other minerals control the concentration of the radioisotope.; Hence, solubility of Np is somewhat greater than in TSPA-VA.
2.1.09.12.00	Rind (altered zone) formation in waste, EBS, and adjacent rock	Included/ Excluded	EBS, NFE WF	ANL-WIS-MD-000009	Included: For determining the radioisotope concentration, the Dissolved Radioisotope Concentration Component assumes that the volume of water available for dissolution in the waste form cell is equal to the pore space of the rind of alteration products that forms as the UO_2 is converted into secondary minerals. Excluded based on low consequences: No credit is taken for the Rind or altered zone ability to reduce advective flow past the waste and, thus, reduce release of radionuclides;

Table 19. FEPs Related to the Dissolved Radioisotope Concentration Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.13.00	Complexation by organics in waste and EBS	Excluded	EBS WF	ANL-WIS-MD-000009	Excluded based on low consequence. Based on the preliminary waste acceptance criteria, organic material will be excluded from the radioactive waste: hence, only small amounts of organics could occur in the drifts and even then it will be insolated incidents. For example, organic complexes have not been associated with Yucca Mountain waters, now or in the past. Furthermore, drift temperatures are sufficient to drive off many volatile organics, should they occur.

Table 20. FEPs Related to the Colloidal Radioisotope Concentration Component (Gray shading indicates Miscellaneous WF FEPs in this AMR)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.05.00	In Package Sorption	Included/ Excluded	WF	ANL-WIS-MD-000009	Included: Sorption on mobile colloidal material is included. Excluded based on low consequence: no credit is taken for sorption on stationary material inside the waste package. This is conservative since inclusion of sorption only decreases the release rate of radioisotopes.
2.1.09.14.00	Colloid formation in waste and EBS	Included	WF, EBS	ANL-WIS-MD-000012	A colloid source term has been developed for the Waste Form Degradation Model of the TSPA-SR analysis using experimental data produced from YMP investigations.
2.1.09.15.00	Formation of true (real) colloids in waste and EBS	Excluded	WF, EBS	ANL-WIS-MD-000012	Excluded based on the low probability (not credible): true colloids would be expected to dissolve when the solution is diluted.
2.1.09.16.00	Formation of pseudo-colloids (natural) in waste and EBS	Included	WF, EBS	ANL-WIS-MD-000012	The Colloidal Radioisotope Concentration Component assumes pseudo-colloids may form from groundwater colloids and can reversibly sorb Am, Pu, and other radionuclides.
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and EBS	Included	WF, EBS	ANL-WIS-MD-000012	The Colloidal Radioisotope Concentration Component assumes colloids are produced from corrosion of the disposal container and other material inside the container and sorb reversibly Am, Pu, and other radionuclides. These colloids are conservatively assumed to be similar to natural colloids (see colloid FEP 2.1.09.16.00).
2.1.09.18.00	Microbial colloid formation in the waste and transport in EBS.	Excluded	WF, EBS	ANL-WIS-MD-000012	Excluded based on low consequence (beneficial). Microbial activity has been screened out as a FEP as concerns the waste form (colloid FEP 2.1.10.01.00). However, even if microbial activity were present, it tends to increase colloid size over time, which would result in increased gravitational settling and filtration.

Table 20. FEPs Related to the Colloidal Radioisotope Concentration Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.19.00	Colloid transport and sorption in the waste and EBS.	Excluded	WF, EBS	ANL-WIS-MD-000012	Excluded based on low consequence (beneficial). The Colloidal Radioisotope Concentration Component assumes colloids reversibly sorb or irreversibly incorporate radionuclides; however, the modeling component only acts as a source-term and no credit is taken for the delaying effect of the transport of these colloids inside the WP. Filtration is also excluded (see colloid FEP 2.1.09.20.00).
2.1.09.20.00	Colloid filtration.	Excluded	WF, EBS	ANL-WIS-MD-000012	Excluded based on low consequence (beneficial). No credit is taken for colloid filtration. The conservative assumption is made that all colloids produced within the WP (the calculated colloid source term) exit the WP and enter the drift/EBS. Colloid transport time is also excluded (see colloid FEP 2.1.09.19.00).
2.1.09.21.00	Suspensions of particles larger than colloids	Excluded	WF, EBS, SZ	ANL-WIS-MD-000012	Excluded based on low consequence. It is unlikely that larger-than-colloid particles will have access to a sufficient number of vertical or sub-vertical fractures whose apertures permit their passage to be transported in significant quantities. The relatively small quantity of particles that may make it through the UZ would encounter low groundwater velocities in the SZ, which would likely result in the particles settling and becoming immobile.
2.1.09.22.00	Colloidal Sorption at the air-water interface	Excluded	WF	ANL-WIS-MD-000012	Excluded based on low consequence (beneficial). Although colloids may be retained at the air-water interface in unsaturated conditions, no credit is taken for it. Neglecting this phenomenon is conservative; thus, the conceptual model of waste package uses a fully saturated mixing cell.

Table 20. FEPs Related to the Colloidal Radioisotope Concentration Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.09.23.00	Colloidal Stability and concentration dependence on aqueous chemistry	Included	WF	ANL-WIS-MD-000012	The Colloidal Radioisotope Concentration Component develops (based on experimental data) stability curves as a function of pH and ionic strength.
2.1.09.24.00	Colloidal diffusion	Included	WF	ANL-WIS-MD-000012	The Colloidal Radioisotope Concentration Component uses a diffusion coefficient of 10^{-2} i.e., the diffusion coefficient of colloids is a factor of 100 less than the diffusion coefficient of dissolved radionuclides.
2.1.09.25.00	Colloidal phases produced by coprecipitation in the waste package or EBS.	Included	WF	ANL-WIS-MD-000012	The Colloidal Radioisotope Concentration Component assumes colloids produced from degradation of HLW glass incorporate Am, Pu, and other radioisotopes irreversibly.
2.1.09.26.00	Colloid Gravitational Settling	Excluded	WF	ANL-WIS-MD-000012	Excluded based on low consequence (beneficial). The conservative assumption is made that gravitational settling does not occur, but instead all of the colloids present at a given ionic strength and pH condition leave a breached waste package.

Table 20. FEPs Related to the Colloidal Radioisotope Concentration Component (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

FEP Number	FEP Title	Include/ Exclude	Process Model Report (PMR)	AMR Document Identifier (DI)	Reason for Include/Exclude Decision
2.1.10.01.00	Biological activity in waste and EBS	Excluded/ Included	WF, EBS, WP	ANL-WIS-MD-000012	Excluded based on low probability (not credible) for microbe-facilitated radionuclide transport. Excluded based on low consequence (beneficial) for microbe-induced bioreduction of dissolved multivalent radionuclides, metals, and sulfate impact of microbial biofilms on colloid-facilitated radionuclide transport. Since organic matter is excluded from the waste based on the preliminary waste acceptance criteria, preliminary analysis shows that sufficient quantities of microbes will not be available to beneficially affect colloid mobility or adversely accelerate corrosion rates significantly. Included for microbe-induced acceleration of corrosion; generation of corrosion-product colloids. Mass concentration of corrosion-product colloids is represented by a steady-state cocnetartion-colloids consisting of iron-(hydr)oxide.
2.1.13.03.00	Mutation	Excluded	WF, EBS	ANL-WIS-MD-000012	Excluded based on low consequence. General principles of population genetics indicate that most mutations are either neutral or deleterious to the fitness of an organism and, in the absence of strong natural selection, are unlikely to produce any definite change in the phenotypes of the organisms.

Table 21. Key Technical Issues

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

Table 22. Mapping between Waste-Form FEPs and TSPAI Key Technical Issues

TSPAI Sub-Issue	Acceptance Criteria	FEPs	
		Number	Title
3) MODEL ABSTRACTION			
Engineered System			
Engineered Barriers			
4) Radionuclide Release Rates and Solubility Limits Pertinent KTI subissues: CLST3, CLST4, ENFE3, RDTME1, RDTME3	Criterion T5: Important design features, physical phenomena and couplings, and consistent and appropriate conditions are incorporated into the radionuclide release rates and solubility limits abstraction.	CLST 3 and CLST 4 correspond to Tables 14-20.	
4) SCENARIO ANALYSIS			
1) Identification of an Initial Set of Processes and Events	Criterion T1: DOE has identified a comprehensive list of processes and events that (1) are present or might occur in the Yucca Mountain region and (2) includes those processes and events that have the potential to influence repository performance.	See Tables 13-20.	

NOTE: See NRC (1998b, Sections 4.3 and 4.4) for discussion of KTIs.

Table 23. Mapping between Waste-Form FEPs and CLST Key Technical Issues

CLST Sub-Issue	Acceptance Criteria	FEPs	
		Number	Title
3. The rate at which radionuclides in Spent Nuclear Fuel are released from the Engineered Barrier System through the oxidation and dissolution of spent fuel	1. DOE has considered all categories of SNF planned for disposal at the potential YM repository.	See Table 15.	
	2. DOE has adequately justified the selection of radionuclides tracked in the release models from SNF and their related release parameters.	2.1.01.01.00	Waste inventory
	3. DOE has identified the range of environmental conditions to be expected inside breached waste packages.	See Table 14.	
	4. DOE has identified and considered likely processes for SNF degradation and the release of radionuclides from the EBS, as follows: dissolution of the irradiated UO_2 matrix, with the consequent formation of secondary minerals and colloids; prompt release of radionuclides; degradation in the dry-air environment; degradation and failure of fuel cladding; preferential dissolution of intermetallics in DOE SNF; and release of radionuclides from the waste-package emplacement drifts.	See Tables 15-20.	
	6. DOE has considered the compatibility of SNF and the internal components of the waste package such as the basket material in the evaluation of radionuclide release. Specifically the SNF should not compromise the performance of the waste package.	2.1.03.06.00	Internal corrosion of waste container.
4. The rate at which radionuclides in HLW glass are leached and released from the EBS.	1. DOE has taken into account all types of HLW glass planned for YM disposal.	See Table 18.	
	3. DOE has identified the range of environmental conditions to be expected inside breached waste packages containing HLW glass and certain types of SNF, as in the co-disposal waste packages.	See Table 14.	
	4. DOE has identified and considered likely processes for the degradation of HLW glass and the release of radionuclides from the EBS, i.e., dissolution of the primary phase; formation of secondary minerals and colloids; microbial action; and radionuclide releases and transport from the waste-package emplacement drifts.	See Tables 18-20.	

Table 23. Mapping between Waste-Form FEPs and CLST Key Technical Issues (Continued)

CLST Sub-Issue	Acceptance Criteria	FEPs	
		Number	Title
	6. DOE has assessed the compatibility of HLW glass with internal components of the waste package in the evaluation of radionuclide release, taking into consideration co-disposal with DOE-owned SNF in the same waste package. Specifically, HLW glass should not compromise the performance of the waste package.	2.1.03.06.00	Internal corrosion of waste container.

NOTE: See NRC (1998a, Sections 4.1, 4.3 and 4.4) for discussion of KTIs.

Table 24. Mapping between Waste-Form FEPs and ENFE Key Technical Issues

ENFE Sub-Issue	Acceptance Criteria	FEPs	
		Number	Title
ENFE-3: Effects of coupled thermo-hydrologic chemical processes on radionuclide transport through engineered and natural barriers.	1. Available data relevant to both temporal and spatial variations in conditions affecting coupled THC effects on the chemical environment for radionuclide release were considered.	See Tables 13, 14.	
	2. DOE's evaluation of coupled THC processes properly considered site characteristics in establishing initial and boundary conditions for conceptual models and simulations of coupled processes that may affect the chemical environment for radionuclide release.	See Table 14.	
	3. Sufficient data were collected on the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC-coupled processes that affect the chemical environment for radionuclide release.	See Tables 14-20.	
	4. A nutrient and energy inventory calculation should be used to determine the potential for microbial activity that could impact radionuclide release.	See Table 20.	
	5. Should microbial activity be sufficient to potentially affect the chemical environment for radionuclide release, then the time-history of temperature, humidity, and dripping should be used to constrain the probability for microbial effects, such as production of organic by-products that act as complexing ligands for actinides and microbial-enhanced dissolution of the HLW glass form.	See Table 20.	

NOTE: See NRC (1997, Section 4.4) for discussion of KTIs.

8. REFERENCES

8.1 DOCUMENTS CITED

Ahlers, C.F.; Finsterle, S.; and Bodvarsson, G.S. 1999. "Characterization and Prediction of Subsurface Pneumatic Response at Yucca Mountain, Nevada." *Journal of Contaminant Hydrology, 38,* (1-3), 47-68. New York, [New York]: Elsevier. TIC: 244160.

Bird, R.B.; Stewart, W.E.; and Lightfoot, E.N. 1960. *Transport Phenomena.* New York, New York: John Wiley & Sons. TIC: 208957.

Cranwell, R.M.; Guzowski, R.V.; Campbell, J.E.; and Ortiz, N.R. 1990. *Risk Methodology for Geologic Disposal of Radioactive Waste, Scenario Selection Procedure.* NUREG/CR-1667. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: NNA.19900611.0073.

Croff, A.G. 1980a. *A User's Manual for the ORIGEN2 Computer Code.* ORNL/TM-7175. Oak Ridge, Tennessee: Oak Ridge National Laboratory. TIC: 225504.

Croff, A.G. 1980b. *ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code.* ORNL-5621. Oak Ridge, Tennessee: Oak Ridge National Laboratory. TIC: 242654

CRWMS M&O 1995. *Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository.* B00000000-01717-2200-00136 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19960724.0188.

CRWMS M&O 1996. *Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada.* BA0000000-01717-2200-00082 REV 0. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971201.0221.

CRWMS M&O 1997a. *UCF WP Static Loads, Thermal Expansion Loads, and Internal Pressure Analysis.* BBAAA0000-01717-0200-00001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971014.0668.

CRWMS M&O 1997b. *Waste Package Materials Selection Analysis.* BBA000000-01717-0200-00020 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980324.0242.

CRWMS M&O 1997c. *Waste Package Design Basis Events.* BBA000000-01717-0200-00037 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971006.0075.

CRWMS M&O 1998a. "Near-Field Geochemical Environment." Chapter 4 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document.* B00000000-01717-4301-00004 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981008.0004.

CRWMS M&O 1998b. "Thermal Hydrology." Chapter 3 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document.* B00000000-01717-4301-00003 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981008.0003.

CRWMS M&O 1998c. "Waste Form Degradation, Radionuclide Mobilization, and Transport Through the Engineered Barrier System." Chapter 6 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document.* B00000000-01717-4301-00006 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981008.0006.

CRWMS M&O 1998d. "Waste Package Degradation Modeling and Abstraction." Chapter 5 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document.* B00000000-01717-4301-00005 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981008.0005.

CRWMS M&O 1998e. "Geochemistry." Book 3 - Section 6 of *Yucca Mountain Site Description.* B00000000-01717-5700-00019 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980729.0052.

CRWMS M&O 1998f. *Waste Form Degradation and Radionuclide Mobilization Expert Elicitation Project.* Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980804.0099.

CRWMS M&O 1999a. Not used.

CRWMS M&O 1999b. Not used.

CRWMS M&O 1999c. *License Application Design Selection Report.* B00000000-01717-4600-00123 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990528.0303.

CRWMS M&O 1999d. *Defense High Level Waste Glass Degradation.* Input Transmittal PA-WP-99314.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991027.0154.

CRWMS M&O 1999e. Not used.

CRWMS M&O 1999f. *Total System Performance Assessment Calculation - Defense-In-Depth Calculations in Support of LA Design Selection Process.* B00000000-01717-0210-00093 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991130.0155.

CRWMS M&O 1999g. Not Used.

CRWMS M&O 1999h. *Total System Performance Assessment-Site Recommendation Methods and Assumptions.* TDR-MGR-MD-000001 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991104.0570.

CRWMS M&O 1999i. *Status of Radionuclide Screening for the Total System Performance Assessment - Site Recommendation (TSPA-SR).* Input Transmittal R&E-PA-99217.Tb. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991122.0074.

CRWMS M&O 1999j. *BWR Source Term Generation and Evaluation.* BBAC00000-01717-0210-00006 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000113.0334.

CRWMS M&O 1999k. *PWR Source Term Generation and Evaluation.* BBAC00000-01717-0210-00010 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000113.0333.

CRWMS M&O 1999l. *Waste Particle Diameter in Magmatic Environment.* Input Request PA-WP-99388.R. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991130.0074.

CRWMS M&O 1999m. *Department of Energy Spent Nuclear Fuel Grouping.* WP-WP-99407.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991209.0106.

CRWMS M&O 1999n. Not used.

CRWMS M&O 1999o. . *Yucca Mountain Performance Assessment Operations (PAO), Waste Form Workshop, February 2-4, 1999, Albuquerque, New Mexico.* [Las Vegas, Nevada: CRWMS M&O]. ACC: MOL.20000912.0220.

CRWMS M&O 1999p. *Draft Analysis/Modeling Report - Department of Energy Spent Fuel Performance Assessment.* Input Transmittal WP-WP-99408.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991209.0105.

CRWMS M&O 2000a. *Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux.* ANL-EBS-HS-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000504.0296.

CRWMS M&O 2000b. *Technical Work Plan for Waste Form Degradation Process Model Report for SR.* TWP-EBS-MD-000006. Las Vegas, NV: CRWMS M&O.

CRWMS M&O 2000c. *CSNF Waste Form Degradation: Summary Abstraction.* ANL-EBS-MD-000015 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000121.0161.

CRWMS M&O 2000d. *Defense High Level Waste Glass Degradation.* ANL-EBS-MD-000016 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000329.1183

CRWMS M&O 2000e. *DSNF and Other Waste Form Degradation Abstraction.* ANL-WIS-MD-000004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000223.0502.

CRWMS M&O 2000f. *EBS Radionuclide Transport Abstraction.* ANL-WIS-PA-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000526.0329.

CRWMS M&O 2000g. *Hydride-Related Degradation of SNF Cladding Under Repository Conditions.* ANL-EBS-MD-000011 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000319.0048.

CRWMS M&O 2000h. *In Drift Microbial Communities.* ANL-EBS-MD-000038 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000331.0661.

CRWMS M&O 2000i. *In-Package Chemistry Abstraction.* ANL-EBS-MD-000037 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000418.0818.

CRWMS M&O 2000j. . *Inventory Abstraction.* ANL-WIS-MD-000006 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001130.0002.

CRWMS M&O 2000k. *Particle Tracking Model and Abstraction of Transport Processes.* ANL-NBS-HS-000026 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000502.0237.

CRWMS M&O 2000l. *Secondary Uranium-Phase Paragenesis and Incorporation of Radionuclides into Secondary Phases.* ANL-EBS-MD-000019 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000414.0644.

CRWMS M&O 2000m. *Summary of In-Package Chemistry for Waste Forms.* ANL-EBS-MD-000050 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000217.0217.

CRWMS M&O 2000n. *Waste Form Degradation Process Model Report.* TDR-WIS-MD-000001 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000713.0362.

CRWMS M&O 2000o. *Waste Package Radionuclide Inventory Approximations for TSPA-SR.* CAL-WIS-MD-000004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000630.0247.

CRWMS M&O 2000p. *Draft of AMR In-Package Source Term Abstraction (ANL-WIS-MD-000018).* Input Transmittal 00366.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000813.0530.

CRWMS M&O 2000q. *Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA.* ANL-NBS-HS-000030 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000526.0330.

CRWMS M&O 2000r. *Draft of AMR TSPA System-Level FEPs (ANL-WIS-MD-000019) for Use in the Report, Total System Performance Assessment - Site Recommendation.* Input Transmittal 00365.T. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000813.0529.

CRWMS M&O 2000s. *In-Drift Thermal-Hydrological-Chemical Model.* ANL-EBS-MD-000026 REV 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000802.0011.

CRWMS M&O 2000t. *Multiscale Thermohydrologic Model.* ANL-EBS-MD-000049 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000609.0267.

CRWMS M&O 2000u. *Drift-Scale Coupled Processes (DST and THC Seepage) Models.* MDL-NBS-HS-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990721.0523.

CRWMS M&O 2000v. *Drift Scale Thermal Analysis.* CAL-WIS-TH-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000420.0401.

CRWMS M&O 2000w. *Clad Degradation–Summary and Abstraction.* ANL-WIS-MD-000007 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000602.0055.

CRWMS M&O 2000x. *WAPDEG Analysis of Waste Package and Drip Shield Degradation.* ANL-EBS-PA-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000526.0332.

CRWMS M&O 2000y. *Summary of Dissolved Concentration Limits.* ANL-WIS-MD-000010 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000525.0372.

CRWMS M&O 2000z. *Waste Form Colloid-Associated Concentrations Limits: Abstraction and Summary.* ANL-WIS-MD-000012 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000525.0397.

CRWMS M&O 2000aa. *Evaluation of Codisposal Viability for UZrH (TRIGA) DOE-Owned Fuel.* TDR-EDC-NU-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000207.0689.

CRWMS M&O 2000ab. *Evaluation of Codisposal Viability for HEU Oxide (Shippingport PWR) DOE-Owned Fuel.* TDR-EDC-NU-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000227.0240.

CRWMS M&O 2000ac. Not Used.

CRWMS M&O 2000ad. Not used.

CRWMS M&O 2000ae. *Clad Degradation – FEPs Screening Arguments.* ANL-WIS-MD-000008 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000525.0378.

CRWMS M&O 2000af. *Mountain-Scale Coupled Processes (TH) Models.* MDL-NBS-HS-000007 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990721.0528.

CRWMS M&O 2000ag. *Unsaturated Zone Flow and Transport Model Process Model Report.* TDR-NBS-HS-000002 REV 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000831.0280.

CRWMS M&O 2000ah. Not used.

CRWMS M&O 2000ai. Site Recommendation Design Baseline. Technical Change Request T2000-0133. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000503.0159.

CRWMS M&O 2000aj. *Heat Decay Data and Repository Footprint for Thermal-Hydrologic and Conduction Only Models for TSPA-SR.* CAL-MGR-HS-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000516.0007.

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

CRWMS M&O 2000al. *The Development of Information Catalogued in REV00 of the YMP FEP Database.* TDR-WIS-MD-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000705.0098.

CRWMS M&O 2000am. *Performance Assessment Sensitivity Analyses of Selected U.S. Department of Energy Spent Fuels.* ANL-WIS-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000710.0530.

CRWMS M&O 2000an. *Analysis of Hydrologic Properties Data.* ANL-NBS-HS-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990721.0519.

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

CRWMS M&O 2000ap. *Clad Degradation - Dry Unzipping.* ANL-EBS-MD-000013 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000503.0200.

CRWMS M&O 2000aq. *Internal Pressurization Due to Fuel Rod Rupture in Waste Packages.* CAL-EBS-ME-000005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000315.0671.

CRWMS M&O 2000ar. *Design Analysis for UCF Waste Packages.* ANL-UDC-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000526.0336.

Cui, D. and Eriksen, T.E. 1996. "Reduction of Pertechnetate in Solution by Heterogeneous Electron Transfer from Fe(II)-Containing Geological Material." *Environmental Science & Technology, 30,* (7), 2263-2269. Washington, D.C.: American Chemical Society. TIC: 246900.

Cunnane, J.C.; Bates, J.K.; Bradley, C.R.; Buck, E.C.; Ebert, W.L.; Feng, X.; Mazer, J.J.; Wronkiewicz, D.J.; Sproull, J.; Bourcier, W.L.; McGrail, B.P.; and Altenhofen, M.K. 1994a. *High-Level Waste Borosilicate Glass: A Compendium of Corrosion Characteristics, Volume 2.* DOE-EM-0177. Washington, D.C.: U.S. Department of Energy. TIC: 242675.

Cunnane, J.C.; Bates, J.K.; Bradley, C.R.; Buck, E.C.; Ebert, W.L.; Feng, X.; Mazer, J.J.; Wronkiewicz, D.J.; Sproull, J.; Bourcier, W.L.; McGrail, B.P.; and Altenhofen, M.K., eds. 1994b. *High-Level Waste Borosilicate Glass: A Compendium of Corrosion Characteristics.* DOE-EM-0177. Volume 1. Washington, D.C.: U.S. Department of Energy. TIC: 242675.

DOE (U.S. Department of Energy) 1995. *Waste Acceptance Product Specifications for Vitrified High-Level Waste Forms.* EM-WAPS Rev. 01. Germantown, Maryland: U.S. Department of Energy, Office of Environmental Restoration and Waste Management. ACC: HQO.19950615.0009.

