OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ANALYSIS/MODEL COVER SHEET

1. QA: QA

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

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Attachment II – ~~5~~ 4 pages RM 05/11/2000
Attachment III – 2 pages
Attachment IV – ~~13~~ 12 pages RM 05/11/2000

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Per Section 5.5.6 of AP-3.10Q, the responsible manager has determined that the subject AMR is not subject to AP-2.14Q review because the analysis does not affect a discipline or area other than the originating organization (Performance Assessment). Therefore, no formal AP-2.14Q reviews were requested or determined to be necessary. However, both the upstream supplier and downstream customers were given the opportunity to provide informal comments on a draft copy. The majority of the informal comments were addressed in the AMR REV 00. Additionally, the Lead/Supervisor of this abstraction AMR also worked closely in its development with the Waste Package Materials Analysis Section Lead (Christine Stockman), an upstream supplier of inputs to this AMR. Further, the Waste Form PMR, which summarizes the results of this AMR, was previously subjected to a formal AP-2.14Q review. Finally, the downstream user of the information resulting from this AMR is Performance Assessment (PA), which is also the originating organization of this work. Therefore, no formal AP-2.14Q reviews were requested or determined to be necessary.

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OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ANALYSIS/MODEL REVISION RECORD

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ACRONYMS AND ABBREVIATIONS

ACC	accession number of an internal document produced by CRWMS M&O
AMR	Analysis/Modeling Report
ANL	Argonne National Laboratory
AP	absorber plate
ASHPLUME	code used to model direct volcanic effects
ATM	approved testing material
BWR	boiling-water reactor
CDSP	codisposal packages
CFR	Code of Federal Regulations
CLST	Container Life and Source Term
CR	control rod
CRWMS	Civilian Radioactive Waste Management System
CSNF	commercial spent nuclear fuel
DID	defense-in-depth (calculations)
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
FY	fiscal year
HIC	high-integrity can
HLW	high-level waste
IDGE	In-Drift Geochemical Environment (model)
INEEL	Idaho National Engineering and Environmental Laboratory
IPFW	immobilized Pu waste form
IRSR	Issue Resolution Status Report
IT	isomeric transition
KTI	Key Technical Issue
LA	license application
LADS	License Application Design Selection
LET	linear energy transfer
M&O	Management and Operating Contractor

ACRONYMS AND ABBREVIATIONS (Continued)

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
NFGE	Near-Field Geochemical Environment (model)
NRC	U.S. Nuclear Regulatory Commission
NSNFP	National Spent Nuclear Fuel Program
OCRWM	Office of Civilian Radioactive Waste Management
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
RH	relative humidity
RIP	code used to model repository performance assessment
RMEI	reasonably maximally exposed individual
RN	radionuclide(s)
SCC	stress corrosion cracking
SEM	scanning electron microscope
SF	spontaneous fission
SKI	Statens Karnkraftinspektion (Swedish Nuclear Power Inspectorate)
SNF	spent nuclear fuel
SR	site recommendation
SZ	saturated zone
TBS	to be supplied
TIC	Technical Information Center
TRIGA	research fuel reactor ($U\text{-}ZrH_x$)
TSPA	Total System Performance Assessment
TSPAI	Total System Performance Assessment and Integration
UZ	unsaturated zone
VA	viability assessment

ACRONYMS AND ABBREVIATIONS (Continued)

WAPDEG	waste package degradation
WF	waste form
WIPP	Waste Isolation Pilot Plant
WP	waste package
YM	Yucca Mountain
YMP	Yucca Mountain Project

1. PURPOSE

Under the provisions of the U.S. Department of Energy's (DOE's) *Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain, Nevada* (Dyer 1999 and herein referred to as DOE's Interim Guidance), the U.S. DOE must provide a reasonable assurance that the performance objectives for the proposed Yucca Mountain Project (YMP) radioactive waste repository can be achieved for a 10,000-year post-closure period. This assurance must be demonstrated in the form of a performance assessment that:

1. Identifies the features, events, and processes (FEPs) that might affect the performance of the geologic repository;

2. Examines the effects of such FEPs on the performance of the geologic repository;

3. Estimates the expected annual dose to a specified receptor group; and

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/Modeling 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 57 miscellaneous WF FEPs of this AMR.

1.1 SCOPE

This AMR has been prepared to satisfy the FEP screening documentation requirements in the Work Scope/Objectives/Tasks section of the Development Plan entitled *Miscellaneous Waste-Form FEPs* TDP-WIS-MD-000008 (CRWMS M&O 1999d).

The current FEPs list consists of 1786 entries. The FEPs have been classified as *primary* and *secondary* FEPs and 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. Several relevant FEPs do not fit neatly into the existing PMR structure. An example is criticality, and it is treated in FEP assignments as if it were a separate item. Some FEPs were best assigned to the TSPA itself (i.e., system-level FEPs), rather than to its component models.

This AMR addresses the 57 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).

1.2 FEPs IDENTIFICATION AND ANALYSIS

To demonstrate that regulatory-specified performance objectives of 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 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 WF FEPs, this AMR considers these first two steps of scenario development.