DOE (U.S. Department of Energy) 1996. *Title 40 CFR Part 191 Compliance Certification Application for the Waste Isolation Pilot Plant.* DOE/CAO-1996-2184. Twenty-one volumes. Carlsbad, New Mexico: U.S. Department of Energy, Carlsbad Area Office. TIC: 240511.

DOE (U.S. Department of Energy) 1997. *EM/RW Repository Task Team Report: Grouping Method to Minimize Testing for Repository Emplacement of DOE SNF.* DOE/SNF/REP-008, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy. TIC: 247664.

DOE (U.S. Department of Energy) 1998a. *High Integrity Can Materials Selection and Design Data.* DOE/SNF/DSN-002. Washington, D.C.: U.S. Department of Energy. TIC: 245058.

DOE (U.S. Department of Energy) 1998b. *Preliminary Design Concept for the Repository and Waste Package.* Volume 2 of *Viability Assessment of a Repository at Yucca Mountain.* DOE/RW-0508. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19981007.0029.

DOE (U.S. Department of Energy) 1998c. *Total System Performance Assessment.* Volume 3 of *Viability Assessment of a Repository at Yucca Mountain.* DOE/RW-0508. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19981007.0030.

DOE (U.S. Department of Energy) 1998d. *Update to Assessment of Direct Disposal in Unsaturated Tuff of Spent Nuclear Fuel and High-Level Waste Owned by U.S. Department of Energy.* DOE/SNF/REP-015. Washington, D.C.: U.S. Department of Energy. TIC: 243859.

DOE (U.S. Department of Energy) 1999a. *Waste Acceptance System Requirements Document, Revision 03 (E00000000-00811-1708-00001).* DOE/RW-0351. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: HQO.19990226.0001.

DOE (U.S. Department of Energy) 1999b. *DOE Spent Nuclear Fuel Information in Support of TSPA-SR.* DOE/SNF/REP-0047, Rev. 0. [Washington, D.C.]: U.S. Department of Energy, Office of Environmental Management. TIC: 245482.

DOE (U.S. Department of Energy) 1999c. U.S. Department of Energy Spent Nuclear Fuel to the Monitored Geologic Repository for Mechanical and Envelope Interfaces. Volume 1 of Integrated Interface Control Document. DOE/RW-0511, Rev. 0. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19990428.0236.

DOE (U.S. Department of Energy) 2000. *DOE Spent Nuclear Fuel Grouping in Support of Criticality, DBE, TSPA-LA.* DOE/SNF/REP-0046 Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. TIC: 248046.

Dyer, J.R. 1999. "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), September 3, 1999, OL&RC:SB-1714, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain (Revision 01)." ACC: MOL.19990910.0079.

Eide, S.A. 2000. *Feature, Event, and Process Identification to Support Disposal of Department of Energy Spent Nuclear Fuel at the Yucca Mountain Repository.* NSNF/EP-3.05/001, Rev. 00. Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory. On Order Library Tracking Number-247665

Einziger, R.E. and Kohli, R. 1984. "Low-Temperature Rupture Behavior of Zircaloy-Clad Pressurized Water Reactor Spent Fuel Rods Under Dry Storage Conditions." *Nuclear Technology, 67,* (1), 107-123. Hinsdale, Illinois: American Nuclear Society. TIC: 216868.

Einziger, R.E.; Thomas, L.E.; Buchanan, H.C.; and Stout, R.B. 1992. "Oxidation of Spent Fuel in Air at 175 to 195°C." *Journal of Nuclear Materials, 190,* 53-60. Amsterdam, The Netherlands: North-Holland Publishing Company. TIC: 238511.

EPA (U.S. Environmental Protection Agency) 1999. *Background Information Document for 40 CFR 197, Environmental Radiation Protection Standards for Yucca Mountain, Nevada.* Washington, D.C.: U.S. Environmental Protection Agency, Office of Radiation and Indoor Air. TIC: 246926.

Farrell, J.; Bostick, W.D.; Jarabek, R.J.; and Fiedor, J.N. 1999a. "Electrosorption and Reduction of Pertechnetate by Anodically Polarized Magnetite." *Environmental Science & Technology, 33,* (8), 1244-1249. Washington, D.C.: American Chemical Society. TIC: 246903.

Farrell, J.; Bostick, W.D.; Jarabek, R.J.; and Fiedor, J.N. 1999b. "Uranium Removal from Ground Water Using Zero Valent Iron Media." *Ground Water, 37,* (4), 618-624. Westerville, Ohio: National Ground Water Association. TIC: 247564.

Felmy, A.R. and Rai, D. 1999. "Application of Pitzer's Equations for Modeling the Aqueous Thermodynamics of Actinide Species in Natural Waters: A Review." *Journal of Solution Chemistry, 28,* (5), 533-553. New York, New York: Plenum Publishing. TIC: 247104.

Fiedor, J.N.; Bostick, W.D.; Jarabek, R.J.; and Farrell, J. 1998. "Understanding the Mechanism of Uranium Removal from Groundwater by Zero-Valent Iron Using X-Ray Photoelectron Spectroscopy." *Environmental Science & Technology, 32,* (10), 1466-1473. Washington, D.C.: American Chemical Society. TIC: 246904.

Finch, R.J.; Buck, E.C.; Finn, P.A.; and Bates, J.K. 1999. "Oxidative Corrosion of Spent UO2 Fuel in Vapor and Dripping Groundwater at 90°C." *Scientific Basis for Nuclear Waste Management XXII, Symposium held November 30-December 4, 1998, Boston, Massachusetts, U.S.A.* Wronkiewicz, D.J. and Lee, J.H., eds. *556,* 431-438. Warrendale, Pennsylvania: Materials Research Society. TIC: 246426.

Finn, P.A.; Finch, R.; Buck, E.; and Bates, J. 1998. "Corrosion Mechanism of Spent Fuel Under Oxidizing Conditions." *Scientific Basis for Nuclear Waste Management XXI, Symposium held*

September 28-October 3, 1997, Davos, Switzerland. McKinley, I.G. and McCombie, C., eds. *506,* 123-131. Warrendale, Pennsylvania: Materials Research Society. TIC: 240702.

Foster, A.R. and Wright, R.L., Jr. 1973. *Basic Nuclear Engineering.* 2nd Edition. Boston, Massachusetts: Allyn and Bacon. TIC: 247586.

Frondel, C. 1958. *Systematic Mineralogy of Uranium and Thorium.* Bulletin 1064. Washington, D.C.: U.S. Geological Survey. TIC: 246837.

Garde, A.M. 1986. *Hot Cell Examination of Extended Burnup Fuel from Fort Calhoun.* DOE/ET/34030-11. Windsor, Connecticut: Combustion Engineering. TIC: 237128.

Grambow, B.; Smailos, E.; Geckeis, H.; Muller, R.; and Hentschel, H. 1996. "Sorption and Reduction of Uranium(VI) on Iron Corrosion Products Under Reducing Saline Conditions." *Radiochimica Acta, 74,* 149-154. München, Germany: R. Oldenbourg Verlag. TIC: 247716.

Gray, W.J. and Einziger, R.E. 1998. *Initial Results from Dissolution Rate Testing of N-Reactor Spent Fuel Over a Range of Potential Geologic Repository Aqueous Conditions.* DOE/SNF/REP-022, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 245380.

Gray, W.J. and Wilson, C.N. 1995. *Spent Fuel Dissolution Studies FY 1991-1994.* PNL-10450. Richland, Washington: Pacific Northwest Laboratory. ACC: MOL.19960802.0035.

Guenther, R.J. 1983. *Results of Simulated Abnormal Heating Events for Full-Length Nuclear Fuel Rods.* PNL-4555. Richland, Washington: Pacific Northwest Laboratory. TIC: 229965.

Guenther, R.J.; Blahnik, D.E.; Campbell, T.K.; Jenquin, U.P.; Mendel, J.E.; Thomas, L.E.; and Thornhill, C.K. 1988a. . *Characterization of Spent Fuel Approved Testing Material ATM--103.* PNL-5109-103. Richland, Washington: Pacific Northwest Laboratory. ACC: NNA.19911017.0104.

Guenther, R.J.; Blahnik, D.E.; Campbell, T.K.; Jenquin, U.P.; Mendel, J.E.; and Thornhill, C.K. 1988b. *Characterization of Spent Fuel Approved Testing Material--ATM-106.* PNL-5109-106. Richland, Washington: Pacific Northwest Laboratory. ACC: NNA.19911017.0105.

Hanson, B.D. 1998. *The Burnup Dependence of Light Water Reactor Spent Fuel Oxidation.* PNNL-11929. Richland, Washington: Pacific Northwest National Laboratory. TIC: 238459.

Hardin, E.L. 1998. *Near-Field/Altered-Zone Models Report.* UCRL-ID-129179. Livermore, California: Lawrence Livermore National Laboratory. ACC: MOL.19980630.0560.

Jarzemba, M.S.; LaPlante, P.A.; and Poor, K.J. 1997. *ASHPLUME Version 1.0 - A Code for Contaminated Ash Dispersal and Deposition, Technical Description and User's Guide.* CNWRA 97-004, Rev. 1. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. TIC: 239303.

Kaplan, I. 1963. *Nuclear Physics.* 2nd edition. Reading, Massachusetts: Addison-Wesley Publishing Company. TIC: 247872.

Lam, P.S.; Sindelar, R.L.; and Peacock, H.B. Jr. 1997. *Vapor Corrosion of Aluminum Cladding Alloys and Aluminum-Uranium Fuel Materials in Storage Environments (U).* WSRC-TR-97-0120. Aiken, South Carolina: Westinghouse Savannah River Company. TIC: 248909.

Lamarsh, J.R. 1983. *Introduction to Nuclear Engineering.* 2nd Edition. Menlo Park, California: Addison-Wesley. TIC: 244841.

Lederer, C.M. and Shirley, V.S., eds. 1978. *Table of Isotopes.* 7th Edition. New York, New York: John Wiley & Sons. TIC: 243883.

Marschman, S.C.; Pyecha, T.D.; and Abrefah, J. 1997. *Metallographic Examination of Damaged N Reactor Spent Nuclear Fuel Element SFEC5,4378.* PNNL-11438. Richland, Washington: Pacific Northwest National Laboratory. On Order Library Tracking Number-248059

McKinley, I.G.; Hagenlocher, I.; Alexander, W.R.; and Schwyn, B. 1997. "Microbiology in Nuclear Waste Disposal: Interfaces and Reaction Fronts." *FEMS Microbiology Reviews, 20,* (3-4), 545-556. Amsterdam, The Netherlands: Elsevier Science B.V. TIC: 235823.

McKinley, P.W.; Long, M.P.; and Benson, L.V. 1991. *Chemical Analyses of Water from Selected Wells and Springs in the Yucca Mountain Area, Nevada and Southeastern California.* Open-File Report 90-355. Denver, Colorado: U.S. Geological Survey. ACC: NNA.19901031.0004.

McVay, G.L. and Buckwalter, C.Q. 1983. "Effect of Iron on Waste-Glass Leaching." *Journal of the American Ceramic Society, 66,* (3), 170-174. Westerville, Ohio: American Ceramic Society. TIC: 238912.

NRC (U.S. Nuclear Regulatory Commission) 1997. *Issue Resolution Status Report Key Technical Issue: Evolution of the Near-Field Environment.* Rev. 0. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19980709.0392.

NRC (U.S. Nuclear Regulatory Commission) 1998a. *Issue Resolution Status Report Key Technical Issue: Container Life and Source Term.* Rev. 0. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19980416.0784.

NRC (U.S. Nuclear Regulatory Commission) 1998b. *Issue Resolution Status Report Key Technical Issue: Total System Performance Assessment and Integration.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0083.

NRC (U.S. Nuclear Regulatory Commission) 1998c. "Proposed Rule 10 CFR Part 63, Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. SECY-98-225." Washington, D.C.: U.S. Nuclear Regulatory Commission. Accessed 10/20/98. TIC: 240520. http://www.nrc.gov/NRC/COMMISSION/SECYS/1998-225scy.html

NRC (U.S. Nuclear Regulatory Commission) 2000. *Issue Resolution Status Report Key Technical Issue: Total System Performance Assessment and Integration.* Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 247614.

Parrington, J.R.; Knox, H.D.; Breneman, S.L.; Baum, E.M.; and Feiner, F. 1996. *Nuclides and Isotopes, Chart of the Nuclides.* 15th Edition. San Jose, California: General Electric Company and KAPL, Inc. TIC: 233705.

Pashalidis, I.; Kim, J.I.; Lierse, Ch.; and Sullivan, J.C. 1993. "The Chemistry of Pu in Concentrated Aqueous NaCl Solution: Effects of Alpha Self-Radiolysis and the Interaction between Hypochlorite and Dioxoplutonium(VI)." *Radiochimica Acta, 60,* (2/3), 99-101. Munchen, Germany: R. Oldenbourg Verlag. TIC: 247917.

Propp, W.A. 1998. *Graphite Oxidation Thermodynamics/Reactions.* DOE/SNF/REP-018, Rev. 0. Idaho Falls, Idaho: U. S. Department of Energy. TIC: 247663.

Rechard, R.P., ed. 1993. *Initial Performance Assessment of the Disposal of Spent Nuclear Fuel and High Level Waste Stored at Idaho National Engineering Laboratory.* SAND93-2330. Volumes 1 and 2. Albuquerque, New Mexico: Sandia National Laboratories. TIC: 247212

Rechard, R.P., ed. 1995a. *Methodology and Results.* Volume 2 of *Performance Assessment of the Direct Disposal in Unsaturated Tuff of Spent Nuclear Fuel and High-Level Waste Owned by U.S. Department of Energy.* SAND94-2563/2. Albuquerque, New Mexico: Sandia National Laboratories. TIC: 237102.

Rechard, R.P., ed. 1995b. *Appendices.* Volume 3 of *Performance Assessment of the Direct Disposal in Unsaturated Tuff of Spent Nuclear Fuel and High-Level Waste Owned by U.S. Department of Energy.* SAND94-2563/3. Albuquerque, New Mexico: Sandia National Laboratories. TIC: 237103.

Reilly, M.A. 1998. *Spent Nuclear Fuel Project Technical Databook.* HNF-SD-SNF-TI-015, Rev. 5. Richland, Washington: Fluor Daniel Hanford. TIC: 247814.

Sagar, B., ed. 1997. *NRC High-Level Radioactive Waste Program Annual Progress Report: Fiscal Year 1996.* NUREG/CR-6513, No. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19970715.0066.

Sanchez, L.C.; Rath, J.S.; Aguilar, R.; Trellue, H.R.; Cochrane, K.; Taylor, L.L.; and Wilson, J.R. 1998. *Nuclear Dynamics Consequence Analysis (NDCA) for the Disposal of Spent Nuclear Fuel in an Underground Geologic Repository.* DOE/SNF/REP-0033, REV 0. Washington, D.C.: U.S. Department of Energy. TIC: 247643.

Shaw, H., ed. 1999. *Plutonium Immobilization Project Input for Yucca Mountain Total Systems Performance Assessment.* PIP-99-107. Livermore, California: Lawrence Livermore National Laboratory. TIC: 245437.

Shoesmith, D.W. and Sunder, S. 1992. "The Prediction of Nuclear Fuel (UO2) Dissolution Rates Under Waste Disposal Conditions." *Journal of Nuclear Materials, 190,* 20-35. Amsterdam, The Netherlands: Elsevier Science. TIC: 246431.

Skagius, K. and Wingefors, S. 1992. *Application of Scenario Development Methods in Evaluation of the Koongarra Analogue.* Volume 16 of *Alligator Rivers Analogue Project.* SKI TR 92:20-16. DOE/HMIP/RR/92/086. Manai, New South Wales, Australia: Australian Nuclear Science and Technology Organisation. TIC: 231268.

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

Stroupe, E.P. 2000. "Approach to Implementing the Site Recommendation Design Baseline." Interoffice correspondence from E.P. Stroupe (CRWMS M&O) to D.R. Wilkins, January 26, 2000, LV.RSO.EPS.1/00-004, with attachment. ACC: MOL.20000214.0480.

Sunder, S. and Shoesmith, D.W. 1991. *Chemistry of UO[sub]2 Fuel Dissolution in Relation to the Disposal of Used Nuclear Fuel.* AECL-10395. Pinawa, Manitoba, Canada: Whiteshell Laboratories, Atomic Energy of Canada Limited. TIC: 246920.

Sunder, S.; Shoesmith, D.W.; and Miller, N.H. 1997. "Oxidation and Dissolution of Nuclear Fuel (UO[sub]2) by the Products of the Alpha Radiolysis of Water." *Journal of Nuclear Materials, 244,* 66-74. Amsterdam, The Netherlands: Elsevier Science. TIC: 246914.

Swift, P.; Barr, G.; Barnard, R.; Rechard, R.; Schenker, A.; Freeze, G.; and Burck, P. 1999. *Feature, Event, and Process Screening and Scenario Development for the Yucca Mountain Total System Performance Assessment.* SAND98-2831C. Albuquerque, New Mexico: Sandia National Laboratories. ACC: MOL.19990617.0240.

Symonds, R.B.; Rose, W.I.; Bluth, G.J.S.; and Gerlach, T.M. 1994. "Volcanic-Gas Studies: Methods, Results, and Applications." Chapter 1 of *Volatiles in Magmas.* Reviews in Mineralogy Volume 30. Carroll, M.J. and Holloway, J.R., eds. 1-66. Washington, D.C.: Mineralogical Society of America. TIC: 238061.

Van Iseghem, P. and Lemmens, K. 1993. "The Interaction Between HLW Glass and Boom Clay Host Rock." *Geological Disposal of Spent Fuel and High Level and Alpha Bearing Wastes, Proceedings of an International Symposium on Geologic Disposal of Spent Fuel, High Level and Alpha Bearing Wastes, Jointly Organized by the International Atomic Energy Agency, the Commission of the European Communities and the OECD Nuclear Energy Agency held in Antwerp, October 19-23, 1992. IAEA-SM-326/36,* 209-223. Vienna, Austria: International Atomic Energy Agency. TIC: 10198.

Vladimirova, M.V. 1990. "Radiation Chemistry of Actinides." *Journal of Radioanalytical and Nuclear Chemistry, 143,* (2), 445-454. Lausanne, Switzerland: Elsevier Sequoia. TIC: 247759.

Westrich, H.R. 1982. "The Solubility of LWR Core Debris in Sacrificial Floor Material." *Journal of Nuclear Material, 110,* 324-332. Amsterdam, The Netherlands: North-Holland Publishing Company. TIC: 234101.

Wick, O.J., ed. 1980. *Plutonium Handbook: A Guide to the Technology.* Volumes I & II. La Grange Park, Illinois: American Nuclear Society. TIC: 248002.

Wilson, M.L.; Gauthier, J.H.; Barnard, R.W.; Barr, G.E.; Dockery, H.A.; Dunn, E.; Eaton, R.R.; Guerin, D.C.; Lu, N.; Martinez, M.J.; Nilson, R.; Rautman, C.A.; Robey, T.H.; Ross, B.; Ryder, E.E.; Schenker, A.R.; Shannon, S.A.; Skinner, L.H.; Halsey, W.G.; Gansemer, J.D.; Lewis, L.C.; Lamont, A.D.; Triay, I.R.; Meijer, A.; and Morris, D.E. 1994. *Total-System Performance Assessment for Yucca Mountain – SNL Second Iteration (TSPA-1993).* SAND93-2675. Executive Summary and two volumes. Albuquerque, New Mexico: Sandia National Laboratories. ACC: NNA.19940112.0123.

Wolery, T.J. 1992. *EQ3/6, A Software Package for Geochemical Modeling of Aqueous Systems. Package Overview and Installation Guide (Version 7.0).* UCRL-MA-110662 PT I. Livermore, California: Lawrence Livermore National Laboratory. TIC: 205087.

Yang, I.C.; Rattray, G.W.; and Yu, P. 1996. *Interpretation of Chemical and Isotopic Data from Boreholes in the Unsaturated Zone at Yucca Mountain, Nevada.* Water-Resources Investigations Report 96-4058. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19980528.0216.

YMP (Yucca Mountain Site Characterization Project) 1998. *Disposal Criticality Analysis Methodology Topical Report.* YMP/TR-004Q, Rev. 0. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.19990210.0236.

YMP (Yucca Mountain Site Characterization Project) 2000. *Q-List.* YMP/90-55Q, Rev. 6. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.20000510.0177.

8.2 SOURCE DATA

LL000404551021.134. An Estimate of Fuel-Particle Sizes for Physically Degraded Spent Fuel Following a Disruptive Volcanic Event Through the Repository. Submittal date: 04/20/2000.

8.3 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

40 CFR 191. 1993. Protection of Environment: Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes. Readily available.

64 FR 46976. Environmental Radiation Protection Standards for Yucca Mountain, Nevada. Readily available.

64 FR 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Readily available.

AP-3.12Q, Rev. 0, ICN 3. *Calculations.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20001026.0084.

AP-3.15Q, Rev. 2, ICN 0. *Managing Technical Product Inputs.* Washington, D.C: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20001109.0051.

AP-2.21Q, Rev. 0, ICN 0. *Quality Determinations and Planning for Scientific, Engineering, and Regulatory Compliance Activities.* [Washington, D.C.]: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000802.0003.

AP-SI.1Q, Rev. 2, ICN 4, ECN 1. *Software Management.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20001103.0117.

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

ASTM C 1454-00. 2000. *Standard Guide for Pyrophoricity/Combustibility Testing in Support of Pyrophoricity Analyses of Metallic Uranium Spent Nuclear Fuel.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 248977.

Nuclear Waste Policy Act of 1982. Public Law No. 97-425. 96 Stat. 2201. Readily available.

Nuclear Waste Policy Amendments Act of 1987. Public Law No. 100-203. 101 Stat. 1330. Readily available.

ATTACHMENT I—
AN ESTIMATE OF FUEL-PARTICLE SIZES FOR PHYSICALLY DEGRADED SPENT FUEL FOLLOWING A DISRUPTIVE VOLCANIC EVENT THROUGH THE POTENTIAL REPOSITORY

I.1 INTRODUCTION

This document addresses estimates of particle-size distributions for spent nuclear fuel exposed to a potential disruptive magmatic event through the potential repository at Yucca Mountain, Nevada. The input request "Waste Particle Diameter in Magmatic Environment" (CRWMS M&O 1999l) asked for a probability distribution for fuel particles. The distribution would consider mechanical and chemical degradation of the fuel at the time of the disruptive event. The following discussions for waste particle diameter is based on investigations and data generated by Argonne National Laboratory and submitted as DTN: LL000404551021.134. The disruptive event may occur at any time, but the estimated extent of fuel degradation that will have occurred at the time of the event is not addressed here. The following discussion is based on laboratory examinations of commercial spent nuclear fuels, which were conducted for purposes outside the realm of understanding particle size. There is no statistical information available for the distribution of particle sizes caused by the disaggregation and grinding of spent UO_2 fuels in the laboratory. There is a similar paucity of data for oxidized and corroded fuels as well.

The following discussion concerns commercial, spent UO_2-based fuels.

I.2 FUEL DEGRADATION

Three states of fuel degradation can be defined: (1) unaltered fuel (i.e., uncorroded and unoxidized); (2) dry-air oxidized fuel; and (3) aqueous-corroded fuel. Particle sizes are estimated for each below.

I.2.1 Unaltered fuel (uncorroded and unoxidized)

Unaltered spent fuel shows a range of physical characteristics that depend largely on fission-gas release and possibly burnup; however, there is no clear understanding of the relationship between such parameters and the relative ease with which fuel may fragment under stress or the grain sizes that might result from fragmentation. Fission-gas release appears to be a crucial parameter affecting fuel microstructure, including grain growth (Guenther et al. 1988a, 1988b), a characteristic that will strongly impact the distribution of fuel-particle sizes from a fuel following exposure to a disruptive volcanic event.

When crushing spent UO_2 fuel during the preparation of corrosion studies on fuel being conducted at Argonne National Laboratory (ANL), it was found that reducing the particle sizes of a fuel of moderate burnup [approved testing material (ATM) 103: ~ 30 MW d/kg-U] was readily achieved by initial crushing with a Platner mortar and pestle followed by a few minutes of grinding in a stainless-steel-ball mill (DTN: LL000404551021.134). The distribution of particles sizes obtained after crushing and milling was approximately bimodal, with numerous large (>0.015 cm diameter) fragments and material less than 0.0045 cm, which subsequent SEM examination revealed to be approximately single fuel grains (~0.0020 cm dia.). A relatively small amount of the fuel particles were between ~ 0.0045 cm and 0.015 cm in diameter. No attempt was made to estimate the relative distribution of these three particle sizes during the initial grinding; however, following the sample preparation procedure, in which the largest fragments (>0.0075 cm) were crushed and milled a second time, the final distribution of particle sizes obtained after preparation for the ANL tests given in Table I-1 was achieved.

Table I-1. Final Distribution of Fuel Particle Sizes After All Grinding Cycles (ANL Tests)

Size Fraction (Particle Diameter)	Mass (gram)	Relative Amount*
<0.0045 cm (ave. ~0.0020 cm) (mostly single fuel grains)	2.3252	81%
0.0045 to 0.015 cm	0.3063	11%
>0.015 cm	0.2520	9%

Note: *Total relative amount exceeds 100% due to rounding. Data from DTN: LL000404551021.134.

Several powders of spent UO_2 fuels were prepared for flow-through dissolution studies conducted at Pacific Northwest National Laboratory (PNNL) by crushing and grinding de-clad segments, and the results are reported by Gray and Wilson (1995), who reproduce SEM micrographs of the prepared powders. Gray and Wilson (1995) do not discuss what fraction of the fuel size fraction exceeded that used in the flow-through studies, and it is reasonably concluded here that the distribution is similar to that given in Table I-1. The most important factor illustrated by Gray and Wilson (1995), in terms of understanding the potential distribution of particle sizes produced during a disruptive volcanic event, is that not all fuels prepared by them show identical particle-size distributions. Several fuels display very small particles - on the order of 0.001 cm or less. Although SEM examinations of the ANL fuel grains revealed relatively few particles of ATM-103 fuel with sizes less than single grains, the PNNL results from a wider variety of fuel types necessitates shifting the potential distribution of grain sizes to smaller particle sizes than that estimated from the ATM-103 results alone. Consider here that 0.0001-cm diameter particles represent a reasonable lower limit on particle sizes for all unaltered fuels exposed to a disruptive volcanic event.

I.2.2 Dry-air oxidized fuel

Spent UO_2 fuel that has been oxidized in the absence of moisture may form a series of oxides, with concomitant degradation of the integrity of the fuel meat (Einziger et al. 1992). Oxidation up to a stoichiometry of $UO_{2.4}$ leads to volume reduction of the UO_2 matrix. This can open grain boundaries and may result in the disaggregation of the fuel into single fuel grains (Einziger et al. 1992). Further oxidation to U_3O_8 and related oxides results in a large volume expansion and potentially extreme degradation of the fuel into a powder with particle sizes less than one micrometer in diameter. SEM examination of spent fuel oxidized to approximately U_3O_8 indicates particle sizes of approximately 2.5 micrometers (0.0025 cm dia.) with lower limits of approximately 0.5 micrometers (0.00005 cm dia.) (Gray and Wilson 1995), with larger particles range up to approximately 50 micrometers diameter (0.005 cm) (Table I-2). An estimate of the larger limit on the range of particle sizes is more difficult to make with much certainty. Based on qualitative observations of ATM-103 fuel following preparation for the ANL corrosion studies, an upper limit of 0.0005 cm diameter is chosen (Table I-2).

I.2.3 Aqueous-corroded fuel

SEM examinations of corroded spent fuel following interaction with simulated groundwater at 90°C are reported by Finch et al. (1999). The grain sizes of uranium(VI) alteration products on corroded fuel commonly reach 0.01 cm (Finch et al. 1999); however, considering the physical properties of uranium(VI) compounds (Frondel, 1958), these phases are similar to gypsum or calcite in terms of hardness and fracture toughness. Therefore, a powerful eruptive event will probably fragment nearly all of the larger crystals of secondary U phases, which is why a smaller upper limit of 0.001 cm diameter is chosen for the range of particle sizes for aqueous-corroded fuel (Table I-2). The lower value for the particle-size range is based on the SEM examinations reported in Finch et al. (1999), who demonstrate the extremely fine-grained nature of many alteration products, with crystal dimensions as small as 0.5 micrometers or less (≤0.00005 cm).

Suggested particle-size ranges and average values for particle sizes of light-water-reactor fuels are listed in Table I-2. No firm statistical foundation underlies the averages or ranges listed in Table I-2; however, based on sources (Frondel 1958), these averages are considered appropriate. Limiting values for the ranges are less-well constrained, perhaps, but it is likely 80 to 90 percent of the fuel particles will fall within the ranges reported in Table I-2.

Table I-2. Estimated Fuel-particle Sizes*

Degradation State	Mean (cm dia.)	Range (cm dia.)
unaltered fuel	0.0020	0.0001 to 0.050
oxidized in dry air	0.00025	0.00005 to 0.0005
corroded fuel	0.0002	0.00005 to 0.001

NOTE: * Sizes indicate particle diameters.

Based on our current level of understanding, it seems reasonable to treat both categories of altered fuel (dry-air oxidized and aqueous corroded) almost the same, since their estimated particle sizes are not very different from each other. The altered fuel is substantially more friable than (most) unaltered fuel (Einziger et al. 1992; Finch et al. 1999), with size distributions that may be skewed to quite small sizes.

I.3 OTHER TYPES OF SPENT FUEL

In addition to CSNF, which constitutes the vast majority of the fuel inventory destined for permanent disposal, there are additional fuel types that may exhibit physical properties that are quite distinct from those of CSNF. These "other" spent fuels include those from research reactors, military-use reactors, and other sources. They are highly variable in their physical characteristics, include materials from metals to carbides, and may be in a variety of forms, from ingots to granules. No attempt is made here to estimate potential particle sizes for this broad category of fuel types. Furthermore, there are too few data currently available on the physical properties of these fuels following physical and/or chemical degradation that may occur in the repository following their disposal.

I.4 DEFENSE HIGH-LEVEL WASTE (HLW) GLASS

Whereas HLW glass will constitute a large volume fraction of the total volume of waste in the repository (DOE 1998d), it is not the major contributor to total activity (DOE 1998c). HLW glass is probably best treated in a manner similar to the tuff rock, which also consists of a large volume of glass. Similarly, an intrusive, rapidly cooling magma is likely going to be glassy as well.

ATTACHMENT II—
RADIONUCLIDE INVENTORY FOR FINAL WASTE FORMS

II.1 RADIONUCLIDE INVENTORY FOR FINAL WASTE FORMS

The masses of the radionuclides to be disposed of in the Yucca Mountain repository were estimated in Sanchez et al. (1998). These values were calculated with use of radioactivity values from CRWMS M&O (1998f) and are presented below in Table A-1. The significant findings of this table are:

- 95.6% of the radionuclide mass to be disposed in the Yucca Mountain repository is anticipated to be due to CSNF. The remaining 4.4% of the mass will come from DSNF and HLW.

- The four radionuclides that dominate the total mass inventory are: ^{238}U, ^{235}U, ^{239}Pu, and ^{236}U. Of these, ^{238}U is the main contributor with 63.9 million kilograms of mass. This corresponds to nearly 97% of the total mass from all the radionuclides to be disposed in the repository.

- When ranking DOE-owned materials by themselves, ^{232}Th is third in its ranking behind ^{238}U and ^{235}U. This radionuclide, however, is not significant to total inventory. Its presence is due to the existence of Thorium fuel-cycle materials that are unique to DOE fuel.

Table II-1. YMP-Scale Source Term Mass Inventory (Calendar Year = 2035, Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Mass Inventory (b)				
	DOE-Owned		Commercial		Total
	SNF	HLW	PWR	BWR	
	(kg) (%)	(kg) (%)	(kg) (%)	(kg) (%)	(kg)
^{227}Ac	4.24E-04 (8.92E+01)	3.91E-05 (8.23E+00)	8.39E-06 (1.76E+00)	3.73E-06 (7.85E-01)	0.000476
^{108m}Ag †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0
^{241}Am	3.09E+02 (5.07E-01)	1.19E+02 (1.94E-01)	4.14E+04 (6.79E+01)	1.92E+04 (3.14E+01)	61000
^{242m}Am	8.27E-02 (5.68E-02)	9.25E-03 (6.35E-03)	9.64E+01 (6.62E+01)	4.92E+01 (3.37E+01)	146
^{243}Am	9.08E+00 (1.08E-01)	8.68E-01 (1.03E-02)	5.87E+03 (6.96E+01)	2.55E+03 (3.03E+01)	8430
^{14}C	2.06E-01 (9.76E-01)	0.00E+00 (0.00E+00)	1.33E+01 (6.32E+01)	7.54E+00 (3.58E+01)	21.1
^{36}Cl	1.32E-01 (5.86E-01)	0.00E+00 (0.00E+00)	1.50E+01 (6.66E+01)	7.39E+00 (3.28E+01)	22.5
^{243}Cm †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0
^{244}Cm	9.54E-01 (6.75E-02)	6.63E-01 (4.69E-02)	1.01E+03 (7.16E+01)	3.99E+02 (2.83E+01)	1410
^{245}Cm	1.77E-02 (1.34E-02)	1.54E-03 (1.16E-03)	9.79E+01 (7.38E+01)	3.47E+01 (2.62E+01)	133
^{246}Cm	1.68E-02 (1.09E-01)	9.80E-05 (6.38E-04)	1.16E+01 (7.53E+01)	3.78E+00 (2.46E+01)	15.4
^{135}Cs	1.77E+02 (5.74E-01)	4.70E+02 (1.52E+00)	2.07E+04 (6.72E+01)	9.46E+03 (3.07E+01)	30800
^{137}Cs	3.94E+02 (7.11E-01)	7.13E+02 (1.29E+00)	3.84E+04 (6.92E+01)	1.60E+04 (2.88E+01)	55400
^{129}I	9.28E+01 (7.02E-01)	5.06E-02 (3.83E-04)	9.23E+03 (6.98E+01)	3.90E+03 (2.95E+01)	13200
^{93}Mo †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0
^{93m}Nb ‡	1.17E-07 (5.37E-01)	5.28E-07 (2.43E+00)	1.41E-05 (6.50E+01)	6.96E-06 (3.21E+01)	0.0000217
^{94}Nb	5.52E-02 (1.87E-02)	7.46E-04 (2.53E-04)	2.84E+02 (9.63E+01)	1.09E+01 (3.71E+00)	295
^{59}Ni	6.04E+00 (2.87E-01)	1.67E+00 (7.95E-02)	1.58E+03 (7.51E+01)	5.17E+02 (2.45E+01)	2110
^{63}Ni	6.49E+00 (1.75E+00)	0.00E+00 (0.00E+00)	2.79E+02 (7.50E+01)	8.62E+01 (2.32E+01)	371
^{237}Np	2.30E+02 (5.55E-01)	1.89E+02 (4.55E-01)	3.01E+04 (7.26E+01)	1.10E+04 (2.64E+01)	41500
^{231}Pa	1.84E+00 (9.29E+01)	9.70E-02 (4.91E+00)	2.94E-02 (1.49E+00)	1.31E-02 (6.65E-01)	1.98
^{210}Pb	1.03E-07 (3.15E+01)	1.68E-09 (5.11E-01)	1.53E-07 (4.67E+01)	6.97E-08 (2.12E+01)	0.000000328
^{107}Pd	5.52E+01 (3.33E-01)	0.00E+00 (0.00E+00)	1.15E+04 (6.96E+01)	4.97E+03 (3.01E+01)	16500
^{238}Pu	3.51E+01 (2.65E-01)	1.10E+02 (8.29E-01)	9.58E+03 (7.23E+01)	3.52E+03 (2.66E+01)	13300
^{239}Pu	6.98E+03 (1.77E+00)	3.59E+02 (9.08E-02)	2.75E+05 (6.96E+01)	1.13E+05 (2.85E+01)	395000
^{240}Pu	1.38E+03 (8.76E-01)	6.80E+01 (4.33E-02)	1.10E+05 (6.98E+01)	4.61E+04 (2.93E+01)	157000
^{241}Pu	6.02E+01 (1.67E-01)	6.76E+00 (1.88E-02)	2.46E+04 (6.83E+01)	1.14E+04 (3.16E+01)	36000
^{242}Pu	6.73E+01 (1.90E-01)	6.18E+00 (1.74E-02)	2.41E+04 (6.81E+01)	1.12E+04 (3.17E+01)	35400
^{226}Ra	7.97E-06 (8.73E+00)	4.46E-07 (4.89E-01)	5.71E-05 (6.25E+01)	2.58E-05 (2.83E+01)	0.0000913
^{228}Ra ‡	3.63E-05 (1.00E+02)	0.00E+00 (0.00E+00)	3.77E-11 (1.04E-04)	1.58E-11 (4.34E-05)	0.0000363
^{79}Se	3.40E+00 (7.77E-01)	6.19E+00 (1.42E+00)	3.03E+02 (6.91E+01)	1.25E+02 (2.87E+01)	438
^{151}Sm	1.93E+01 (1.93E+00)	0.00E+00 (0.00E+00)	6.88E+02 (6.88E+01)	2.93E+02 (2.93E+01)	1000
^{121m}Sn †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0
^{126}Sn	1.11E+01 (5.46E-01)	0.00E+00 (0.00E+00)	1.43E+03 (7.02E+01)	5.95E+02 (2.93E+01)	2030
^{90}Sr	2.12E+02 (8.80E-01)	3.13E+02 (1.30E+00)	1.68E+04 (6.95E+01)	6.85E+03 (2.84E+01)	24100
^{99}Tc	4.36E+02 (7.62E-01)	9.13E+02 (1.60E+00)	3.92E+04 (6.86E+01)	1.66E+04 (2.91E+01)	57200
^{229}Th	1.28E-01 (9.97E+01)	3.34E-04 (2.59E-01)	6.96E-05 (5.40E-02)	2.61E-05 (2.03E-02)	0.129
^{230}Th	7.23E-02 (8.12E+00)	2.89E-03 (3.24E-01)	5.65E-01 (6.34E+01)	2.51E-01 (2.81E+01)	0.891
^{232}Th ‡	7.94E+04 (9.46E+01)	4.51E+03 (5.37E+00)	1.32E-01 (1.58E-04)	5.47E-02 (6.52E-05)	83900

Table II-1. YMP-Scale Source Term Mass Inventory (Calendar Year = 2035, Time = 0 yr) [(a)] (Continued)

Nuclide ID	YMP Radionuclide Mass Inventory [(b)]				
	DOE-Owned		Commercial		Total
	SNF	HLW	PWR	BWR	
	(kg) (%)	(kg) (%)	(kg) (%)	(kg) (%)	(kg)
^{232}U †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0
^{233}U	1.24E+03 (9.99E+01)	2.84E-01 (2.29E-02)	2.66E-01 (2.14E-02)	9.57E-02 (7.71E-03)	1240
^{234}U	2.67E+02 (1.92E+00)	3.76E+01 (2.70E-01)	9.49E+03 (6.82E+01)	4.13E+03 (2.97E+01)	13900
^{235}U	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	666000
^{236}U	1.15E+04 (3.87E+00)	3.17E+01 (1.07E-02)	2.03E+05 (6.82E+01)	8.30E+04 (2.79E+01)	297000
^{238}U	2.60E+06 (4.06E+00)	5.30E+04 (8.29E-02)	4.02E+07 (6.28E+01)	2.11E+07 (3.30E+01)	63900000
^{93}Zr	6.24E+02 (9.47E-01)	1.31E+03 (1.99E+00)	4.29E+04 (6.52E+01)	2.10E+04 (3.19E+01)	65900
Total	2.84E+06 (4.30E+00) (b)	6.23E+04 (9.44E-02) (c)	4.14E+07 (6.28E+01)	2.16E+07 (3.28E+01)	66000000
	2.90E+06 (4.39%)		6.31E+07 (95.6%)		
Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)					
^{238}U	2.60E+06 (4.06E+00)	5.30E+04 (8.29E-02)	4.02E+07 (6.28E+01)	2.11E+07 (3.30E+01)	63900000
^{235}U	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	666000
^{239}Pu	6.98E+03 (1.77E+00)	3.59E+02 (9.08E-02)	2.75E+05 (6.96E+01)	1.13E+05 (2.85E+01)	395000
^{236}U	1.15E+04 (3.87E+00)	3.17E+01 (1.07E-02)	2.03E+05 (6.82E+01)	8.30E+04 (2.79E+01)	297000
^{240}Pu	1.38E+03 (8.76E-01)	6.80E+01 (4.33E-02)	1.10E+05 (6.98E+01)	4.61E+04 (2.93E+01)	157000
^{232}Th ‡	7.94E+04 (9.46E+01)	4.51E+03 (5.37E+00)	1.32E-01 (1.58E-04)	5.47E-02 (6.52E-05)	83900
^{93}Zr	6.24E+02 (9.47E-01)	1.31E+03 (1.99E+00)	4.29E+04 (6.52E+01)	2.10E+04 (3.19E+01)	65900
^{241}Am	3.09E+02 (5.07E-01)	1.19E+02 (1.94E-01)	4.14E+04 (6.79E+01)	1.92E+04 (3.14E+01)	61000
^{99}Tc	4.36E+02 (7.62E-01)	9.13E+02 (1.60E+00)	3.92E+04 (6.86E+01)	1.66E+04 (2.91E+01)	57200
^{137}Cs	3.94E+02 (7.11E-01)	7.13E+02 (1.29E+00)	3.84E+04 (6.92E+01)	1.60E+04 (2.88E+01)	55400

Table II-1. YMP-Scale Source Term Mass Inventory (Calendar Year = 2035, Time = 0 yr) [(a)] (Continued)

Nuclide ID	YMP Radionuclide Mass Inventory [(b)]				
	DOE-Owned		Commercial		Total
	SNF	HLW	PWR	BWR	
	(kg) (%)	(kg) (%)	(kg) (%)	(kg) (%)	(kg)
Top 10 Radionuclides (Ranked on Inventory for DOE-Owned Wastes Only)					
^{238}U	2.60E+06 (4.06E+00)	5.30E+04 (8.29E-02)	-- --	-- --	2650000
^{235}U	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	-- --	-- --	137000
^{232}Th ‡	7.94E+04 (9.46E+01)	4.51E+03 (5.37E+00)	-- --	-- --	83900
^{236}U	1.15E+04 (3.87E+00)	3.17E+01 (1.07E-02)	-- --	-- --	11500
^{239}Pu	6.98E+03 (1.77E+00)	3.59E+02 (9.08E-02)	-- --	-- --	7340
^{93}Zr	6.24E+02 (9.47E-01)	1.31E+03 (1.99E+00)	-- --	-- --	1930
^{240}Pu	1.38E+03 (8.76E-01)	6.80E+01 (4.33E-02)	-- --	-- --	1450
^{99}Tc	4.36E+02 (7.62E-01)	9.13E+02 (1.60E+00)	-- --	-- --	1350
^{233}U	1.24E+03 (9.99E+01)	2.84E-01 (2.29E-02)	-- --	-- --	1240
^{137}Cs	3.94E+02 (7.11E-01)	7.13E+02 (1.29E+00)	-- --	-- --	1110
Top 10 Radionuclides (Ranked on Inventory for Commercial SNFs Only)					
^{238}U	-- --	-- --	4.02E+07 (6.28E+01)	2.11E+07 (3.30E+01)	61300000
^{235}U	-- --	-- --	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	529000
^{239}Pu	-- --	-- --	2.75E+05 (6.96E+01)	1.13E+05 (2.85E+01)	388000
^{236}U	-- --	-- --	2.03E+05 (6.82E+01)	8.30E+04 (2.79E+01)	286000
^{240}Pu	-- --	-- --	1.10E+05 (6.98E+01)	4.61E+04 (2.93E+01)	156000
^{93}Zr	-- --	-- --	4.29E+04 (6.52E+01)	2.10E+04 (3.19E+01)	63900
^{241}Am	-- --	-- --	4.14E+04 (6.79E+01)	1.92E+04 (3.14E+01)	60600
^{99}Tc	-- --	-- --	3.92E+04 (6.86E+01)	1.66E+04 (2.91E+01)	55800
^{137}Cs	-- --	-- --	3.84E+04 (6.92E+01)	1.60E+04 (2.88E+01)	54400
^{241}Pu	-- --	-- --	2.46E+04 (6.83E+01)	1.14E+04 (3.16E+01)	36000

† Data Values for radionuclides were previously reported in Wilson et al. (1994).

‡ Data Values for radionuclides were not previously reported in Wilson et al. (1994).

(a) Table after Sanchez et al. (1998), data for spent nuclear fuel (SNF) and high-level waste (HLW) inventory data originally taken from INEEL/PA Parameters Database (DOE 1998d) (values represent intermediate database values, upgraded values can be found in Attachment II). (In total 41 radionuclides are inventoried in the INEEL/PA-DB).

(b) Mass inventory values calculated using half-lives from the Decay Libraries from ORIGEN2 (Croff 1980a; 1980b).

(c) Note, the total DOE-owned mass load (due to radionuclides) is only 2.90E+06 kg. Thus only 4.39% of the total mass load (due to radionuclides) in YMP is due to DOE-Owned inventory.

(d) Note, the total commercial mass load (due to radionuclides) is 6.30E+07 kg. Thus 95.6% of the total mass load (due to radionuclides) in YMP is due to Commercial inventory.

ATTACHMENT III—
ALPHA-RECOIL MECHANICS

The alpha recoil is analyzed per conservation of momentum in the center-of-mass (COM) frame of reference. Consider Figure III-1, a radionuclide X (e.g., ^{238}U), at rest in the lab-system frame of reference (and also in the COM frame), undergoes an alpha decay. The radionuclide X decays to radionuclide Y (e.g., ^{234}Th) by emitting an alpha particle, (see Figure III-2 for COM frame).

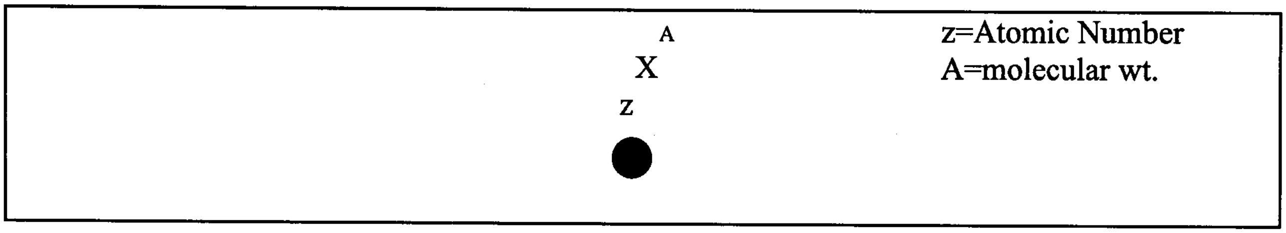


Figure III-1. Initial Center-of-Mass (COM) Frame Conditions for Alpha Emitting Radionuclide

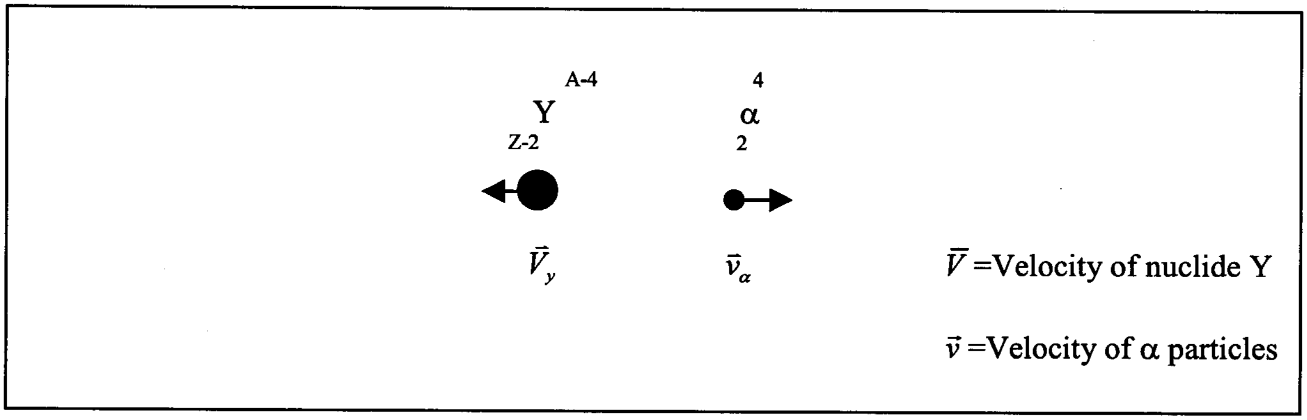


Figure III-2. Final Center-of-Mass (COM) Frame Conditions after Alpha Decay Resulting in Two Decay Products, an Alpha Particle and the Decay Daughter

Applying the conservation of momentum in the COM system leads to Equations III-1 and III-2.

$$\begin{array}{c} \text{Momentum} \\ \text{Before} \end{array} = \begin{array}{c} \text{Momentum} \\ \text{After} \end{array} \qquad \text{(Eq. III-1)}$$

$$0 = M_y\vec{V}_y + m_\alpha\vec{v}_\alpha \qquad \text{(Eq. III-2)}$$

where,

M_y = Mass Nuclide Y
m_α = Mass of α particles

Thus the velocity of the recoil nucleus, in terms of the velocity of the alpha particle, is given by Equation III-3:

$$V_y = -v_\alpha \left(\frac{m_\alpha}{M_y} \right) \qquad \text{(Eq. III-3)}$$

where

m_α = mass of the alpha particle
M_y = mass of the recoil nucleus.

The kinetic energy (KE) of the recoil nucleus can now be determined in terms of the kinetic energy of the emitted alpha particle, see EquationIII-4:

$$KE(Y) = \frac{1}{2} M_Y V_Y^2 = \frac{m_\alpha}{M_Y} \left(\frac{1}{2} m_\alpha V_\alpha^2 \right) = \frac{m_\alpha}{M_Y} KE(\alpha) \qquad \text{(Eq. III-4)}$$

Thus the kinetic energy of the recoil nucleus is a small fraction of that given to the alpha particle. The energy of the alpha particle is dependent upon the proper mass defect value, the amount of mass converted into energy. For the case of ^{238}U decaying to ^{234}Th, there is: $KE(\alpha) = 4.196$ MeV (Lederer and Shirley, 1978), $m_\alpha \cong 4.0$, $M_{Th} \cong 234.0$. This results in a kinetic energy value of 0.072 MeV for the recoil nucleus.

It is important to note that: 1) the velocity of the recoil nucleus is in the opposite direction of the alpha particle, and 2) the velocities of both the alpha and recoil nucleus are not a function of direction (i.e. they are isotropic in direction in the COM frame). Because the COM is not moving with respect to the lab-system frame, the velocities are also isotropic in the lab-system frame. Thus, for any recoil nucleus moving in a given direction (towards the subsurface groundwater, etc.) there is an equal probability that another recoil nucleus is moving in the opposite direction with the same speed. This isotropy means that only one half of the recoil nuclei are initially moving away from the fuel and into the groundwater.

ATTACHMENT IV— RADIONUCLIDE INVENTORY FOR PRESSURIZED-WATER REACTOR SPENT NUCLEAR FUELS AND BOILING-WATER REACTOR SPENT NUCLEAR FUELS

This attachment contains time-dependent radionuclide information for CSNF with average burnup histories. CSNF dominates the projected waste inventory for the Yucca Mountain Project (YMP) repository (only up to ten percent by mass of the YMP inventory is allocated for DSNF) (CRWMS M&O 2000j). CSNF has considerably higher burnup than that expected for DSNF (e.g., DOE N-reactor fuels typically have less than one-twentieth of the burnup of CSNF). Table IV-1 presents radionuclide inventories for pressurized-water reactors (PWRs) at various decay times (time after burnup). Table IV-2 presents radionuclide inventories for boiling-water reactors (BWRs) at the same decay times used in Table IV-1. The data for both tables was obtained from the *PWR Source Term Generation and Evaluation* (CRWMS M&O 1999k) and *BWR Source Term Generation and Evaluation* (CRWMS M&O 1999j). In each table only a limited portion of the original data (timeframes from 125 years up to 10,025 years) is reproduced. The only additions to the tables are (1) presentation of decay mode and half-lives for the radionuclides in columns 2 and 3, and (2) the generation of the total radionuclide inventories at the bottom of each table. The decay modes and the half-lives were obtained from Parrington et al. (1996), which demonstrates qualitatively that a significant portion of the radionuclides in the expected YMP inventory are alpha emitters. For simplicity, the net results presented in Table IV-3 present bounding calculations. That is, the calculations consider 100% of the radionuclides to be alpha emitters, which are the worst-case radiolysis generators. The only numerical values from these tables that are used in Table IV-3 are those for the total radionuclide inventory for the PWR and BWR fuel assemblies (see bottom on Tables IV-1 and IV-2). As can be identified when comparing the final results in Table IV-3 (corrosion rates due to alpha radiolysis) to the dissolution rates in Figure 6, the radiolysis rates are very insignificant. This means that if future estimates for the quantities of radionuclides in Tables IV-1 and IV-2 result in slightly different inventories, their differences will not significantly impact the screening argument for the FEP, YMP 2.1.13.02.00, Radiation, Damage in Waste and EBS.

Note: decay times listed in Tables IV-1 and IV-2 include the 25-year decay time of the fuels prior to its emplacement into the YMP repository. Thus, fuel with a date of 525 years corresponds to fuel that has been stored for 500 years (calendar year 2535) in the repository.

Table IV-1. Yucca Mountain Project Radionuclide Inventory for Pressurized-water Reactor Fuels

Nuclide ID	YMP/PWR Radionuclide Inventory (Data from *PWR Source Term Generation and Evaluation,* CRWMS M&O 1999k)					
	Decay Mode(a)	Half-Life(b) (yr)	Radioactivity Inventory [Ci/Assembly]			
			125 yr	225 yr	325 yr	425 yr
Ac227	α,β^-,γ	21.77 a	4.00E-05	5.59E-05	7.14E-05	8.70E-05
Ag108	$\beta^-,\gamma,\varepsilon\beta^+$	2.39m	2.23E-04	1.29E-04	7.48E-05	4.34E-05
Ag108m	$\gamma,\varepsilon\beta^+,ITe^-$	130. a	2.56E-03	1.48E-03	8.60E-04	4.98E-04
Am241	α,γ,SF	432.7 a	2.40E+03	2.05E+03	1.75E+03	1.49E+03
Am242	$\beta^-,\gamma,\varepsilon e^-$	16.02 h	3.89E+00	2.38E+00	1.45E+00	8.90E-01
Am242m	α,γ,ITe^-,SF	141. a	3.91E+00	2.39E+00	1.46E+00	8.94E-01
Am243	α,γ,SF	7370 a	2.18E+01	2.16E+01	2.14E+01	2.12E+01
Ar39	β^-	269 a	4.04E-05	3.12E-05	2.41E-05	1.86E-05
Ba137m	IT	2.552 m	3.85E+03	3.82E+02	3.79E+01	3.76E+00
Bi211	α,β^-,γ	2.14 m	4.00E-05	5.59E-05	7.14E-05	8.70E-05
Bi212	α,β^-,γ	1.009 h	7.81E-03	2.89E-03	1.07E-03	3.98E-04
C14	β^-	5730 a	3.28E-01	3.24E-01	3.21E-01	3.17E-01
Ca41	ε	1.03E5 a	9.24E-05	9.24E-05	9.23E-05	9.22E-05
Cd113m	β^-,IT	14.1 a	5.61E-02	4.12E-04	3.02E-06	2.21E-08
Ce142	β^-,γ	Stable	1.86E-05	1.86E-05	1.86E-05	1.86E-05
Ce144	α,γ,SF	284.6 d	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cf249	α,γ,SF	351 a	6.34E-05	5.20E-05	4.27E-05	3.50E-05
Cf250	$\beta^-,\beta^+,\varepsilon$	13.1 a	4.20E-07	2.10E-09	1.06E-11	1.55E-13
Cl36	α,γ,SF	3.01E5 a	6.80E-03	6.80E-03	6.80E-03	6.79E-03
Cm242	$\alpha,\gamma,SF,\varepsilon$	1.2E3 a	3.22E+00	1.97E+00	1.20E+00	7.36E-01
Cm243	α,γ,SF	162.8 d	9.01E-01	7.91E-02	6.95E-03	6.10E-04
Cm244	α,γ,SF	29.1 a	2.96E+01	6.42E-01	1.40E-02	3.03E-04
Cm245	α,γ,SF	18.1 a	3.04E-01	3.02E-01	2.99E-01	2.97E-01
Cm246	β^-,γ	8.5E3 a	1.03E-01	1.01E-01	9.98E-02	9.83E-02
Co60	$\beta^-,\gamma,\varepsilon$	4.76E3 a	6.06E-04	1.17E-09	2.27E-15	4.40E-21
Cs134	β^-	5.271 a	6.33E-14	1.59E-28	0.00E+00	0.00E+00
Cs135	β^-,γ	2.065 a	3.50E-01	3.50E-01	3.50E-01	3.50E-01
Cs137	γ,ε	2.3E6 a	4.08E+03	4.05E+02	4.01E+01	3.98E+00
Eu150	$\beta^-,\gamma,\varepsilon\beta^+$	36. a	2.99E-06	4.31E-07	6.23E-08	8.98E-09
Eu152	$\beta^-,\gamma,\varepsilon\gamma$	13.48 a	7.23E-03	3.98E-05	2.20E-07	1.21E-09
Eu154	β^-,γ	8.59 a	2.10E-01	6.57E-05	2.06E-08	6.44E-12
Eu155	ε	4.71 a	1.90E-05	7.04E-12	2.60E-18	9.61E-25
Fe55	β^-	2.73 a	3.25E-11	3.04E-22	0.00E+00	0.00E+00
H3	β^-,γ	12.3 a	4.10E-01	1.48E-03	5.37E-06	1.94E-08
Ho166m	β^-,γ	1.2E3 a	5.08E-04	4.79E-04	4.52E-04	4.27E-04
I129	β^-,γ	1.57E7 a	2.19E-02	2.19E-02	2.19E-02	2.19E-02
Kr85	γ,ε,e^-	10.73 a	1.76E+00	2.74E-03	4.26E-06	6.63E-09
Mo93	β^+,ε	~3.5E3 a	4.06E-02	3.98E-02	3.90E-02	3.83E-02
Nb91	ITe^-	7E2 a	1.77E-05	1.60E-05	1.44E-05	1.30E-05
Nb93m	β^-,γ	16.1 a	1.09E+00	9.29E-01	9.27E-01	9.26E-01
Nb94	ε	2.0E4 a	8.36E-01	8.33E-01	8.30E-01	8.28E-01
Ni59	β^-	7.6e4 a	2.09E+00	2.09E+00	2.08E+00	2.08E+00
Ni63	α,γ	100. a	1.26E+02	6.30E+01	3.15E+01	1.58E+01
Np237	β^-,γ	2.14E6 a	3.26E-01	3.98E-01	4.60E-01	5.12E-01
Np238		2.117 d	1.76E-02	1.08E-02	6.58E-03	4.02E-03

Table IV-1. Yucca Mountain Project Radionuclide Inventory for Pressurized-water Reactor Fuels (Continued)

Nuclide ID	YMP/PWR Radionuclide Inventory (Data from *PWR Source Term Generation and Evaluation*, CRWMS M&O 1999k)					
	Decay Mode(a)	Half-Life(b) (yr)	Radioactivity Inventory [Ci/Assembly]			
			125 yr	225 yr	325 yr	425 yr
Np237	β^-,γ	1.2E3 a	2.18E+01	2.16E+01	2.14E+01	2.12E+01
Pa231	α,γ	3.28E4 a	4.53E-05	6.08E-05	7.63E-05	9.18E-05
Pa233	β^-,γ	27.0 d	3.26E-01	3.98E-01	4.60E-01	5.12E-01
Pa234	β^-,γ	6.69 h	1.92E-04	1.92E-04	1.92E-04	1.92E-04
Pa234m	$\beta^-,\gamma,\mathit{IT}$	1.17 m	1.48E-01	1.48E-01	1.48E-01	1.48E-01
Pb211	β^-,γ	36.1 m	4.00E-05	5.59E-05	7.14E-05	8.70E-05
Pb212	β^-,γ	10.64 h	7.81E-03	2.89E-03	1.07E-03	3.98E-04
Pd107	β^-	6.5E6 a	8.41E-02	8.41E-02	8.41E-02	8.41E-02
Pm145	$\alpha,\gamma,\varepsilon$	17.7 a	2.13E-05	4.24E-07	8.45E-09	1.68E-10
Pm146	$\beta^-,\gamma,\varepsilon\gamma$	5.53 a	4.22E-07	1.52E-12	5.45E-18	1.96E-23
Pm147	β^-,γ	2.6234 a	3.99E-10	1.33E-21	0.00E+00	0.00E+00
Po212	α	0.298 μs	5.00E-03	1.85E-03	6.87E-04	2.55E-04
Po215	α,β^-,γ	1.780 ms	4.00E-05	5.59E-05	7.14E-05	8.70E-05
Po216	α,γ	0.145 s	7.81E-03	2.89E-03	1.07E-03	3.98E-04
Pr144	β^-,γ	17.28 m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu236	$\alpha,\gamma,\mathit{SF}$	2.87 a	8.86E-07	8.85E-07	8.85E-07	8.84E-07
Pu238	$\alpha,\gamma,\mathit{SF}$	87.7 a	1.04E+03	4.73E+02	2.16E+02	9.83E+01
Pu239	$\alpha,\gamma,\mathit{SF}$	2.410E4	1.77E+02	1.77E+02	1.76E+02	1.76E+02
Pu240	$\alpha,\gamma,\mathit{SF}$	6.56E3	3.18E+02	3.15E+02	3.12E+02	3.09E+02
Pu241	α,β^-,γ	14.4 a	1.97E+02	1.87E+00	3.12E-01	2.97E-01
Pu242	$\alpha,\gamma,\mathit{SF}$	3.75E5 a	1.64E+00	1.64E+00	1.64E+00	1.64E+00
Ra223	α,γ	11.435 d	4.00E-05	5.59E-05	7.14E-05	8.70E-05
Ra224	α,γ	3.66 d	7.81E-03	2.89E-03	1.07E-03	3.98E-04
Rb87	β^-	4.88E10 a	1.39E-05	1.39E-05	1.39E-05	1.39E-05
Rh102	$\beta^-,\gamma,\varepsilon\beta^+$	207. d	1.26E-13	5.26E-24	0.00E+00	0.00E+00
Rh106	β^-,γ	29.9 s	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rn219	α,γ	3.96 s	4.00E-05	5.59E-05	7.14E-05	8.70E-05
Rn220	α,γ	55.6 s	7.81E-03	2.89E-03	1.07E-03	3.98E-04
Ru106	β^-	1.02 a	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb125	β^-,γ	2.758 a	9.11E-11	8.53E-22	0.00E+00	0.00E+00
Sb126	β^-,γ	12.4 d	5.39E-02	5.38E-02	5.38E-02	5.38E-02
Sb126m	γ,ITe-	11. s	3.85E-01	3.85E-01	3.84E-01	3.84E-01
Se79	β^-	6.5E4 a	4.57E-02	4.57E-02	4.57E-02	4.57E-02
Sm151	β^-,γ	90 a	9.75E+01	4.51E+01	2.09E+01	9.67E+00
Sn121	β^-	1.128 d	3.51E-01	9.95E-02	2.82E-02	8.00E-03
Sn121m	$\beta^-,\gamma,\mathit{ITe}^-$	55 a	4.52E-01	1.28E-01	3.64E-02	1.03E-02
Sn126	β^-,γ	1.0E5 a	3.85E-01	3.85E-01	3.84E-01	3.84E-01
Sr90	β^-	29.1 a	2.32E+03	1.97E+02	1.68E+01	1.43E+00
Tc99	β^-,γ	2.13E5 a	8.98E+00	8.98E+00	8.98E+00	8.97E+00
Te125m	γ,ITe-	58. d	2.23E-11	2.08E-22	0.00E+00	0.00E+00
Th227	α,β^-,γ	18.72 d	3.95E-05	5.51E-05	7.05E-05	8.58E-05
Th228	α,γ	1.913 a	7.81E-03	2.89E-03	1.07E-03	3.98E-04
Th230	α,β^-,γ	7.54E4 a	1.00E-03	2.14E-03	3.40E-03	4.73E-03
Th231	β^-,γ	1.063 d	7.38E-03	7.40E-03	7.42E-03	7.43E+00
Th234	β^-,γ	24.10 d	1.48E-01	1.48E-01	1.48E-01	1.48E-01

Table IV-1. Yucca Mountain Project Radionuclide Inventory for Pressurized-water Reactor Fuels (Continued)

Nuclide ID	YMP/PWR Radionuclide Inventory (Data from *PWR Source Term Generation and Evaluation*, CRWMS M&O 1999k)					
	Decay Mode[a]	Half-Life[b] (yr)	Radioactivity Inventory [Ci/Assembly]			
			125 yr	225 yr	325 yr	425 yr
Tl207	β^-,γ	4.77 m	3.99E-05	5.57E-05	7.12E-05	8.67E-05
Tl208	β^-,γ	3.053 m	2.81E-03	1.04E-03	3.85E-04	1.43E-04
U232	α,γ,SF	70 a	7.59E-03	2.81E-03	1.04E-03	3.87E-04
U233	α,γ,SF	1.592E5 a	1.62E-04	3.21E-04	5.08E-04	7.19E-04
U234	α,γ,SF	2.46E5 a	1.12E+00	1.33E+00	1.42E+00	1.46E+00
U235	α,γ,SF	7.04E8 a	7.38E-03	7.40E-03	7.42E-03	7.43E-03
U236	α,γ,SF	2.342E7 a	1.73E-01	1.74E-01	1.74E-01	1.75E-01
U237	β^-,γ	6.75 d	4.71E-03	4.48E-05	7.46E-06	7.11E-06
U238	α,γ,SF	4.47E9 a	1.48E-01	1.48E-01	1.48E-01	1.48E-01
Y90	β^-,γ	2.67 d	2.32E+03	1.97E+02	1.68E+01	1.43E+00
Zr93	β^-,γ	1.5E6 a	8.94E-01	8.94E-01	8.94E-01	8.94E-01
		Total[c] =	1.70E+04	4.38E+03	2.69E+03	2.18E+03

NOTES:

(a) Decay modes identified from Parrington et al. (1996).

(b) Half-lives identified from Parrington et al. (1996).

(c) Total radionuclide activities are calculated here and are not part of original reference.

Table IV-1. Yucca Mountain Project Radionuclide Inventory for Pressurized-water Reactor Fuels (Continued)

Nuclide ID	YMP/PWR Radionuclide Inventory (Data from *PWR Source Term Generation and Evaluation*, CRWMS M&O 1999k)				
	Radioactivity Inventory [Ci/Assembly]				
	525 yr	1,025 yr	2,025 yr	5,025 yr	10,025 yr
Ac227	1.03E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Ag108	2.51E-05	1.64E-06	6.99E-09	5.41E-16	7.61E-28
Ag108m	2.89E-04	1.88E-05	8.03E-08	6.22E-15	8.74E-27
Am241	1.27E+03	5.69E+02	1.15E+02	1.15E+00	1.36E-01
Am242	5.44E-01	4.66E-02	3.42E-04	1.34E-10	2.84E-21
Am242m	5.47E-01	4.68E-02	3.43E-04	1.35E-10	2.86E-21
Am243	2.10E+01	2.00E+01	1.82E+01	1.37E+01	8.57E+00
Ar39	1.44E-05	3.97E-06	3.02E-07	1.32E-10	3.36E-16
Ba137m	3.73E-01	3.58E-06	3.30E-16	0.00E+00	0.00E+00
Bi211	1.03E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Bi212	1.48E-04	1.91E-06	8.94E-07	9.08E-07	9.37E-07
C14	3.13E-01	2.95E-01	2.61E-01	1.82E-01	9.91E-02
Ca41	9.22E-05	9.19E-05	9.12E-05	8.94E-05	8.65E-05
Cd113m	1.62E-10	3.43E-21	0.00E+00	0.00E+00	0.00E+00
Ce142	1.86E-05	1.86E-05	1.86E-05	1.86E-05	1.86E-05
Ce144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cf249	2.87E-05	1.07E-05	1.48E-06	3.92E-09	1.99E-13
Cf250	1.03E-13	1.01E-13	9.67E-14	8.58E-14	7.03E-14
Cl36	6.79E-03	6.79E-03	6.77E-03	0.00672	0.00665
Cm242	4.50E-01	3.85E-02	2.82E-04	1.12E-10	2.36E-21
Cm243	5.36E-05	2.80E-10	7.66E-21	0.00E+00	0.00E+00
Cm244	6.58E-06	3.18E-14	7.30E-31	0.00E+00	0.00E+00
Cm245	2.94E-01	2.83E-01	2.60E-01	2.04E-01	1.36E-01
Cm246	9.69E-02	9.01E-02	7.78E-02	5.01E-02	2.41E-02
Co60	8.52E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs135	3.50E-01	3.50E-01	3.50E-01	3.49E-01	3.49E-01
Cs137	3.95E-01	3.79E-06	3.49E-16	0.00E+00	0.00E+00
Eu150	1.30E-09	8.11E-14	3.17E-22	0.00E+00	0.00E+00
Eu152	6.67E-12	3.39E-23	0.00E+00	0.00E+00	0.00E+00
Eu154	2.01E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu155	3.65E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
H3	7.02E-11	4.35E-23	0.00E+00	0.00E+00	0.00E+00
Ho166m	4.03E-04	3.02E-04	1.69E-04	2.99E-05	1.67E-06
I129	2.19E-02	2.19E-02	2.19E-02	2.19E-02	2.19E-02
Kr85	1.03E-11	9.36E-26	0.00E+00	0.00E+00	0.00E+00
Mo93	3.75E-02	3.40E-02	2.79E-02	1.54E-02	5.71E-03
Nb91	1.18E-05	7.06E-06	2.55E-06	1.20E-07	7.32E-10
Nb93m	9.25E-01	9.22E-01	9.16E-01	9.05E-01	8.95E-01
Nb94	8.25E-01	8.11E-01	7.84E-01	7.07E-01	5.96E-01
Ni59	2.08E+00	2.07E+00	2.05E+00	2.00E+00	1.90E+00
Ni63	7.89E+00	2.47E-01	2.43E-04	2.31E-13	2.12E-28
Np237	5.57E-01	6.98E-01	7.90E-01	8.12E-01	8.11E-01
Np238	2.46E-03	2.11E-04	1.54E-06	6.08E-13	1.29E-23
Np239	2.10E+01	2.00E+01	1.82E+01	1.37E+01	8.57E+00
Pa231	1.07E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Pa233	5.57E-01	6.98E-01	7.90E-01	8.12E-01	8.11E-01
Pa234	1.92E-04	1.92E-04	1.92E-04	1.92E-04	1.92E-04
Pa234m	1.48E-01	1.48E-01	1.48E-01	1.48E-01	1.48E-01
Pb211	1.03E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Pb212	1.48E-04	1.91E-06	8.94E-07	9.08E-07	9.37E-07

Table IV-1. Yucca Mountain Project Radionuclide Inventory for Pressurized-water Reactor Fuels (Continued)

Nuclide ID	YMP/PWR Radionuclide Inventory (Data from *PWR Source Term Generation and Evaluation*, CRWMS M&O 1999k)				
	Radioactivity Inventory [Ci/Assembly]				
	525 yr	1,025 yr	2,025 yr	5,025 yr	10,025 yr
Pd107	8.41E-02	8.41E-02	8.41E-02	8.41E-02	8.40E-02
Pm145	3.35E-12	1.05E-20	0.00E+00	0.00E+00	0.00E+00
Pm146	7.03E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Po212	9.47E-05	1.22E-06	5.73E-07	5.81E-07	6.00E-07
Po215	1.03E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Po216	1.48E-04	1.91E-06	8.94E-07	9.08E-07	9.37E-07
Pr144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu236	8.84E-07	8.81E-07	8.76E-07	8.6E-07	8.35E-07
Pu238	4.49E+01	9.43E-01	6.33E-04	2.97E-10	6.27E-21
Pu239	1.75E+02	1.73E+02	1.69E+02	1.56E+02	1.37E+02
Pu240	3.05E+02	2.90E+02	2.61E+02	1.90E+02	1.12E+02
Pu241	2.95E-01	2.83E-01	2.61E-01	2.04E-01	1.36E-01
Pu242	1.64E+00	1.64E+00	1.63E+00	1.62E+00	1.61E+00
Pa223	1.03E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Ra224	1.48E-04	1.91E-06	8.94E-07	9.08E-07	9.37E-07
Rb87	1.39E-05	1.39E-05	1.39E-05	1.39E-05	1.39E-05
Rh102	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rn219	1.03E-04	1.85E-04	3.41E-04	8.09E-04	1.59E-03
Rn220	1.48E-04	1.91E-06	8.94E-07	9.08E-07	9.37E-07
Ru106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb125	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb126	5.37E-02	5.35E-02	5.32E-02	5.21E-02	5.03E-02
Sb126m	3.84E-01	3.82E-01	3.80E-01	3.72E-01	3.59E-01
Se79	4.56E-02	4.56E-02	4.55E-02	4.52E-02	4.47E-02
Sm151	4.48E+00	9.52E-02	4.30E-05	3.96E-15	6.85E-32
Sn121	2.27E-03	4.16E-06	1.40E-11	5.36E-28	0.00E+00
Sn121m	2.92E-03	5.36E-06	1.81E-11	6.90E-28	0.00E+00
Sn126	3.84E-01	3.82E-01	3.80E-01	3.72E-01	3.59E-01
Sr90	1.22E-01	5.48E-07	1.11E-17	0.00E+00	0.00E+00
Tc99	8.97E+00	8.96E+00	8.93E+00	8.84E+00	8.69E+00
Te125m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Th227	1.01E-04	1.83E-04	3.36E-04	7.98E-04	1.57E-03
Th228	1.48E-04	1.91E-06	8.94E-07	9.08E-07	9.37E-07
Th230	6.08E-03	1.29E-02	2.64E-02	6.61E-02	1.29E-01
Th231	7.45E-03	7.54E-03	7.71E-03	8.19E-03	8.91E-03
Th234	1.48E-01	1.48E-01	1.48E-01	1.48E-01	1.48E-01

Table IV-1. Yucca Mountain Project Radionuclide Inventory for Pressurized-Water Reactor Fuels (Continued)

Nuclide ID	YMP/PWR Radionuclide Inventory (Data from *PWR Source Term Generation and Evaluation*, CRWMS M&O 1999k)				
	Radioactivity Inventory [Ci/Assembly]				
	525 yr	1,025 yr	2,025 yr	5,025 yr	10,025 yr
Tl207	1.02E-04	1.85E-04	3.40E-04	8.07E-04	1.58E-03
Tl208	5.32E-05	6.86E-07	3.21E-07	3.26E-07	3.37E-07
U232	1.44E-04	1.87E-06	8.76E-07	8.60E-07	8.35E-07
U233	9.52E-04	2.33E-03	5.61E-03	1.60E-02	3.32E-02
U234	1.48E+00	1.49E+00	1.49E+00	1.48E+00	1.46E+00
U235	7.45E-03	7.54E-03	7.71E-03	8.19E-03	8.91E-03
U236	1.76E-01	1.81E-01	1.89E-01	2.09E-01	2.30E-01
U237	7.05E-06	6.77E-06	6.24E-06	4.88E-06	3.25E-06
U238	1.48E-01	1.48E-01	1.48E-01	1.48E-01	1.48E-01
Y90	1.22E-01	5.48E-07	1.11E-17	0.00E+00	0.00E+00
Zr93	8.94E-01	8.94E-01	8.93E-01	8.92E-01	8.90E-01
Total (c) =	1.87E+03	1.09E+03	6.03E+02	3.95E+02	2.86E+02

NOTES:

(a) Decay modes identified from Parrington et al. 1996.

(b) Half-lives identified from Parrington et al. 1996.

(c) Total radionuclide activities are calculated here and are not part of original reference.

Table IV-2. Yucca Mountain Project Radionuclide Inventory for Boiling-Water Reactor Fuels

Nuclide ID	YMP/BWR Radionuclide Inventory (Data from *BWR Source Term Generation and Evaluation*, CRWMS M&O 1999j)					
	Decay Mode[a]	Half-Life[b] (yr)	Radioactivity Inventory [Ci/Assembly]			
			125 yr	225 yr	325 yr	425 yr
Ag108	$\beta^-,\gamma,\varepsilon\beta^+$	2.39m	1.93E-04	1.12E-04	6.48E-05	3.76E-05
Ag108m	$\gamma,\varepsilon\beta^+$,*ITe*$^-$	130. a	2.22E-03	1.29E-03	7.46E-04	4.32E-04
Am241	α,γ,*SF*	432.7 a	6.72E+02	5.74E+02	4.89E+02	4.17E+02
Am242	$\beta^-,\gamma,\varepsilon e^-$	16.02 h	1.32E+00	8.07E-01	4.94E-01	3.02E-01
Am242m	α,γ,*ITe*$^-$,*SF*	141. a	1.33E+00	8.10E-01	4.96E-01	3.03E-01
Am243	α,γ,*SF*	7370 a	5.29E+00	5.24E+00	5.19E+00	5.14E+00
Ar39	β^-	269 a	1.24E-05	9.59E-06	7.41E-06	5.73E-06
Ba137m	*IT*	2.552 m	1.30E+03	1.29E+02	1.28E+01	1.27E+00
Bi212	α,β^-,γ	1.009 h	1.77E-03	6.55E-04	2.43E-04	9.01E-05
C14	β^-	5730 a	1.73E-01	1.71E-01	1.69E-01	1.67E-01
Ca41	ε	1.03E5 a	3.73E-05	3.72E-05	3.72E-05	3.72E-05
Cd113m	β^-,*IT*	14.1 a	1.66E-02	1.22E-04	8.91E-07	6.53E-09
Ce144	β^-,γ	284.6 d	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cl36	$\beta^-,\beta^+,\varepsilon$	3.01E5 a	2.93E-03	2.93E-03	2.93E-03	2.93E-03
Cm242	α,γ,*SF*	1.2E3 a	1.09E+00	6.67E-01	4.08E-01	2.50E-01
Cm243	α,γ,*SF*,ε	162.8 d	2.17E-01	1.91E-02	1.68E-03	1.47E-04
Cm244	α,γ,*SF*	29.1 a	5.54E+00	1.20E-01	2.61E-03	5.67E-05
Cm245	α,γ,*SF*	18.1 a	4.00E-02	3.97E-02	3.93E-02	3.90E-02
Cm246	α,γ,*SF*	8.5E3 a	1.43E-02	1.41E-02	1.39E-02	1.37E-02
Co60	β^-,γ	4.76E3 a	8.51E-05	1.65E-10	3.19E-16	6.18E-22
Cs134	$\beta^-,\gamma,\varepsilon$	5.271 a	1.59E-14	3.98E-29	0.00E+00	0.00E+00
Cs135	β^-	2.065 a	1.39E-01	1.39E-01	1.39E-01	1.39E-01
Cs137	β^-,γ	2.3E6 a	1.37E+03	1.36E+02	1.35E+01	1.34E+00
Eu152	$\beta^-,\gamma,\varepsilon\beta^+$	13.48 a	2.92E-03	1.61E-05	8.86E-08	4.88E-10
Eu154	$\beta^-,\gamma,\varepsilon\gamma$	8.59 a	5.45E-02	1.71E-05	5.34E-09	1.68E-12
Eu155	β^-,γ	4.71 a	5.91E-06	2.19E-12	8.07E-19	2.98E-25
Fe55	ε	2.73 a	1.02E-11	9.56E-23	0.00E+00	0.00E+00
H3	β^-	12.3 a	1.43E-01	5.17E-04	1.87E-06	6.75E-09
Ho166m	β^-,γ	1.2E3 a	1.11E-03	1.05E-03	9.93E-04	9.37E-04
I129	β^-,γ	1.57E7 a	7.42E-03	7.42E-03	7.42E-03	7.42E-03
Kr85	β^-,γ	10.73 a	5.93E-01	9.21E-04	1.43E-06	2.23E-09
Mo93	γ,ε,e^-	~3.5E3 a	2.17E-04	2.13E-04	2.09E-04	2.05E-04
Nb93m	*ITe*$^-$	16.1 a	3.41E-01	3.39E-01	3.39E-01	3.39E-01
Nb94	β^-,γ	2.0E4 a	1.86E-02	1.86E-02	1.85E-02	1.84E-02
Ni59	ε	7.6e4 a	5.02E-01	5.01E-01	5.01E-01	5.00E-01
Ni63	β^-	100. a	2.93E+01	1.47E+01	7.34E+00	3.67E+00
Np237	α,γ	2.14E6 a	9.09E-02	1.11E-01	1.28E-01	1.43E-01
Np238	β^-,γ	2.117 d	5.96E-03	3.65E-03	2.23E-03	1.36E-03
Np239	β^-,γ,*SF*	1.2E4 a	5.29E+00	5.24E+00	5.19E+00	5.14E+00
Pa231	α,γ	3.28E4 a	1.95E-05	2.50E-05	3.05E-05	3.60E-05
Pa233	β^-,γ	27.0 d	9.09E-02	1.11E-01	1.28E-01	1.43E-01
Pa234	β^-,γ	6.69 h	8.21E-05	8.21E-05	8.21E-05	8.21E-05
Pa234m	β^-,γ,*IT*	1.17 m	6.32E-02	6.32E-02	6.32E-02	6.32E-02
Pb212	β^-,γ	10.64 h	1.77E-03	6.55E-04	2.43E-04	9.01E-05
Pd107	β^-	6.5E6 a	2.65E-02	2.65E-02	2.65E-02	2.65E-02

Table IV-2. Yucca Mountain Project Radionuclide Inventory for Boiling-Water Reactor Fuels (Continued)

Nuclide ID	YMP/BWR Radionuclide Inventory (Data from *BWR Source Term Generation and Evaluation*, CRWMS M&O 1999j)					
	Decay Mode[a]	Half-Life[b] (yr)	Radioactivity Inventory [Ci/Assembly]			
			125 yr	225 yr	325 yr	425 yr
Pm145	$\alpha,\gamma,\varepsilon$	17.7 a	5.20E-06	1.03E-07	2.06E-09	4.10E-11
Pm146	$\beta^-,\gamma,\varepsilon\gamma$	5.53 a	8.95E-08	3.22E-13	1.16E-18	4.16E-24
Pm147	β^-,γ	2.6234 a	1.33E-10	4.45E-22	0.00E+00	0.00E+00
Po212	α	0.298 μs	1.13E-03	4.20E-04	1.56E-04	5.77E-05
Po216	α,γ	0.145 s	1.77E-03	6.55E-04	2.43E-04	9.01E-05
Pr144	β^-,γ	17.28 m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu236	α,γ,SF	2.87 a	1.64E-07	1.64E-07	1.64E-07	1.64E-07
Pu238	α,γ,SF	87.7 a	2.66E+02	1.21E+02	5.53E+01	2.52E+01
Pu239	α,γ,SF	2.410E4	5.33E+01	5.32E+01	5.31E+01	5.29E+01
Pu240	α,γ,SF	6.56E3	1.13E+02	1.12E+02	1.11E+02	1.10E+02
Pu241	α,β^-,γ	14.4 a	5.41E+01	4.71E-01	4.28E-02	3.91E-02
Pu242	α,γ,SF	3.75E5 a	5.08E-01	5.08E-01	5.08E-01	5.08E-01
Ra224	α,γ	3.66 d	1.77E-03	6.55E-04	2.43E-04	9.01E-05
Rh102	$\beta^-,\gamma,\varepsilon\beta^+$	207. d	2.72E-14	1.13E-24	0.00E+00	0.00E+00
Rh106	β^-,γ	29.9 s	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rn220	α,γ	55.6 s	1.77E-03	6.55E-04	2.43E-04	9.01E-05
Ru106	β^-	1.02 a	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb125	β^-,γ	2.758 a	2.71E-11	2.54E-22	0.00E+00	0.00E+00
Sb126	β^-,γ	12.4 d	1.77E-02	1.77E-02	1.77E-02	1.77E-02
Sb126m	γ,ITe^-	11. s	1.27E-01	1.27E-01	1.26E-01	1.26E-01
Se79	β^-	6.5E4 a	1.59E-02	1.59E-02	1.59E-02	1.59E-02
Sm151	β^-,γ	90 a	2.50E+01	1.16E+01	5.35E+00	2.48E+00
Sn121	β^-	1.128 d	1.32E-01	3.74E-02	1.06E-02	3.00E-03
Sn121m	β^-,γ,ITe^-	55 a	1.70E-01	4.81E-02	1.37E-02	3.87E-03
Sn126	β^-,γ	1.0E5 a	1.27E-01	1.27E-01	1.26E-01	1.26E-01
Sr90	β^-	29.1 a	8.13E+02	6.92E+01	5.90E+00	5.03E-01
Tc99	β^-,γ	2.13E5 a	3.20E+00	3.20E+00	3.20E+00	3.20E+00
Te125m	γ,ITe^-	58. d	6.61E-12	6.21E-23	0.00E+00	0.00E+00
Th228	α,γ	1.913 a	1.77E-03	6.55E-04	2.43E-04	9.01E-05
Th230	α,γ	7.54E4 a	3.49E-04	7.10E-04	1.10E-03	1.51E-03
Th231	β^-,γ	1.063 d	2.63E-03	2.63E-03	2.64E-03	2.64E-03
Th234	β^-,γ	24.10 d	6.32E-02	6.32E-02	6.32E-02	6.32E-02
Tl208	β^-,γ	3.053 m	6.35E-04	2.35E-04	8.73E-05	3.24E-05
U232	α,γ,SF	70 a	1.72E-03	6.37E-04	2.36E-04	8.76E-05
U233	α,γ,SF	1.592E5 a	4.53E-05	8.94E-05	1.42E-04	2.01E-04
U234	α,γ,SF	2.46E5 a	3.64E-01	4.16E-01	4.39E-01	4.50E-01
U235	α,γ,SF	7.04E8 a	2.63E-03	2.63E-03	2.64E-03	2.64E-03
U236	α,γ,SF	2.342E7 a	6.30E-02	6.33E-02	6.36E-02	6.39E-02
		Total[c] =	4.72E+03	1.24E+03	7.71E+02	6.32E+02

NOTES:

(a) Decay modes identified from Parrington et al. (1996).
(b) Half-lives identified from Parrington et al. (1996).
(c) Total radionuclide activities are calculated here and are not part of original reference.

Table IV-2. Yucca Mountain Project Radionuclide Inventory for Boiling-Water Reactor Fuels (Continued)

Nuclide ID	YMP/BWR Radionuclide Inventory (Data from *BWR Source Term Generation and Evaluation*, CRWMS M&O 1999j)				
	Radioactivity Inventory [Ci/Assembly]				
	525 yr	1,025 yr	2,025 yr	5,025 yr	10,025 yr
Ag108	2.18E-05	1.42E-06	6.06E-09	4.68E-16	6.59E-28
Ag108m	2.50E-04	1.63E-05	6.96E-08	5.39E-15	7.58E-27
Am241	3.55E+02	1.59E+02	3.21E+01	2.90E-01	1.80E-02
Am242	1.85E-01	1.58E-02	1.16E-04	4.56E-11	9.64E-22
Am242m	1.85E-01	1.59E-02	1.16E-04	4.58E-11	9.69E-22
Am243	5.09E+00	4.86E+00	4.42E+00	3.34E+00	2.08E+00
Ar39	4.43E-06	1.22E-06	9.28E-08	4.07E-11	1.03E-16
Ba137m	1.26E-01	1.21E-06	1.11E-16	0.00E+00	0.00E+00
Bi212	3.35E-05	3.98E-07	1.69E-07	1.77E-07	1.92E-07
C14	1.65E-01	1.56E-01	1.38E-01	9.59E-02	5.24E-02
Ca41	3.72E-05	3.70E-05	3.68E-05	0.000036	0.0000349
Cd113m	4.79E-11	1.01E-21	0.00E+00	0.00E+00	0.00E+00
Ce144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cl36	2.93E-03	2.92E-03	2.92E-03	2.90E+00	2.86E+00
Cm242	1.53E-01	1.31E-02	9.58E-05	3.78E-11	8.00E-22
Cm243	1.29E-05	6.76E-11	1.85E-21	0.00E+00	0.00E+00
Cm244	1.23E-06	5.94E-15	1.37E-31	0.00E+00	0.00E+00
Cm245	3.87E-02	3.71E-02	3.42E-02	2.68E-02	1.78E-02
Cm246	1.35E-02	1.25E-02	1.08E-02	6.97E-03	3.35E-03
Co60	1.20E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs135	1.39E-01	1.39E-01	1.39E-01	1.39E-01	1.38E-01
Cs137	1.33E-01	1.28E-06	1.18E-16	0.00E+00	0.00E+00
Eu152	2.69E-12	1.37E-23	0.00E+00	0.00E+00	0.00E+00
Eu154	5.23E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu155	1.14E-31	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
H3	2.45E-11	1.51E-23	0.00E+00	0.00E+00	0.00E+00
Ho166m	8.84E-04	6.62E-04	3.72E-04	6.58E-05	3.66E-06
I129	7.42E-03	7.42E-03	7.42E-03	7.42E-03	7.42E-03
Kr85	3.46E-12	3.15E-26	0.00E+00	0.00E+00	0.00E+00
Mo93	2.01E-04	1.82E-04	1.49E-04	8.24E-05	3.06E-05
Nb93m	3.39E-01	3.39E-01	3.38E-01	3.38E-01	3.36E-01
Nb94	1.84E-02	1.81E-02	1.75E-02	1.58E-02	1.33E-02
Ni59	5.00E-01	4.97E-01	4.93E-01	4.79E-01	4.58E-01
Ni63	1.84E+00	5.76E-02	5.66E-05	5.38E-14	4.94E-29
Np237	1.55E-01	1.95E-01	2.21E-01	2.27E-01	2.26E-01
Np238	8.35E-04	7.15E-05	5.24E-07	2.06E-13	4.36E-24
Np239	5.09E+00	4.86E+00	4.42E+00	3.34E+00	2.08E+00
Pa231	4.15E-05	6.91E-05	1.24E-04	2.89E-04	5.59E-04
Pa233	1.55E-01	1.95E-01	2.21E-01	2.27E-01	2.26E-01
Pa234	8.21E-05	8.21E-05	8.21E-05	8.21E-05	8.21E-05
Pa234m	6.32E-02	6.32E-02	6.32E-02	6.32E-02	6.32E-02
Pb212	3.35E-05	3.98E-07	1.69E-07	1.77E-07	1.92E-07
Pd107	2.65E-02	2.65E-02	2.65E-02	0.0265	0.0265

Table IV-2. Yucca Mountain Project Radionuclide Inventory for Boiling-Water Reactor Fuels (Continued)

Nuclide ID	YMP/BWR Radionuclide Inventory (Data from *BWR Source Term Generation and Evaluation*, CRWMS M&O 1999j)				
	Radioactivity Inventory [Ci/Assembly]				
	525 yr	1,025 yr	2,025 yr	5,025 yr	10,025 yr
Pm145	8.18E-13	2.56E-21	0.00E+00	0.00E+00	0.00E+00
Pm146	1.49E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Po212	2.14E-05	2.55E-07	1.08E-07	1.13E-07	1.23E-07
Po216	3.35E-05	3.98E-07	1.69E-07	1.77E-07	1.92E-07
Pr144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu236	1.64E-07	1.63E-07	1.62E-07	1.59E-07	1.55E-07
Pu238	1.16E+01	2.49E-01	1.88E-04	1.01E-10	2.13E-21
Pu239	5.28E+01	5.21E+01	5.08E+01	4.69E+01	4.10E+01
Pu240	1.09E+02	1.03E+02	9.28E+01	6.76E+01	3.98E+01
Pu241	3.88E-02	3.72E-02	3.43E-02	2.69E-02	1.79E-02
Pu242	5.08E-01	5.07E-01	5.06E-01	5.04E-01	4.99E-01
Ra224	3.35E-05	3.98E-07	1.69E-07	1.77E-07	1.92E-07
Rh102	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rn220	3.35E-05	3.98E-07	1.69E-07	1.77E-07	1.92E-07
Ru106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb125	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb126	1.77E-02	1.76E-02	1.75E-02	1.71E-02	1.66E-02
Sb126m	1.26E-01	1.26E-01	1.25E-01	1.22E-01	1.18E-01
Se79	1.59E-02	1.59E-02	1.58E-02	1.57E-02	1.56E-02
Sm151	1.15E+00	2.43E-02	1.10E-05	1.01E-15	2.28E-32
Sn121	8.52E-04	1.56E-06	5.27E-12	2.01E-28	0.00E+00
Sn121m	1.10E-03	2.02E-06	6.78E-12	2.59E-28	0.00E+00
Sn126	1.26E-01	1.26E-01	1.25E-01	1.22E-01	1.18E-01
Sr90	4.28E-02	1.92E-07	3.88E-18	0.00E+00	0.00E+00
Tc99	3.20E+00	3.19E+00	3.18E+00	3.15E+00	3.10E+00
Te125m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Th228	3.35E-05	3.98E-07	1.69E-07	1.77E-07	1.92E-07
Th230	1.93E-03	4.02E-03	8.17E-03	2.03E-02	3.97E-02
Th231	2.65E-03	2.68E-03	2.73E-03	2.87E-03	3.09E-03
Th234	6.32E-02	6.32E-02	6.32E-02	6.32E-02	6.32E-02
Tl208	1.20E-05	1.43E-07	6.06E-08	6.34E-08	6.88E-08
U232	3.25E-05	3.88E-07	1.62E-07	1.59E-07	1.55E-07
U233	2.66E-04	6.51E-04	1.57E-03	4.47E-03	9.26E-03
U234	4.55E-01	4.58E-01	4.57E-01	4.54E-01	4.48E-01
U235	2.65E-03	2.68E-03	2.73E-03	2.87E-03	3.09E-03
U236	6.43E-02	6.58E-02	6.87E-02	7.58E-02	8.35E-02
Total (c) =	5.49E+02	3.31E+02	1.91E+02	1.28E+02	9.15E+01

NOTES:

(a) Decay modes identified from Parrington et al. 1996.

(b) Half-lives identified from Parrington et al. 1996.

(c) Total radionuclide activities are calculated here and are not part of original reference.

Table IV-3. Alpha-Recoil Enhanced (from Both α and α-Recoil Atom) Dissolution Rates Due to the Major Mass-Based Constituents of SNF and HLW to be Disposed in the Yucca Mountain Potential Repository

Nuclide ID	Decay Mode	Half–Life [a] (yr)	Fraction Decay Rate [b] (1/yr)	α–Decay Rate in 13 Mono-Layers[c] (g/m^2–yr)
^{238}U	α, γ, SF	4.47×10^{9}	1.55×10^{-10}	6.05×10^{-12}
^{235}U	α, γ, SF	7.04×10^{8}	9.85×10^{-10}	3.84×10^{-11}
^{239}Pu	α, γ, SF	2.410×10^{4}	2.88×10^{-5}	1.12×10^{-6}
^{236}U	α, γ, SF	2.342×10^{7}	2.96×10^{-8}	1.15×10^{-9}

(a) Half-life values in Sanchez et al. (1998).

(b) The fraction decay rate, also known as the decay constant, is given by $\lambda=\ln(2)/t_{½}$, where $t_{½}$ is the radionuclide half-life given by values in column 3.

(c) Each mono-layer thickness is 3.0 Å (3.0×10^{-10}m), and the density is upper bounded at 19.86 g/cm^3 (theoretical density of pure plutonium metal [Wick 1980]).

ATTACHMENT V— SECONDARY FEPs RELATED TO THE PRIMARY MISCELLANEOUS WASTE FORM FEPs ADDRESSED IN THIS AMR

This attachment provides the identification and a brief accounting of all the secondary FEPs that are subordinate to the primary FEPs covered in this AMR. Each section heading identifies the primary FEP and the section of the main text of the AMR that this FEP is from.

V-1. Magma Interacts with Waste—YMP No. 1.2.04.04.00 from Section 6.2.1

FEP Number and Name: 1.2.04.04.01, Magmatic Volatiles Attack Waste

Relationship to Primary FEP: Appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 1.2.04.04.02, Dissolution of Spent Fuel in Magma

Relationship to Primary FEP: Appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 1.2.04.04.03, Dissolution of Other Waste In Magma

Relationship to Primary FEP: Not applicable. Only spent nuclear fuel will be disposed.

Screening and Disposition: Excluded based on low probability (not credible).

FEP Number and Name: 1.2.04.04.04, Heating of Waste Container by Magma (Without Contact)

Relationship to Primary FEP: Appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 1.2.04.04.05, Failure of Waste Container by Direct Contact W/Magma

Relationship to Primary FEP: Appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 1.2.04.04.06, Fragmentation

Relationship to Primary FEP: Appropriately addressed in Attachment I of the primary FEP.

Screening and Disposition: Same as the primary FEP.

V-2. Waste Inventory—YMP No. 2.1.01.01.00 from Section 6.2.2

FEP Number and Name: 2.1.01.01.01, Inventory

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.02, Inventory

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.03, Changes in Radionuclide Inventory (in Waste Form)

Relationship to Primary FEP: Changes in radionuclide inventory occur over time due to decay, reactor conditions and burnup. These specific processes are adequately covered in the broader context of the primary FEP.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.04, Waste Product (Glass)

Relationship to Primary FEP: Waste materials specific to HLW glass is a subtopic of waste inventory and is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.05, Exotic Fuels

Relationship to Primary FEP: This FEP is a place holder. The spent fuel waste inventory will be well characterized before shipment to the repository; therefore, any exotic fuels would be appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.06, DOE SNF Gap Radionuclide Inventory

Relationship to Primary FEP: DOE SNF gap radionuclide inventory is a feature of the waste inventory and is appropriately included in the broader context of the primary FEP. However, it is not addressed specifically because no credit is taken for DSNF cladding due to its generally poor condition. Therefore, the inventory available for DSNF gap-and-grain release is lumped with releases occurring congruently with waste degradation.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.07, DOE SNF Initial Radionuclide Inventory

Relationship to Primary FEP: DOE SNF initial radionuclide inventory is a specific feature of the waste inventory and, therefore, this FEP is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.08, DOE SNF Structure

Relationship to Primary FEP: This secondary FEP is concerned with the fact that DSNF consists of a wide variety of fuel types that can be substantially different than CSNF. This feature of DSNF is a subtopic of waste inventory and is appropriately accounted for in the broader context of the primary FEP.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.09, DOE SNF Initial Radionuclide Inventory

Relationship to Primary FEP: DOE SNF initial radionuclide inventory is a feature of the primary FEP, which appropriately addresses the waste inventory for DOE SNF.

Screening and Disposition: Included.

FEP Number and Name: 2.1.01.01.10, DOE SNF Hazardous Chemical Inventory

Relationship to Primary FEP: Specifically addressed by primary FEP.

Screening and Disposition: Excluded (not credible) because hazardous materials are prohibited by the waste-acceptance criteria.

V-3. Codisposal/Collocation of Waste—YMP No. 2.1.01.02.00 from Section 6.2.3

FEP Number and Name: 2.1.01.02.01, Other Waste

Relationship to Primary FEP: If other waste is taken to be other than CSNF, DSNF, and HLW, this FEP is not credible. All waste planned for disposal is accounted for in the broader context of the primary FEP.

Screening and Disposition: Excluded.

FEP Number and Name: 2.1.01.02.02, Codisposal of Reactive Wastes

Relationship to Primary FEP: Codisposal of reactive wastes is a redundant feature addressed in the primary FEP and is retained in the FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.03, Near Storage of Other Waste

Relationship to Primary FEP: Near storage of other waste is a redundant feature equivalent to collocation. Collocation is the issue that different types of waste packages will be stored in close proximity. Therefore, this FEP is a redundant feature addressed in the primary FEP and is retained in the FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.04, DOE SNF/HLW Glass Interactions

Relationship to Primary FEP: DSNF/HLW interactions within a codisposal package is a specific process that is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.05, DOE SNF Waste Package Placement

Relationship to Primary FEP: DSNF waste-package placement is a specific feature (DSNF/HLW and CSNF collocation) that is appropriately addressed in the broader context of collocation in the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.06, DOE SNF Canister Arrangement Within Waste Package

Relationship to Primary FEP: DOE SNF canister arrangement within waste package is a specific DSNF feature that is appropriately addressed in the broader context of codisposal in the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.07, DOE SNF Collocation With HLW

Relationship to Primary FEP: DSNF collocation with HLW is equivalent to DSNF/HLW codisposal which is a specific feature that is appropriately addressed in the broader context of codisposal in the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.08, DOE SNF Geometry

Relationship to Primary FEP: DOE SNF geometry is a specific feature that is appropriately included in the broader context of codisposal in the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.02.09, DOE SNF Waste Package Placement

Relationship to Primary FEP: Redundant secondary FEP (see 2.1.01.02.05) retained in FEP list for completeness.

Screening and Disposition: Same as secondary FEP 2.1.01.02.05.

FEP Number and Name: 2.1.01.02.10, DOE SNF Collocation With HLW (Waste Form Degradation Impact)

Relationship to Primary FEP: The impact of DSNF/HLW collocation (actually codisposal) on waste-form degradation is a specific process that is appropriately addressed in the primary FEP.

Screening and Disposition: Same as the primary FEP. Degradation of DSNF and HLW within a single codisposal waste package is treated by handling the degradation of each waste form independently within a common chemical environment that can be altered by either waste form.

FEP Number and Name: 2.1.01.02.11, DOE SNF Collocation With HLW (Radionuclide Mobilization Impact)

Relationship to Primary FEP: The impact of DSNF/HLW collocation (actually codisposal) on radionuclide mobilization is a specific process that is appropriately addressed in the primary FEP.

Screening and Disposition: Same as the primary FEP. Mobilization of radionuclides within a single DSNF/HLW codisposal waste package is treated by handling mobilization of radionuclides from each waste form independently within a common chemical environment governed by solubility limits and the limited amount of available water.

FEP Number and Name: 2.1.01.02.12, DOE SNF Collocation With HLW (Cladding Degradation Impact)

Relationship to Primary FEP: The impact of DSNF/HLW collocation (actually codisposal) on DSNF cladding is a process that is not specifically addressed in the primary FEP. However, it is addressed in the primary FEP 2.1.02.25.00, DSNF Cladding Degradation.

Screening and Disposition: Same as the primary FEP 2.1.02.25.00, DSNF Cladding Degradation, which concludes that no credit will be taken for DSNF cladding because it is, generally, in poor condition. Therefore, codisposal of DSNF and HLW will have no impact on DSNF cladding degradation and is excluded from consideration.

V-4. Heterogeneity of Waste Forms—YMP No. 2.1.01.03.00 from Section 6.2.4

FEP Number and Name: 2.1.01.03.01, Damaged or Deviating Fuel

Relationship to Primary FEP: Damaged or deviating fuel is a feature affecting heterogeneity of waste forms. However, its effect on inventory and the degradation of the waste has been more appropriately addressed in the primary FEPs 2.1.01.01.00, Inventory; 2.1.02.02.00, CSNF Alteration, Dissolution and Radio Nuclide Release; 2.1.02.01.00, DSNF Alteration, Degradation, and Dissolution; and Glass Degradation, 2.1.02.03.00, Alteration and Dissolution.

Screening and Disposition: Included in development of waste form degradation models. See primary FEPs 2.1.01.01.00, Inventory; 2.1.02.02.00, CSNF Alteration, Dissolution and Radio Nuclide Release; 2.1.02.01.00, DSNF Alteration, Degradation, and Dissolution; and Glass Degradation, 2.1.02.03.00, Alteration and Dissolution.

FEP Number and Name: 2.1.01.03.02, Heterogeneity of Waste Form

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.01.03.03, Deviant Inventory Flask (Inventory)

Relationship to Primary FEP: This FEP is concerned with waste packages that might have inventories significantly different than the typical package. This FEP is more appropriately addressed in the broader context of the primary FEP 2.1.01.01.00 on Waste Inventory.

Screening and Disposition: Same as FEP 2.1.01.01.00, Waste Inventory.

FEP Number and Name: 2.1.01.03.04, DOE SNF Canister Atmosphere

Relationship to Primary FEP: Implicitly included in YMP No. 2.1.02.09.00, Void Space (in Waste Package). The influence of void volume on the uncertainty in the in-package chemistry is small compared to variability in water inflow rates into the waste package used to define the uncertainty in the in-package chemistry.

Screening and Disposition: Included.

V-5. Spatial Heterogeneity of Emplaced Waste—YMP No. 2.1.01.04.00 from Section 6.2.5

No Secondary FEPs.

V-6. DSNF Degradation, Alteration, and Dissolution—YMP No. 2.1.02.01.00 from Section 6.2.6

FEP Number and Name: 2.1.02.01.01, DOE SNF Dissolution

Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.01.02, Alteration/Dissolution of DOE SNF

Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.01.03, Oxidation of DOE SNF

Relationship to Primary FEP: Oxidation is a specific DOE SNF degradation process that is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.01.04, Alteration/Dissolution of Pu Ceramic Waste

Relationship to Primary FEP: Alteration/Dissolution of Pu Ceramic Waste is a subtopic, which is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP. Alteration/dissolution of Pu ceramic waste is conservatively bounded by degradation of HLW glass. Therefore, it is modeled by specifying that the Pu ceramic inventory is uniformly distributed in the HLW glass phase.

FEP Number and Name: 2.1.02.01.05, High Integrity Canisters for DOE SNF

Relationship to Primary FEP: The effect of the_high integrity canisters (HIC) on DOE SNF degradation is a specific process that is appropriately addressed in the primary FEP

Screening and Disposition: HICs have been conservatively excluded from consideration in the TSPA because analysis shows release of the inventory in HICs is delayed by 60,000 years and peak dose due to the HIC inventory is 2 to 3 orders of magnitude less than TSPA-VA base case.

V-7. CSNF Alteration, Dissolution, and Radionuclide Release—YMP No. 2.1.02.02.00 from Section 6.2.7

FEP Number and Name: 2.1.02.02.01, Source Terms (Expected)

Relationship to Primary FEP: This FEP is concerned with modes of radionuclide release—congruent and instantaneous. Congruent release is appropriately addressed in the broader context of the primary FEP. Instant release or gap-and-grain release is appropriately addressed in the primary FEP 2.1.02.07.00, Gap-and-Grain Release.

Screening and Disposition: Congruent release is included as discussed in the primary FEP. Instantaneous release is included as discussed in primary FEP 2.1.02.07.00, Gap and Grain Release.

FEP Number and Name: 2.1.02.02.02, Source Terms (Other) (in Waste Form)

Relationship to Primary FEP: This FEP is concerned with other release modes such as mechanical breakdown, phase change, and selective leaching. These processes are not specifically addressed in the primary FEP. However, they are subsumed in the model.

Screening and Disposition: Implicitly included as discussed in the primary FEP since the coefficients for the kinetic-rate equation are evaluated through regression analysis on high-flow-rate experimental data and would, therefore, implicitly include all release modes.

FEP Number and Name: 2.1.02.02.03, Stability of UO_2 (in Waste Form)

Relationship to Primary FEP: This FEP is concerned with the effects of the chemical environment on the stability of UO_2. These effects are appropriately addressed in the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.02.04, Degradation of Fuel Elements

Relationship to Primary FEP: Redundant topic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.02.05, Corrosion of Metal Parts (in Waste Form)

Relationship to Primary FEP: Corrosion of cladding has been more appropriately addressed in specific cladding FEPs 2.1.02.12.00 through 2.1.02.27.00.

Screening and Disposition: See cladding FEPs 2.1.02.12.00 through 2.1.02.27.00.

FEP Number and Name: 2.1.02.02.06, Corrosion Prior to Wetting

Relationship to Primary FEP: Prior to the breach of the waste packages, there should be minimal corrosion because of the inert environment in the package.

Screening and Disposition: This subject is better treated under YMP FEP 2.1.03.06.00, Internal Corrosion of Waste Container.

FEP Number and Name: 2.1.02.02.07, Radionuclide Release (Diffusion) Through Failed Cladding

Relationship to Primary FEP: DCCG is excluded as a creep-rupture mechanism. Creep rupture, as a cladding perforation process, however, is included in the CSNF Cladding Degradation Component of the Waste Form Degradation Model as described in FEP 2.1.02.19.00.

Screening and Disposition: This subject is treated under YMP FEP 2.1.02.26.00, Diffusion-Controlled Cavity Growth (DCCG).

FEP Number and Name: 2.1.02.02.08, Water Turnover, Steel Vessel

Relationship to Primary FEP: In the process model for evaluating the In-Package Chemistry Component, the mixing-cell volume is equal to the fixed void volume in the waste package; hence, void space is indirectly included in the In-Package Chemistry Component.

Screening and Disposition: Better treated in YMP No. 2.1.02.09.00, Void Space (in Waste Package).

FEP Number and Name: 2.1.02.02.09, Dissolution Chemistry (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.02.10, Release from Fuel Matrix (Release/Migration Factors)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP. The degradation rate bounds the availability of radionuclides for potential mobilization.

FEP Number and Name: 2.1.02.02.11, Release from Metal Parts

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as 2.1.02.02.05, Corrosion of Metal Parts (in Waste Form).

FEP Number and Name: 2.1.02.02.12, Total Release from Fuel Elements

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.02.13, Dissolution of Waste (Release/Migration Factors)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.02.14, Release of Radionuclides from the Failed Canister

Relationship to Primary FEP: Release of radionuclides is really tied to two primary FEPs: YMP No. 2.1.02.02.00, CSNF Alteration, Dissolution, and Radionuclide Release, and YMP No. 2.1.09.04.00, Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and EBS.

Screening and Disposition: Included as described in the primary FEP 2.1.09.04.00, Radionuclide Solubility, Solubility Limits, and Speciation.

FEP Number and Name: 2.1.02.02.15, Transport and Release of Nuclide, Failed Canister

Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-8. Glass Degradation, Alteration, and Dissolution—YMP No. 2.1.02.03.00 from Section 6.2.8

FEP Number and Name: 2.1.02.03.01, Degradation and Alteration of Glass Waste Form

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.03.02, Phase Separation (in Waste Form)

Relationship to Primary FEP: Phase separation is a subtopic of glass degradation that is specifically addressed in the broader context of the primary FEP.

Screening and Disposition: Excluded on low consequence because of waste-production controls and acceptance processes.

FEP Number and Name: 2.1.02.03.03, Congruent Dissolution (in Waste Form)

Relationship to Primary FEP: Congruent dissolution is a feature that is specifically addressed in the broader context of the primary FEP.

Screening and Disposition: Included as discussed in the primary FEP.

FEP Number and Name: 2.1.02.03.04, Rate of Glass Dissolution

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.03.05, Selective Leaching (in Waste Form)

Relationship to Primary FEP: Selective leaching is a sub-process that is specifically addressed in the broader context of the primary FEP.

Screening and Disposition: Excluded by primary. No credit is taken for selective leaching because it would reduce release from waste packages by selectively immobilizing certain radionuclides. That is, the chosen conservative modeling approach bounds the rate of release.

FEP Number and Name: 2.1.02.03.06, Coprecipitates/Solid Solutions (in Waste Form)

Relationship to Primary FEP: Coprecipitates/solid solutions is a sub-feature of the primary FEP that is specifically addressed in the broader context of the primary FEP. Secondary phases are also addressed in primary FEP 2.1.09.10.00, Secondary Phase Effects on Dissolved Radionuclide Concentrations at the Waste Form.

Screening and Disposition: Excluded by primary. No credit is taken for the retardation effects of secondary phases.

FEP Number and Name: 2.1.02.03.07, Precipitation of Silicates/Silica Gel (in Waste Form)

Relationship to Primary FEP: Precipitation of silicates/silica gel is a subtopic that is specifically addressed in the broader context of the primary FEP. This FEP is also a redundant subtopic of secondary FEP 2.1.02.03.06.

Screening and Disposition: Excluded by primary. No credit is taken for the retardation affects of secondary phases.

FEP Number and Name: 2.1.02.03.08, Iron Corrosion Products

Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.03.09, Radionuclide Release from Glass

Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.03.10, Composition of DHLW Glass

Relationship to Primary FEP: Composition of DHLW glass is a feature that is accounted for in the model development discussed in the primary FEP.

Screening and Disposition: The glass dissolution model is derived from a reference glass composition applicable to glasses whose composition lies within specified bounds.

V-9. Alpha Recoil Enhances Dissolution—YMP No. 2.1.02.04.00 from Section 6.2.9

FEP Number and Name: 2.1.02.04.01, Recoil of Alpha-Decay

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-10. Glass Cracking and Surface Area—YMP No. 2.1.02.05.00 from Section 6.2.10

FEP Number and Name: 2.1.02.05.01, Solute Transport Resistance (in Waste Form)

Relationship to Primary FEP: The limited dimension of cracks in the glass block is expected to offer a resistance to radionuclide (and silica) diffusion/transport. This feature is not specifically addressed in this primary FEP. However, it is implicitly included in the discussion of pathways for unsaturated flow and transport in primary FEP 2.1.08.07.00, Pathways for Unsaturated Flow and Transport.

Screening and Disposition: No credit is taken for resistance to radionuclide (and silica) diffusion/transport through cracks in the glass block. Instead a conservative mixing-cell approach is used as discussed in primary FEP 2.1.08.07.00, Pathways for Unsaturated Flow and Transport.

V-11. Glass Recrystallization—YMP No. 2.1.02.06.00 from Section 6.2.11

No Secondary FEPs.

V-12. Pyrophoricity—YMP No. 2.1.02.08.00 from Section 6.2.12

FEP Number and Name: 2.1.02.08.01, DOE SNF Pyrophoricity

Relationship to Primary FEP: DOE SNF pyrophoricity is a redundant subtopic of pyrophoricity that is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.02.08.02, DOE SNF Pyrophoric Event (Waste Heat Impact)

Relationship to Primary FEP: The waste-heat impact of DOE SNF pyrophoric events (PEs) is a specific subtopic of pyrophoricity that is appropriately subsumed in the broader context of the analysis in the primary FEP. The assumption in the primary FEP analysis that a PE causes breach of the two adjacent waste packages provides reasonable bound to the effects of waste heat from a PE.

Screening and Disposition: Waste-heat impact of a PE is excluded based on low consequence to dose as discussed in the primary FEP.

FEP Number and Name: 2.1.02.08.03, DOE SNF Pyrophoric Event (Waste Package Degradation Impact)

Relationship to Primary FEP: The impact of a PE on waste-package degradation is appropriately subsumed in the broader context of the analysis in the primary FEP. The assumption in the primary FEP analysis that a PE causes breach of the two adjacent waste packages, provides a reasonable bound to the effects of a PE on waste-package degradation.

Screening and Disposition: The impact of a PE on waste-package degradation has been excluded based on low consequence to dose as discussed in the primary FEP.

FEP Number and Name: 2.1.02.08.04, Acetylene Generation From DSNF

Relationship to Primary FEP: Elevated to a primary FEP, YMP 2.1.02.29.00, Flammable Gases Generation from DSNF (recommended new title), Section 6.2.13.

Screening and Disposition: See primary FEP on topic.

FEP Number and Name: 2.1.02.08.05, DOE SNF Pyrophoric Event (Waste Form Degradation Impact)

Relationship to Primary FEP: The impact of a PE on waste-form degradation is appropriately subsumed in the broader context of the analysis in the primary FEP. The assumption in the primary FEP analysis that a PE causes breach of and radionuclide release from the two adjacent waste packages, provides a reasonable bound to the effects of a PE on waste-form degradation.

Screening and Disposition: The impact of a PE on waste-form degradation has been excluded based on low consequence to dose as discussed in the primary FEP.

FEP Number and Name: 2.1.02.08.06, DOE SNF Pyrophoric Event (Cladding Degradation Impact)

Relationship to Primary FEP: The impact of a PE on cladding degradation is appropriately subsumed in the broader context of the analysis in the primary FEP. The assumption in the primary FEP analysis that a PE causes breach of the two adjacent waste packages and cladding provides a reasonable bound to the effects of a PE on cladding degradation.

Screening and Disposition: The impact of a PE on cladding degradation has been excluded based on low consequence to dose as discussed in the primary FEP.

V-13. Flammable Gases Generation from DSNF—YMP No. 2.1.02.29.00 from Section 6.2.13

No Secondary FEPs.

V-14. Void Space (in Waste Package)—YMP No. 2.1.02.09.00 from Section 6.2.14

No Secondary FEPs.

V-15. Cellulosic Degradation—YMP No. 2.1.02.10.00 from Section 6.2.15

No Secondary FEPs.

V-16. DSNF Cladding Degradation—YMP No. 2.1.02.25.00 from Section 6.2.16

FEP Number and Name: 2.1.02.25.01, DOE SNF Cladding Material
Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.
Screening and Disposition: Same as the Primary FEP.

FEP Number and Name: 2.1.02.25.02, DOE SNF Cladding Condition
Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.
Screening and Disposition: Same as the Primary FEP.

FEP Number and Name: 2.1.02.25.03, Internal Canister/Cladding Corrosion Due to DOE SNF
Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness. Addressed in YMP FEP 2.1.03.06.00, Internal Corrosion of Waste Container
Screening and Disposition: Same as the Primary FEP.

V-17. Various Features of the Approximately 250 DSNF Types and Grouping into Waste Categories—YMP No. 2.1.02.28.00 from Section 6.2.17

No Secondary FEPs.

V-18. Internal Corrosion of Waste Container—YMP No. 2.1.03.06.00 from Section 6.2.18

FEP Number and Name: 2.1.03.06.01, DOE SNF Waste Package Internal Corrosion
Relationship to Primary FEP: DOE SNF waste-package internal corrosion is a redundant subtopic of internal corrosion that is appropriately addressed in the broader context of the primary FEP.
Screening and Disposition: Same as the primary FEP.

V-19. Pathways for Unsaturated Flow and Transport in the Waste and EBS—YMP No. 2.1.08.07.00 from Section 6.2.19

FEP Number and Name: 2.1.08.07.01, Residual Canister (Crack/Holes Effects)
Relationship to Primary FEP: This FEP is concerned with the fact that a failed waste package can still provide resistance to flow and transport. This feature is not specifically addressed by

the primary FEP. Primary FEP 2.1.08.08.00, Induced Hydrological Changes in the Waste and EBS, discusses breach cross-sectional areas of breaches in waste packages.

Screening and Disposition: The cross-sectional area of breaches in waste packages are estimated by waste-package-degradation modeling and used to calculate fluid flow rates and transport by advection and diffusion.

FEP Number and Name: 2.1.08.07.02, Properties of Failed Canister

Relationship to Primary FEP: This FEP is concerned with the chemical and physical (transport) properties of the failed waste package. Physical (transport) properties is a redundant topic addressed in secondary FEP 2.1.08.07.01, Residual Canister (Crack/Holes Effects). Chemical properties are more appropriately addressed in primary FEP 2.1.09.02.00, Interaction with Corrosion Products.

Screening and Disposition: Included as discussed in 2.1.08.07.01, Residual Canister, and 2.1.09.02.00, Interaction with Corrosion Products.

FEP Number and Name: 2.1.08.07.03, Container-Partial Corrosion

Relationship to Primary FEP: Redundant topic of secondary FEP 2.1.08.07.01.

Screening and Disposition: Same as secondary FEP 2.1.08.07.01.

FEP Number and Name: 2.1.08.07.04, Hydraulic Conductivity (In Waste And EBS)

Relationship to Primary FEP: Redundant subtopic of primary FEP 2.1.08.08.00, Induced Hydrological Changes in the Waste and EBS.

Screening and Disposition: Same as primary FEP 2.1.08.08.00, Induced Hydrological Changes.

FEP Number and Name: 2.1.08.07.05, Consolidation of Waste

Relationship to Primary FEP: Elevated to a primary FEP, YMP 2.1.08.15.00, addressed in Section 6.2.20, Waste Form and Backfill Consolidation.

Screening and Disposition: See FEP discussion in Section 6.2.20.

FEP Number and Name: 2.1.08.07.06, Channeling Within the Waste

Relationship to Primary FEP: Channeling within the waste is a subtopic that is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.08.07.07, Unsaturated Transport (Water Transport)

Relationship to Primary FEP: Unsaturated transport is a specific subtopic that is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.08.07.08, Radionuclide Transport (Water Transport)

Relationship to Primary FEP: Radionuclide transport is a specific subtopic that is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

V-20. Waste-Form and Backfill Consolidation—YMP No. 2.1.08.15.00 from Section 6.2.20

Treated as Primary FEP in Section 6.2.20.

V-21. Induced Hydrological Changes in the Waste and EBS—YMP No. 2.1.08.08.00 from Section 6.2.21

No Secondary FEPs.

V-22. Desaturation/Dewatering of the Repository—YMP No. 2.1.08.10.00 from Section 6.2.22

FEP Number and Name: 2.1.08.10.01, Dewatering of Host Rock (In Waste and EBS)

Relationship to Primary FEP: Redundant for waste and EBS, retained in FEP list for completeness. Dewatering of host rock is beyond the scope of this AMR.

Screening and Disposition: Included as discussed in primary FEP.

FEP Number and Name: 2.1.08.10.02, Dewatering

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-23. Properties of the Potential Carrier Plume in the Waste and EBS—YMP No. 2.1.09.01.00 from Section 6.2.23

FEP Number and Name: 2.1.09.01.01, Reactions with Cement Pore Water

Relationship to Primary FEP: Reactions with cement pore water (if there is any) could affect properties of the potential carrier plume and, therefore, are appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Excluded as discussed in the primary FEP. Changes to water chemistry outside the waste package are overwhelmed by the range of uncertainty used for in-package chemistry. Also, the reactions with cement pore water can be excluded based on the expected minimal use of cementitious material in the repository design.

FEP Number and Name: 2.1.09.01.02, Reactions With Cement Pore Water

Relationship to Primary FEP: Redundant topic of secondary FEP 2.1.09.01.01, Reactions With Cement Pore Water. Retained in FEP list for completeness.

Screening and Disposition: Same as the secondary FEP 2.1.09.01.01.

FEP Number and Name: 2.1.09.01.03, Induced Chemical Changes (in Waste and EBS)

Relationship to Primary FEP: Chemical properties of the carrier plume could change as it flows through the EBS and waste package. These processes are appropriately addressed in the broader context of the primary FEP. Changes to water chemistry outside the waste package are overwhelmed by the range of uncertainty used for in-package chemistry.

Screening and Disposition: Same as the primary FEP. Induced chemical changes outside the waste package are excluded. Incoming water is assumed to have the properties of J-13 well water. Possible chemical changes within the waste package are included in establishing the uncertainty band in the in-package chemistry model.

FEP Number and Name: 2.1.09.01.04, Interactions of Host Materials And Ground Water With Repository Material

Relationship to Primary FEP: This FEP is only a rephrasing of the secondary FEP 2.1.09.01.03 and is, therefore, redundant. It is retained in FEP list for completeness.

Screening and Disposition: Same as secondary FEP 2.1.09.01.03.

FEP Number and Name: 2.1.09.01.05, TRU Silos Cementitious Plume

Relationship to Primary FEP: TRU specific subtopic of secondary FEP 2.1.09.01.01, Reactions with cement pore water. Retained in FEP list for completeness.

Screening and Disposition: Same as the secondary FEP. 2.1.09.01.01.

FEP Number and Name: 2.1.09.01.06, Water Chemistry, Canister

Relationship to Primary FEP: In-package water chemistry is a subtopic which is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Included as discussed in primary FEP. Possible chemical changes within the waste package are included in establishing the uncertainty band for the In-Package Chemistry Component.

FEP Number and Name: 2.1.09.01.07, Transport of Chemically-Active Substances into the Near-Field

Relationship to Primary FEP: Transport of chemically active substances into the near field is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Excluded as discussed in the primary FEP. The invert will be filled with fine quartz sand or crushed rock that is relatively inert in any anticipated repository environment. Sorption is conservatively assumed to zero.

FEP Number and Name: 2.1.09.01.08, Incomplete Near-Field Chemical Conditioning

Relationship to Primary FEP: This FEP is concerned with the development of hyperalkaline, reducing conditions in the near-field as the principal barrier to release of radionuclides. If this chemical conditioning were incomplete or were lost, the only barrier to radionuclides would be the far field. This is not specifically addressed by the primary FEP. However, YMP is not relying on near-field chemical conditioning as a barrier to nuclide release. Instead, the two layer waste package, cladding, drip shield, and large unsaturated zone beneath the repository form the multi-barrier system for YMP.

Screening and Disposition: Conservatively excluded based on low consequence.

FEP Number and Name: 2.1.09.01.09, Chemical Processes (In Waste and EBS)

Relationship to Primary FEP: This a rewording of secondary FEP 2.1.09.01.03, Induced Chemical Changes (in Waste and EBS), and is, therefore, redundant. It is retained in the FEP list for completeness.

Screening and Disposition: Same as secondary FEP 2.1.09.01.03, Induced Chemical Changes in Waste and EBS.

FEP Number and Name: 2.1.09.01.10, Hyperalkaline Carrier Plume Forms

Relationship to Primary FEP: This is a special case which is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.01.11, Chemical Interactions (In Waste And EBS)

Relationship to Primary FEP: This is a rewording of secondary FEP 2.1.09.01.03, Induced Chemical Changes (in Waste and EBS), and is, therefore, redundant. It is retained in the FEP list for completeness.

Screening and Disposition: Same as secondary FEP 2.1.09.01.03.

FEP Number and Name: 2.1.09.01.12, TRU Alkaline or Organic Plume

Relationship to Primary FEP: This FEP is a TRU-specific case. TRU waste is not part of the inventory for YMP.

Screening and Disposition: Excluded based on low probability (not credible).

FEP Number and Name: 2.1.09.01.13, Interactions of Waste and Repository Materials With Host Materials

Relationship to Primary FEP: This a rewording of secondary FEP 2.1.09.01.04, Interactions of Host Materials And Ground Water With Repository Material, and is, therefore, redundant. It is retained in the FEP list for completeness.

Screening and Disposition: Same as secondary FEP 2.1.09.01.04.

FEP Number and Name: 2.1.09.01.14, TRU Alkaline or Organic Plume

Relationship to Primary FEP: This FEP is a TRU-specific case. TRU waste is not part of the inventory for YMP.

Screening and Disposition: Excluded based on low probability (not credible).

V-24. Interaction with Corrosion Products—YMP No. 2.1.09.02.00 from Section 6.2.24

FEP Number and Name: 2.1.09.02.01, Interactions with Corrosion Products and Waste

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the Primary FEP.

FEP Number and Name: 2.1.09.02.02, Effects of Metal Corrosion (in Waste and EBS)

Relationship to Primary FEP: 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 2.1.03.06.00, Internal Corrosion of Waste Container.

Screening and Disposition: Better treated in YMP No. 2.1.03.06.00, Internal Corrosion of Waste Container.

FEP Number and Name: 2.1.09.02.03, Container Corrosion Products

Relationship to Primary FEP: 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 2.1.03.06.00, Internal Corrosion of Waste Container.

Screening and Disposition: Better treated in YMP No. 2.1.03.06.00, Internal Corrosion of Waste Container.

FEP Number and Name: 2.1.09.02.04, Chemical Buffering (Canister Corrosion Products)

Relationship to Primary FEP: 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 YMP 2.1.03.06.00, Internal Corrosion of Waste Container.

Screening and Disposition: Better treated in YMP 2.1.03.06.00, Internal Corrosion of Waste Container.

FEP Number and Name: 2.1.09.02.05, Radionuclide Sorption and Coprecipitation (in EBS)

Relationship to Primary FEP: Specifically addressed in primary FEP.

Screening and Disposition: No credit is taken for the retarding of radionuclides by sorption within the waste package or EBS.

V-25. Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and EBS—YMP No. 2.1.09.04.00 from Section 6.2.25

FEP Number and Name: 2.1.09.04.01, Elemental Solubility (in Waste and EBS)

Relationship to Primary FEP: This FEP is a sub-topic addressing solubility of radionuclides released from HLW glass. It is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.02, Speciation (in Waste and EBS)

Relationship to Primary FEP: This FEP is concerned with temperature and chemistry effects on chemical speciation. The topic is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.03, Geochemical Pump (in Waste and EBS)

Relationship to Primary FEP: This FEP is interpreted as the processes of dissolution, transport, and precipitation. Dissolution and precipitation are appropriately addressed in the broader context of the primary FEP. Transport is beyond the scope of the primary FEP and is addressed in the primary FEP 2.1.08.07.00, Pathways for Unsaturated Flow and Transport.

Screening and Disposition: Included as discussed in primary and 2.1.08.07.00, Pathways for Unsaturated Flow and Transport.

FEP Number and Name: 2.1.09.04.04, Precipitation and Dissolution (in Waste and EBS)

Relationship to Primary FEP: Redundant subtopics of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.05, Selective Dissolution of Contaminants Contained in SNF

Relationship to Primary FEP: This FEP is concerned with the selective precipitation of dissolved uranium species as minerals from water percolating through this repository. The precipitation reaction removes uranium from solution and could possibly increase the rate of fuel dissolution. This FEP is more closely related to the following primary FEPs: 2.1.02.02.00, CSNF Degradation, and 2.1.02.01.00, DSNF Degradation, which more appropriately addressed chemical processes affecting degradation.

Screening and Disposition: Same as primary FEPs 2.1.02.02.00, CSNF Degradation, and 2.1.02.01.00, DSNF degradation.

FEP Number and Name: 2.1.09.04.06, Precipitation (Release/Migration Factors)

Relationship to Primary FEP: This FEP is concerned with precipitation of secondary minerals and its effect on contaminant transport, which is the topic of primary FEP 2.1.09.10.00, Secondary Phase Effects on Dissolved Radionuclide Concentrations at the Waste Form. The FEP is redundant and is retained in the FEP list for completeness.

Screening and Disposition: Same as FEP 2.1.09.10.00.

FEP Number and Name: 2.1.09.04.07, Speciation Control of Contaminants by Hyperalkaline Plume Formed in the EBS

Relationship to Primary FEP: This FEP is concerned with the fact that speciation of the contaminant solutes, and therefore solubility, is controlled by the chemical properties of the carrier plume. This is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.08, Solubility within Fuel Matrix

Relationship to Primary FEP: This FEP is concerned with the dependence of solubility of dissolved species on water chemistry, which is appropriately addressed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.09, Solubility and Precipitation (Contaminant Speciation and Solubility)

Relationship to Primary FEP: Redundant subtopics of primary FEP retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.10, Solubility Limit (Contaminant Speciation and Solubility)

Relationship to Primary FEP: Redundant subtopics of the primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.11, Radionuclide Source Term (Contaminant Speciation and Solubility)

Relationship to Primary FEP: Rewording of primary FEP and therefore redundant. It is retained in the FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.12, Elemental Solubility/Precipitation (Contaminant Speciation and Solubility)

Relationship to Primary FEP: Redundant subtopic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.04.13, Speciation (Contaminant Speciation and Solubility)

Relationship to Primary FEP: Redundant subtopic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-26. In-Package Sorption—YMP No. 2.1.09.05.00 from Section 6.2.26

FEP Number and Name: 2.1.09.05.01, Selective Sorption of Pu from Solution

Relationship to Primary FEP: Selective sorption of Pu is a specific case that is adequately addressed in the broader context of the primary FEP.

Screening and Disposition: Conservatively excluded as discussed in the primary FEP.

FEP Number and Name: 2.1.09.05.02, Sorption

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.05.03, Radionuclide Retardation

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.05.04, Sorption on Filling Material

Relationship to Primary FEP: Sorption on filling material is a subtopic that is adequately addressed in the broader contest of the primary FEP.

Screening and Disposition: Same as the primary FEP.

V-27. Reduction-Oxidation Potential in Waste and EBS—YMP No. 2.1.09.06.00 from Section 6.2.27

FEP Number and Name: 2.1.09.06.01, Redox Front (in Waste and EBS)

Relationship to Primary FEP: Redox front is a subtopic adequately addressed by primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.06.02, Reduction-Oxidation Fronts (in Waste and EBS)

Relationship to Primary FEP: Redundant topic of secondary FEP 2.1.09.06.01, Redox Front in Waste and EBS, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.06.03, Localized Reducing Zones (in Waste and EBS)

Relationship to Primary FEP: Localized-reducing zones is a subtopic that is appropriately addressed in broader context of primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.06.04, Redox Front (in Buffer/Backfill)

Relationship to Primary FEP: Redundant subtopic of secondary FEP 2.1.09.06.01, Redox Front (in Waste and EBS).

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.06.05, Fe Control of Oxidation State of Contaminants

Relationship to Primary FEP: It is well documented that iron corrosion reduces actinide, technetium, and other radionuclides to lower, less soluble oxidation states.

Screening and Disposition: Same as the primary FEP.

V-28. Reaction Kinetics in Waste and EBS—YMP No. 2.1.09.07.00 from Section 6.2.28

FEP Number and Name: 2.1.09.07.01, Chemical Kinetics (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-29. Chemical Gradients/Enhanced Diffusion in Waste and EBS—YMP No. 2.1.09.08.00 from Section 6.2.29

FEP Number and Name: 2.1.09.08.01, Enhanced Diffusion (in Waste and EBS)

Relationship to Primary FEP: Redundant subtopic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.08.02, Chemical Gradients (in Waste and EBS)

Relationship to Primary FEP: Redundant subtopic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.08.03, Diffusion In and Through Failed Canister

Relationship to Primary FEP: This FEP is concerned with solute-concentration gradient due to a diffusion from a breached waste package. This FEP could also be mapped to 2.1.08.07.00, Pathways for Unsaturated Flow and Transport, which concludes that the pathways through the EBS can be modeled as a series of mixing cells. The assumed instantaneous transport of radionuclides to the invert through a breach in the waste package can occur as either a diffusive or advective flux, depending on the saturated/unsaturated flow conditions (CRWMS M&O 2000f, Section 6.3).

Screening and Disposition: Although enhanced diffusion has been excluded by the primary FEP, diffusion from a failed waste package is included as a possible process that quantifies the instantaneous transport between the waste package and the invert as discussed in primary FEP 2.1.08.07.00, Pathways for Unsaturated Flow and Transport.

V-30. Electrochemical Effects (Electrophoresis, Galvanic Coupling) in Waste and EBS—YMP No. 2.1.09.09.00 from Section 6.2.30

FEP Number and Name: 2.1.09.09.01, Repository Induced Pb/Cu Electrochemical Reactions

Relationship to Primary FEP: Specific process that is subsumed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.02, Natural Telluric Electrochemical Reactions (in Waste and EBS)

Relationship to Primary FEP: Specific process that is subsumed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.03, Electro-Chemical Cracking (in Waste and EBS)

Relationship to Primary FEP: Specific process that is subsumed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.04, Electrochemical Effects/Gradients (in Waste and EBS)

Relationship to Primary FEP: Specific feature that is subsumed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.05, Electrochemical Effects of Metal Corrosion

Relationship to Primary FEP: Specific feature that is subsumed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.06, Electrochemical Effects (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.07, Galvanic Coupling (in Waste and EBS)

Relationship to Primary FEP: Redundant subcategory of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.08, Electrophoresis (in Waste and EBS)

Relationship to Primary FEP: Redundant subcategory of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.09, Electrochemical Gradients (in Waste and EBS)

Relationship to Primary FEP: Specific feature that is subsumed in the broader context of the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.10, Galvanic Coupling (in Waste and EBS)

Relationship to Primary FEP: Redundant subtopic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.09.11, Galvanic Coupling (in Waste and EBS)

Relationship to Primary FEP: Redundant subtopic of primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-31. Secondary Phase Effects on Dissolved Radionuclide Concentrations at the Waste Form—YMP No. 2.1.09.10.00 from Section 6.2.31

No Secondary FEPs.

V-32. Waste-Rock Contact—YMP No. 2.1.09.11.00 from Section 6.2.32

No Secondary FEPs.

V-33. Rind (Altered Zone) Formation in Waste, EBS, and Adjacent Rock—YMP No. 2.1.09.12.00 from Section 6.2.33

FEP Number and Name: 2.1.09.12.01, Deep Alteration of the Porosity of Drift Walls

Relationship to Primary FEP: This FEP should more appropriately be placed in a related NFE primary FEP.

Screening and Disposition: See NFE treatment of this secondary FEP.

V-34. Complexation by Organics in Waste and EBS—YMP No. 2.1.09.13.00 from Section 6.2.34

FEP Number and Name: 2.1.09.13.01, Methylation (in Waste and EBS)

Relationship to Primary FEP: Specifically addressed in the primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.02, Humic and Fulvic Acids

Relationship to Primary FEP: Specifically addressed in primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.03, Complexation by Organics

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.04, Fulvic Acid

Relationship to Primary FEP: Redundant subtopic of secondary FEP 2.1.09.12.02, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.05, Humic Acid

Relationship to Primary FEP: Redundant subtopic of secondary FEP 2.1.09.12.02, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.06, Complexing Agents

Relationship to Primary FEP: Subtopic that is appropriately addressed in broader context of primary FEP.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.07, Organics (Complexing Agents)

Relationship to Primary FEP: Redundant subtopic of secondary FEP 2.1.09.13.06, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.08, Organics (Complexing Agents)

Relationship to Primary FEP: Redundant topic of secondary FEP 2.1.09.13.07, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.09, Organic Complexation

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.10, Organic Ligands

Relationship to Primary FEP: Redundant topic of secondary FEP 2.1.09.13.07, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.09.13.11, Kinetics of Organic Complexation

Relationship to Primary FEP: Specific topic that is appropriately addressed in primary FEP.

Screening and Disposition: Excluded because complexation by organics is excluded as discussed in primary FEP.

FEP Number and Name: 2.1.09.13.12, Introduced Complexing Agents

Relationship to Primary FEP: Redundant subtopic of secondary FEP 2.1.09.13.06, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-35. Heat Output/Temperature in Waste and EBS—YMP No. 2.1.11.01.00 from Section 6.2.35

FEP Name and Number: 2.1.11.01.01, Glass Temperature (in Waste and EBS)

Relationship to Primary FEP: The temperature of the glass at the reference disposal time (40 years after removal from nuclear reactors) is calculated to be about 195°C (max. 150°C at the bentonite-canister interface [this FEP is written for the NAGRA program]), which falls to about 100°C after 100 years.

Screening and Disposition: Specifics of this secondary FEP (temperature of the glass and bentonite at the reference disposal time) are relevant to the NAGRA waste-isolation program and not relevant to Yucca Mountain; therefore, it is excluded. With respect to Waste Form Degradation Model, subsumed in primary FEP argument.

FEP Name and Number: 2.1.11.01.02, Canister Temperature

Relationship to Primary FEP: Canister (i.e., waste package) temperature will be determined by the thermal conductivity of the backfill, the radiogenic heat output of the glass and the ambient rock temperature. Since the temperature transient is relatively short, its effect on corrosion can be ignored, etc. (NAGRA).

Screening and Disposition: Thermal conductivity of the backfill is relevant to the NAGRA waste-isolation program and not relevant to Yucca Mountain; therefore, it is excluded. All other aspects of this FEP are subsumed in the primary FEP argument. The dependence of glass-degradation temperature on other factors is included in the TSPA-SR. Details are described in CRWMO M&O (2000d), *Defense High Level Waste Glass Degradation*.

FEP Name and Number: 2.1.11.01.03, Temperature, Bentonite Buffer

Relationship to Primary FEP: In the bentonite backfill, there will be temperature changes with respect to time and location, thus producing time-varying temperature gradients.

Screening and Disposition: Specifics of this secondary FEP are relevant to waste-isolation programs using backfill and buffering agents, and not Yucca Mountain, which does not use buffers or backfill. Therefore, it is not relevant to the Waste Form Degradation Model and excluded. This secondary FEP is retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.11.01.04, Temperature, Canister

Relationship to Primary FEP: The temperature in the canister (i.e., waste package) is influenced by the heat generated by radioactive decay in the fuel, the thermal properties of the canister and the temperature of the surrounding bentonite.

Screening and Disposition: Specifics of this secondary FEP are relevant to waste-isolation programs using backfill and buffering agents, and not Yucca Mountain which does not use buffers or backfill. All other aspects of this FEP are subsumed in primary FEP argument.

FEP Name and Number: 2.1.11.01.05, Temperature, Tunnel Backfill

Relationship to Primary FEP: Refers to changes in temperature with time, and temperature gradients in the tunnel backfill.

Screening and Disposition: Specifics of this secondary FEP are relevant to waste-isolation programs using backfill and buffering agents, and not Yucca Mountain, which does not use buffers or backfill. Therefore, it is not relevant to Waste Form Degradation Model and excluded. This secondary FEP is retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.11.01.06, Heat Generation from Waste Containers

Relationship to Primary FEP: Heat is generated as a consequence of radioactive decay of the radionuclides in a waste container. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.11.01.07, Radioactive Decay Heat

Relationship to Primary FEP: The radioactive decay of the fuel in the repository is well known. It is modeled in the fuel model, which tracks the time-dependent isotope inventory in the fuel matrix, etc.

Screening and Disposition: Specifics of this secondary FEP (tracking the time-dependent isotope inventory in the fuel matrix) are relevant to the Swedish waste-isolation program numerical and conceptual model. Therefore, this FEP is not relevant to Waste Form Degradation Model and excluded. Retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.11.01.08, DOE SNF Expected Waste Heat Generation

Relationship to Primary FEP: The DOE SNF to be disposed of in Yucca Mountain may have heat-generation rates that are different from the heat-generation rates in the commercial SNF to be disposed. Differences in heat-generation rates could impact thermal-hydrology and seepage onto the waste packages.

Screening and Disposition: Subsumed in primary FEP argument.

FEP Name and Number: 2.1.11.01.09, DOE SNF Expected Waste Heat Generation

Relationship to Primary FEP: The DOE SNF to be disposed of in Yucca Mountain may have heat-generation rates that are different from the heat-generation rates in the commercial SNF to be disposed. Differences in heat-generation rates could impact thermal-hydrology and seepage onto the waste packages. This FEP was obtained from Reference 1 and is identified as MLD-6 in that reference.

Screening and Disposition: Redundant to secondary FEP 2.1.11.01.08, retained in FEP list for completeness.

V-36. Exothermic Reactions and other Thermal Effects in Waste and EBS—YMP No. 2.1.11.03.00 from Section 6.2.36

FEP Number and Name: 2.1.11.03.01, Concrete Hydration

Relationship to Primary FEP: Redundant process addressed in primary FEP, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-37. Temperature Effects/Coupled Processes in Waste and EBS—YMP No. 2.1.11.04.00 from Section 6.2.37

FEP Name and Number: 2.1.11.04.01, Thermal (Processes)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.11.04.02, Temperature Effects (Unexpected Effects) (in Waste and EBS)

Relationship to Primary FEP: Local temperature increases could be large with unexpected effects such as: formation of molten materials, very high stresses, percolation of groundwater, formation of an ore body, and fire or explosion.

Screening and Disposition: With respect to waste-form degradation, this FEP is subsumed in primary FEP argument. Processes other than waste-form degradation that are related to Yucca Mountain are addressed in the EBS conceptual model, *Multiscale Thermohydrologic Model* (CRWMS M&O 2000t).

FEP Name and Number: 2.1.11.04.03, Heat from Radioactive Decay (in Waste and EBS)

Relationship to Primary FEP: Radioactive decay of the waste emplaced in the repository will generate heat. The importance of heat from radioactive decay depends on the effects that induced temperature changes would have on mechanics, fluid flow, and geochemical processes.

Screening and Disposition: With respect to the Waste Form Degradation Model, subsumed in primary FEP argument. Processes other than waste-form degradation (mechanics, fluid flow, and geochemical processes) are addressed in EBS conceptual model, *Multiscale Thermohydrologic Model* (CRWMS M&O 2000t).

FEP Name and Number: 2.1.11.04.04, Long-Term Transients (in Waste and EBS)

Relationship to Primary FEP: Long-term transient events, singly or coupled, may affect contaminant transport. These effects include the thermal pulse due to decaying fuel, formation of steam, reflooding, excavation stress relief, and altered flows due to grouting.

Screening and Disposition: With respect to the Waste Form Degradation Model, subsumed in primary FEP argument. Processes other than waste-form degradation (formation of steam, reflooding, excavation stress relief and altered flows, etc.) are addressed in EBS conceptual model, *Multiscale Thermohydrologic Model* (CRWMS M&O 2000t).

FEP Name and Number: 2.1.11.04.05, Time Dependence (in Waste and EBS)

Relationship to Primary FEP: There will be time-dependent effects from factors such as the thermal pulse due to decaying fuel and reflooding of the repository. These effects should be reflected in the flow fields and diffusive coefficients and the models in which they are used.

Screening and Disposition: With respect to the Waste Form Degradation Model, subsumed in primary FEP argument. Processes other than waste-form degradation (parameter changes in flow fields) are addressed in the EBS conceptual model.

FEP Name and Number: 2.1.11.04.06, Coupled Processes (in Waste and EBS)

Relationship to Primary FEP: Many coupled processes and irreversible thermodynamics may affect contaminant transport in the backfill and buffer.

Screening and Disposition: Specifics of this secondary FEP are relevant to waste isolation programs using buffering and backfill agents, and are not relevant to Yucca Mountain; therefore, the FEP is Excluded but retained in the Yucca Mountain FEP database for completeness.

V-38. Differing Thermal Expansion of Repository Components—YMP No. 2.1.11.05.00 from Section 6.2.38

FEP Name and Number: 2.1.11.05.01, Differential Thermal Expansion of Near-Field Barriers

Relationship to Primary FEP: Differential thermal expansion of surrounding rocks creates stresses that shear waste packages.

Screening and Disposition: Subsumed in the primary FEP argument.

FEP Name and Number: 2.1.11.05.02, Shearing of Waste Containers by Secondary Stresses from Thermal Expansion of the Rock

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Subsumed in the primary FEP argument.

FEP Name and Number: 2.1.11.05.03, Differential Elastic Response (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Subsumed in the primary FEP argument.

FEP Name and Number: 2.1.11.05.04, Non-Elastic Response (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-39. Thermally Induced Stress Changes in Waste and EBS—YMP No. 2.1.11.07.00 from Section 6.2.39

FEP Name and Number: 2.1.11.07.01, Changes in *In Situ* Stress Field (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.11.07.02, Stress Field Changes, Settling, Subsidence or Caving

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: With respect to the Waste Form Degradation Model, subsumed in primary FEP argument. Processes other than waste-form degradation such as settling, subsidence, and caving are addressed in EBS conceptual model, *Multiscale Thermohydrologic Model* (CRWMS M&O 2000t).

V-40. Thermal Effects: Chemical and Microbiological Changes in the Waste and EBS—YMP No. 2.1.11.08.00 from Section 6.2.40

No Secondary FEPs.

V-41. Thermal Effects on Liquid or Two-Phase Fluid Flow in the Waste and EBS—YMP No. 2.1.11.09.00 from Section 6.2.41

FEP Number and Name: 2.1.11.09.01: Convection Effects on Transport (Enhanced Vapor Diffusion)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.11.09.02: Multiphase Flow and Gas-Driven Transport (Water Transport)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-42. Thermal Effects on Diffusion (Soret Effect) in Waste and EBS—YMP No. 2.1.11.10.00 from Section 6.2.42

FEP Name and Number: 2.1.11.10.01, Soret Effect (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.11.10.02, Thermal Effects: Transport (Diffusion) Effects (in Waste and EBS)

Relationship to Primary FEP: The thermal pulse generated in the intermediate-level waste may cause diffusion rates and, consequently, radionuclide-transport rates to increase. This would only be significant if a diffusive barrier were still operating in the near field.

Screening and Disposition: Specifics of this secondary FEP (emplacement of intermediate-level waste and/or diffusive barriers operating in the near-field) are relevant to the United Kingdom waste-isolation program and not Yucca Mountain. Therefore, this secondary FEP is excluded and retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.11.10.03, Soret Effect (Water Transport)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-43. Gas Generation—YMP No. 2.1.12.01.00 from Section 6.2.43

FEP Name and Number: 2.1.12.01.01, Formation of Gases (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.12.01.02, Gas generation

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.12.01.03, Gas Generation, Buffer/Backfill

Relationship to Primary FEP: Radiolytic decomposition of water in the vicinity of the canister may lead to hydrogen formation in the buffer, and microbial degradation of organics may lead to generation of hydrogen, carbon dioxide, and methane in the bentonite buffer and in the backfill.

Screening and Disposition: Specifics of this secondary FEP (generation of hydrogen, carbon dioxide and methane in the bentonite buffer and in the backfill) are relevant to Swedish waste-isolation program and not Yucca Mountain; therefore, the FEP is excluded and retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.12.01.04, Chemotoxic Gases (in Waste and EBS)

Relationship to Primary FEP: Production of chemotoxic gases may influence other processes in the repository near-field, far-field, or biosphere with radiological consequences. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.12.01.05, Pressurization (in Waste and EBS)

Relationship to Primary FEP: Increased gas pressure may slow the rate of salt creep. Closure and consolidation can be slowed by fluid pressure in the repository.

Screening and Disposition: This FEP is specific to the repositories placed in salt (such as Waste Isolation Pilot Plant [WIPP]) and not relevant to Yucca Mountain, therefore, the FEP is excluded and retained in the Yucca Mountain FEP database for completeness.

V-44. Gas Generation (He) from Fuel Decay—YMP No. 2.1.12.02.00 from Section 6.2.44

FEP Name and Number: 2.1.12.02.01, Helium Gas Production

Relationship to Primary FEP: Within the WIPP, helium-gas production will occur by the reduction of alpha-particles (helium nuclei) emitted from the waste. The maximum amount of helium that could be produced can be estimated by assuming that all of the alpha particles generated during radioactive decay are converted to helium gas.

Screening and Disposition: This FEP is specific to the WIPP conceptual model. FEP issue, helium-gas production, subsumed in the primary FEP. Retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.12.02.02, Internal Pressure (in Waste and EBS)

Relationship to Primary FEP: After about 1 million years the He production in the spent fuel will have caused a higher internal pressure than the surrounding hydrostatic pressure and the swelling pressure from the bentonite.

Screening and Disposition: This FEP is specific to the Swedish conceptual model and not relevant to Yucca Mountain; therefore, the FEP is excluded and retained in FEP list for completeness.

FEP Name and Number: 2.1.12.02.03, Gas Generation, Canister

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.12.02.04, Internal Pressure (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.12.02.05, He Gas Production (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-45. Gas Generation (H_2) from Metal Corrosion—YMP No. 2.1.12.03.00 from Section 6.2.45

FEP Name and Number: 2.1.12.03.01, Chemical Effects of Corrosion

Relationship to Primary FEP: The corrosion process may produce hydrogen constituents that further enhance corrosion and/or gas-generation rates.

Screening and Disposition: Same as the primary FEP. The effects of hydrogen production that would enhance further corrosion and gas-generation rates are subsumed in the primary FEP screening argument.

FEP Name and Number: 2.1.12.03.02, Effect of Hydrogen on Corrosion

Relationship to Primary FEP: The production of hydrogen by waste-package corrosion could enhance corrosion rates on uncorroded waste packages.

Screening and Disposition: The effects of hydrogen production on waste-package corrosion is subsumed in the primary FEP screening argument. Redundant to 2.1.12.03.01, Chemical Effects of Corrosion. Retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.12.03.03, Hydrogen Production (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.12.03.04, Hydrogen Production by Metal Corrosion

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-46. Gas Generation (CO_2, CH_4, H_2S) from Microbial Degradation—YMP No. 2.1.12.04.00 from Section 6.2.46

FEP Name and Number: 2.1.12.04.01, Effect of Temperature on Microbial Gas Generation

Relationship to Primary FEP: Temperature variations in the waste, with time, may affect the amount of microbial gas generated.

Screening and Disposition: Subsumed in Primary FEPs YMP - 2.1.10.01.00, Biological Activity in Waste and EBS and 2.1.12.04.00, Gas Generation (CO_2, CH_4, H_2S) from Microbial Degradation. Excluded based on low consequence.

FEP Name and Number: 2.1.12.04.02, Effect of Pressure on Microbial Gas Generation

Relationship to Primary FEP: Microbial-gas generation within the waste package could vary as waste package pressures vary. Microbial-gas generation outside the waste package could vary as repository pressures vary.

Screening and Disposition: Excluded based on low consequence: pressure effects on in-package microbial gas generation rates, same as primary FEP. Excluded pressure effects on microbial-gas generation outside waste package, based on low consequence, same as primary FEP.

FEP Name and Number: 2.1.12.04.03, Effect of Radiation on Microbial Gas Generation

Relationship to Primary FEP: Alpha radiation within and outside the waste package could enhance microbial-gas generation.

Screening and Disposition: Excluded based on low consequence, argument subsumed in primary FEP.

FEP Name and Number: 2.1.12.04.04, Effect of Biofilms on Microbial Gas Generation

Relationship to Primary FEP: The location of microbial activity within the repository is likely to be controlled by the availability of substrates and nutrients. Biofilms may develop on surfaces where nutrients are concentrated. They consist of one or more layers of cells with extra polymeric material and serve to maintain an optimum environment for growth.

Screening and Disposition: Subsumed in the primary FEP.

FEP Name and Number: 2.1.12.04.05, Methane and Carbon Dioxide by Microbial Degradation

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-47. Gas Transport in Waste and EBS—YMP No. 2.1.12.06.00 from Section 6.2.47

FEP Number and Name: 2.1.12.06.01, Thermo-Chemical Effects (Related to Gas in Waste and EBS)

Relationship to Primary FEP: Gas generation may affect the thermal and chemical properties of the waste and waste packages, which could potentially cause an increase in expected doses.

Screening and Disposition: Same as the primary FEP above and primary FEP 2.1.12.03, Gas Generation from Metal Corrosion.

FEP Number and Name: 2.1.12.06.02, Gas Transport

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.12.06.03, Gas Effects (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.12.06.04, Gas Escape from Canister

Relationship to Primary FEP: Concerns the escape through the failed copper canister (i.e., waste package) of hydrogen gas formed in the gap between the copper canister and steel vessel as well as the escape of gas from the interior of the steel vessel after failure. This design is used for other waste-isolation programs but not for Yucca Mountain.

Screening and Disposition: This FEP is relevant to waste-isolation programs other than Yucca Mountain; therefore, it is excluded. Other aspects of this FEP screening subsumed in primary FEP argument.

FEP Number and Name: 2.1.12.06.05, Gas Flow and Transport, Buffer/Backfill

Relationship to Primary FEP: Gas flow in bentonite occurs when the gas pressure exceeds the threshold value required to open up gas channels in the material.

Screening and Disposition: There will be no buffer/backfill placed in the repository; therefore, not relevant to Yucca Mountain repository design and excluded. Retained in FEP list for completeness.

FEP Number and Name: 2.1.12.06.06, Gas Transport

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Number and Name: 2.1.12.06.07, Unsaturated Flow Due to Gas Production (in Waste and EBS)

Relationship to Primary FEP: Gas generation in the waste and/or EBS may produce preferential flow paths from the repository to the accessible environment.

Screening and Disposition: FEP screening subsumed in primary FEP argument.

FEP Number and Name: 2.1.12.06.08, Gas Permeability (in Buffer/Backfill)

Relationship to Primary FEP: Hydrogen generation rates (for the NAGRA nuclear-repository program) are estimated to be in the range 0.02 to 0.2 moles/m^2/yr. Aqueous hydrogen diffusion at the canister (i.e., waste package) is estimated at about 0.2 moles/yr/canister (0.02 moles/m^2/yr). If a gas phase forms around the canister (waste package) surface, this in itself is likely to reduce the rate of corrosion and the rate of production of further hydrogen. It may be necessary to consider special design features, such as the incorporation of a sand capillary-breaking layer to provide additional buffer volume and surface area for hydrogen dissolution.

At low gas pressures, bentonite permeability is low but can increase by 100 times, due to creation of "channeling" through connected pore spaces to around 30 to 70% of swelling pressure. Gas can overcome the bentonite capillary forces and hydrostatic pressure and pass through larger connected capillary pores with typical diameters of 10 nm. Thus, it is likely that no significant physical disturbance of the clay is involved, and no degradation of the bentonite hydraulic properties or colloid retention properties is expected due to the passage of the gas.

Screening and Disposition: There will be no buffer/backfill placed in the repository. Retained on FEP list for completeness. Specifics of this secondary FEP are relevant to NAGRA waste isolation programs and not Yucca Mountain. Therefore, it is excluded, not relevant to Yucca Mountain repository design.

V-48. Radioactive Gases in Waste and EBS—YMP No. 2.1.12.07.00 from Section 6.2.48

FEP Name and Number: 2.1.12.07.01, Radioactive Gas (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.12.07.02, Gaseous and Volatile Isotopes

Relationship to Primary FEP: Some radionuclides may be isotopes of gaseous elements or may form volatile compounds. Initial inventories of gaseous radionuclides, e.g., Kr isotopes, will be lost during vitrification. Some radionuclides, e.g., isotopes of I and Se are known to be capable of forming (organic) volatile compounds, but this is unlikely.

Screening and Disposition: Specifics of the FEP are not relevant to Yucca Mountain and is excluded. This FEP is specific to the NAGRA waste-isolation program and is retained in the Yucca Mountain FEP database for completeness.

V-49. Gas Explosions—YMP No. 2.1.12.08.00YMP from Section 6.2.49

FEP Name and Number: 2.1.12.08.01, H2/O2 Explosions (in Waste and EBS)

Relationship to Primary FEP: Gases may form by alpha and beta radiolysis. The probability that they gather in a gaseous and explosive composition, and in a cavity in the repository area, are judged to be small. The fuel itself, the canister or its corrosion products, the backfill and surrounding rock and groundwater, will in general act as a reductant and consume the oxygen.

Screening and Disposition: This FEP is specific to the repository design of the NAGRA waste-isolation program (corrosion will act as a reductant and consume the oxygen), not relevant to Yucca Mountain but is retained in the Yucca Mountain FEP database for completeness. The main FEP issue, gas explosions in the repository, is subsumed in the primary FEP.

FEP Name and Number: 2.1.12.08.02, Flammability (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.12.08.03, Explosions

Relationship to Primary FEP: Explosions within the vault, and bomb blasts (military ordinances) at the land surface, could affect the performance of the vault.

Screening and Disposition: With respect to waste-form degradation, this FEP is subsumed in primary FEP argument. Vault (drift) integrity due to explosions is addressed in the NFE and EBS models.

FEP Name and Number: 2.1.12.08.04, Explosion

Relationship to Primary FEP: Explosions near the disposal system, or in the disposal system during its operation, may affect the performance of the geosphere (AECL).

Screening and Disposition: With respect to waste-form degradation, this FEP is subsumed in primary FEP argument. Explosions affecting the disposal system and geosphere issues are addressed in the NFE and EBS models.

V-50. Radiolysis—YMP No. 2.1.13.01.00 from Section 6.2.50

FEP Name and Number: 2.1.13.01.01, Radiolysis (in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP; see detailed discussion given in Attachment I of CRWMS M&O (2000i), *In-Package Chemistry Abstraction.*

FEP Name and Number: 2.1.13.01.02, Radiolysis

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP; see detailed discussion given in Attachment I of CRWMS M&O (2000i), *In-Package Chemistry Abstraction.*

FEP Name and Number: 2.1.13.01.03, Radiolysis (in Waste and EBS)

Relationship to Primary FEP: Following canister (i.e., waste package) failure, hydrogen will also be formed by radiolytic decomposition of water close to the HLW glass. Assuming no recombination of radiolysis products, the maximum rate of production is estimated at about 0.01 moles/yr/canister, at 1000 years and will decline as the radioactivity of the waste decreases.

Screening and Disposition: This FEP is specific to the NAGRA waste-isolation program, (bounding H_2 production rate not adopted by Yucca Mountain) but is retained in the Yucca Mountain FEP database for completeness. The main issue, radiolysis of water, is subsumed in the primary FEP. See detailed discussion given in Attachment I of CRWMS M&O (2000i), *In-Package Chemistry Abstraction.*

FEP Name and Number: 2.1.13.01.04, Radiolysis (in waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP, see detailed discussion given in Attachment I of CRWMS M&O (2000i), *In-Package Chemistry Abstraction.*

FEP Name and Number: 2.1.13.01.05, Radiolysis Prior to Wetting (in Waste and EBS)

Relationship to Primary FEP: If waste-package interiors have a relatively high humidity, radiolysis of water molecules within the waste package could produce nitrous and nitric acids, which could cause the waste package to corrode from the inside out.

Screening and Disposition: Excluded secondary FEP argument subsumed in the primary FEP.

FEP Name and Number: 2.1.13.01.06, Radiolysis of Brine

Relationship to Primary FEP: Radiolysis of brine in the WIPP disposal rooms, and of water in the waste, will lead to production of gases and may significantly affect the oxygen content of the rooms.

Screening and Disposition: This FEP is specific to the WIPP conceptual model and not relevant to Yucca Mountain Waste Form Degradation Model. It is excluded on low consequence but is retained in the Yucca Mountain FEP database for completeness.

FEP Name and Number: 2.1.13.01.07, Radiolysis of Cellulose (in Waste and EBS)

Relationship to Primary FEP: Experimental data on radiolytic gas production of cellulose is compared with bacterial (microbial) gas-production rates on cellulose and other waste materials.

Screening and Disposition: This FEP is specific to gas-generation processes in the WIPP repository but is retained in the Yucca Mountain FEP database for completeness. By design, no cellulosic waste will be emplaced in the Yucca Mountain Repository; therefore, this FEP excluded on low consequence.

FEP Name and Number: 2.1.13.01.08, Radiolysis

Relationship to Primary FEP: Radiolysis due to gamma and neutron radiation is possible as long as the container is intact. It will have some impact on water chemistry in the vicinity of the canister (i.e., waste package). Alpha and beta radiolysis, occurring up to .03 mm from the fuel pellets, will be of importance after canister failure, when water comes in contact with the fuel matrix.

Screening and Disposition: Specifics of this secondary FEP (alpha and beta radiolysis, occurring up to .03 mm from the fuel pellets) are relevant to Swedish repository program. The main issue, radiolysis due to gamma and neutron radiation, is subsumed in primary FEP argument.

FEP Name and Number: 2.1.13.01.09, Radiolysis

Relationship to Primary FEP: Radiolysis may change the chemical environment in the vault (repository drift), locally affecting the electrochemical potential (Eh), acidity (pH), and concentration of reactive radicals, with consequent impacts on buffer, backfill, waste matrices, and surrounding rock. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-51. Radiation Damage in Waste and EBS—YMP No. 2.1.13.02.00 from Section 6.2.51

FEP Name and Number: 2.1.13.02.01, Radiation Effects (in Waste and EBS)

Relationship to Primary FEP: Radiation damage and radiolysis products could affect the properties of the rock, including structural integrity, sorptive capabilities, and the chemistry of the groundwater.

Screening and Disposition: Radiation damage in host rock is relevant to EBS and NFE models and not waste-form degradation; therefore, it is excluded on low consequence.

FEP Name and Number: 2.1.13.02.02, Radiation Effects on Bentonite

Relationship to Primary FEP: If a bentonite backfill is used, its properties could be affected by radiation from dissolved fuel passing through or depositing on the bentonite clay.

Screening and Disposition: Bentonite backfill will not be used in Yucca Mountain; therefore, this FEP is not relevant to Yucca Mountain and excluded. Retained in FEP list for completeness.

FEP Name and Number: 2.1.13.02.03, Material Property Changes (Due to Radiation in Waste and EBS)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.13.02.04, Radiation Damage (in Waste and EBS)

Relationship to Primary FEP: Alpha doses expected in the HLW-glass cause volume changes of less than 1%, which have no effect on the mechanical integrity of the glass.

Screening and Disposition: Subsumed in primary FEPs 2.1.02.03, Glass Degradation, Alteration, and Dissolution, and 2.1.02.03, Glass Recrystallization.

FEP Name and Number: 2.1.13.02.05, Radiation Shielding (in Waste and EBS)

Relationship to Primary FEP: The waste package acts as a radiation shield so that radiolysis outside the waste package prior to canister failure is negligible. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.13.02.06, Radiation Effects on Buffer/Backfill

Relationship to Primary FEP: If a bentonite buffer is used as backfill, exposure to radiation could change the mineralogy of the buffer/backfill.

Screening and Disposition: Bentonite backfill will not be used in Yucca Mountain. Therefore, this FEP is not relevant to Yucca Mountain and excluded. Retained in FEP list for completeness.

FEP Name and Number: 2.1.13.02.07, Radiation Effects on Canister

Relationship to Primary FEP: Radiation to the canister (i.e., waste package) may lead to brittleness of the canister material, which could affect its mechanical properties. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.13.02.08, Radiological Effects on Waste

Relationship to Primary FEP: Strong radiation fields could lead to effects on the physical properties of the waste matrices. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.13.02.09, Radiological Effects on Containers

Relationship to Primary FEP: Ionizing radiation can change the physical properties of the metal containers. Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 2.1.13.02.10, Radiological Effects on Seals

Relationship to Primary FEP: Ionizing radiation can change the physical properties and crystalline structure in the repository seals.

Screening and Disposition: This FEP is specific to the WIPP repository design and PA implementation and was retained in the Yucca Mountain FEP database for completeness. This FEP is not relevant to Yucca Mountain and excluded.

FEP Name and Number: 2.1.13.02.11, Radiation Effects on Canister

Relationship to Primary FEP: Radiation may lead to brittleness of any copper material.

Screening and Disposition: This FEP is specific to the Swedish nuclear waste program. There will no copper in the Yucca Mountain waste packages. Therefore, this FEP is not relevant to Yucca Mountain and excluded. Retained in the Yucca Mountain FEP database for completeness.

V-52. Use of J-13 Well Water as a Surrogate for Water Flowing into the EBS and Waste—YMP No. 2.2.08.12.00 from Section 6.2.52

No Secondary FEPs.

V-53. Radioactive Decay and Ingrowth—YMP No. 3.1.01.01.00 from Section 6.2.53

<u>FEP Name and Number: 3.1.01.01.01, Radioactive Decay</u>

Relationship to Primary FEP: Radioactive decay and ingrowth will affect concentrations of radionuclides in the waste matrices. Redundant, retained in FEP list for completeness.

Screening and Disposition: Redundant, retained in FEP list for completeness.

<u>FEP Name and Number: 3.1.01.01.02, Radioactive Decay</u>

Relationship to Primary FEP: Radioactive decay and ingrowth will affect the movement and concentrations of contaminants in the geosphere. Redundant, retained in FEP list for completeness.

Screening and Disposition: This FEP is addressed in the context of the primary FEP and is retained in FEP list for completeness.

<u>FEP Name and Number: 3.1.01.01.03, Radioactive Decay</u>

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Redundant. This FEP is addressed in the context of the primary FEP and is retained in FEP list for completeness.

<u>FEP Name and Number: 3.1.01.01.04, Radioactive Decay and Ingrowth</u>

Relationship to Primary FEP: Redundant, same as secondary FEPs 3.1.01.01.01, Radioactive Decay, 3.1.01.01.02, Radioactive Decay, and 3.1.01.01.03, Radioactive Decay, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

<u>FEP Name and Number: 3.1.01.01.05, Radioactive Decay</u>

Relationship to Primary FEP: Radionuclides decay to form daughter products at characteristic rates. Decay schemes and half-lives are relatively well established, but there may be some variation in exact data used between different models, etc.

Screening and Disposition: The variation in decay rates is relatively small compared to the regulatory time frames being considered. Therefore, this FEP is screened out on low consequence.

FEP Name and Number: 3.1.01.01.06, Radioactive Decay

Relationship to Primary FEP: The radioactive decay of the fuel in the repository is well known. It changes the initial radionuclide content in the fuel with time and generates heat. The type of radiation generated by the decay depends on the radionuclide, and the penetrating distance of the radiation depends on the type of radiation, its energy and the surrounding medium.

Screening and Disposition: With respect to radionuclide inventory, this FEP is addressed in the context of the primary FEP and CRWMS M&O (2000j), *Inventory Abstraction*. Issues related to heat generation due to radionuclide decay are included in the TSPA-SR; their implementation is addressed in the following AMRs: *Multiscale Thermohydrologic Model* (CRWMS M&O 2000t); *In-Drift Thermal-Hydrological-Chemical Model* (CRWMS M&O 2000s); *Mountain-Scale Coupled Processes (TH) Models* (CRWMS M&O 2000af); and *Heat Decay Data and Repository Footprint for Thermal-Hydrologic and Conduction Only Models for TSPA-SR* (CRWMS M&O (2000aj).

FEP Name and Number: 3.1.01.01.07, Radioactive Decay of Mobile Nuclides

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: This FEP is addressed in the context of the primary FEP.

FEP Name and Number: 3.1.01.01.08, Radionuclide Decay and Ingrowth

Relationship to Primary FEP: The PA code (for WIPP), NUTS, calculates the overall movement and decay of radionuclides in the repository and disposal system.

Screening and Disposition: The overall movement and decay of radionuclides in the repository and disposal system, etc., for WIPP's 1996 Compliance Certification Application (CCA) was calculated using the numerical code NUTS. This FEP is specific to the 1996 WIPP PA implementation and was retained in the Yucca Mountain FEP database for completeness. The numerical code used for WIPP's CCA is irrelevant to the Yucca Mountain Project. The FEP issue, radionuclide decay and ingrowth, is addressed in the context of the primary FEP.

FEP Name and Number: 3.1.01.01.09, Radiological Events and Processes

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

V-54. Isotopic Dilution—YMP No. 3.2.07.01.00 from Section 6.2.54

FEP Name and Number: 3.2.07.01.01, Mass, Isotopic and Species Dilution

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.

FEP Name and Number: 3.2.07.01.02, Natural Radionuclides/Elements (in Host Rock Disturbed Zone)

Relationship to Primary FEP: Redundant, retained in FEP list for completeness.

Screening and Disposition: Same as the primary FEP.