The first step of the scenario-development methodology is the identification of FEPs potentially relevant to the performance of the Yucca Mountain repository. The initial set of FEPs was created for the Yucca Mountain TSPA by Swift et al. (1999). They 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 intermediateand low-level waste programs of the United Kingdom, and from the United States' Waste Isolation Pilot Plant (WIPP) program. The NEA grouped these 1261 into 151 categories for the Yucca Mountain Project. These categoreis were also entered into the databese as FEPS. In addition, one more category was addeed (far-field criticalitiy), thus, the total NEA FEPs 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. 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.

All potentially relevant FEPs have been included, regardless of origin, giving a FEP list consisting of 1786 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 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 essence of all the secondary FEPs that are included in 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.

Each primary FEP and its corresponding secondary FEPs are assigned to various Process Modeling Reports (PMRs). Table 1 shows the 86 primary FEPs assigned to the WF 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 three criteria, which are stated in 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.

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.

Table 1. Primary Waste-Form FEPs (Gray shading indicates Miscellaneous WF FEPs in this AMR)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)	Include/ Exclude
1.2.04.04.00	Magma interacts with waste	ANL-WIS-MD-000009	Include
2.1.01.01.00	Waste inventory	ANL-WIS-MD-000009 ANL-WIS-MD-000006	Include
2.1.01.02.00	Codisposal/co-location of waste	ANL-WIS-MD-000009	Include
2.1.01.03.00	Heterogeneity of waste forms	ANL-WIS-MD-000009	Include
2.1.01.04.00	Spatial heterogeneity of emplaced waste	ANL-WIS-MD-000009	Exclude
2.1.02.01.00	DSNF degradation, alteration, and dissolution	ANL-WIS-MD-000009	Include
2.1.02.02.00	CSNF alteration, dissolution, and radionuclide release	ANL-WIS-MD-000009	Include
2.1.02.03.00	Glass degradation, alteration, and dissolution	ANL-WIS-MD-000009	Include/ Exclude
2.1.02.04.00	Alpha recoil enhances dissolution	ANL-WIS-MD-000009	Exclude
2.1.02.05.00	Glass cracking and surface area	ANL-WIS-MD-000009	Include
2.1.02.06.00	Glass recrystallization	ANL-WIS-MD-000009	Exclude
2.1.02.07.00	Gap and grain release of radionuclides after cladding perforation	ANL-WIS-MD-000009 ANL-WIS-MD-000007	Include/ Exclude
2.1.02.08.00	Pyrophoricity	ANL-WIS-MD-000009	Exclude
2.1.02.08.04	Flammable gas generation from DSNF	ANL-WIS-MD-000009	Exclude
2.1.02.09.00	Void space (in disposal container)	ANL-WIS-MD-000009	Include
2.1.02.10.00	Cellulosic degradation	ANL-WIS-MD-000009	Exclude
2.1.02.11.00	Waterlogged rods	ANL-WIS-MD-000008	Exclude
2.1.02.12.00	Cladding degradation before YMP receives it	ANL-WIS-MD-000008 ANL-WIS-MID-000048	Include
2.1.02.13.00	General corrosion of cladding	ANL-WIS-MD-000008	Exclude
2.1.02.14.00	Microbial corrosion (MIC) of cladding	ANL-WIS-MD-000008	Exclude
2.1.02.15.00	Acid corrosion of cladding from radiolysis	ANL-WIS-MD-000008	Exclude
2.1.02.16.00	Localized corrosion of cladding through pitting	ANL-WIS-MD-000008	Exclude
2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000012	Exclude
2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding	ANL-WIS-MD-000008	Exclude
2.1.02.19.00	Creep rupture of cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000007	Include
2.1.02.20.00	Pressurization from He production causes cladding failure	ANL-WIS-MD-000008	Include
2.1.02.21.00	Stress corrosion cracking (SCC) of cladding	ANL-WIS-MD-000008 ANL-WIS-MD-000007	Include
2.1.02.22.00	Hydride embrittlement of cladding	ANL-WIS-MD-000008 ANL-EBS-MD-000011	Exclude
2.1.02.23.00	Cladding unzipping	ANL-WIS-MD-000008 ANL-EBS-MD-000013 ANL-EBS-MD-000014	Include
2.1.02.24.00	Mechanical failure of cladding	ANL-WIS-MD-000008	Include
2.1.02.25.00	DSNF cladding degradation	ANL-WIS-MD-000008	Exclude
2.1.09.27.00	Localized corrosion perforation of a cladding by fluoride	ANL-WIS-MD-000008	Include
2.1.02.28.00	Diffusion controlled cavity growth (DCCG) concerns	ANL-WIS-MD-000007 ANL-WIS-MD-000008	Exclude
2.1.03.06.00	Internal corrosion of waste container	ANL-WIS-MD-000009	Exclude/ Include
2.1.07.01.00	Rockfall (large block)	ANL-WIS-MD-000008 ANL-WIS-MD-000009	Exclude
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and EBS	ANL-WIS-MD-000009	Exclude/ Include
2.1.08.07.05	Waste-form and backfill consolidation	ANL-WIS-MD-000009	Exclude
2.1.08.08.00	Induced hydrological changes in the waste and EBS	ANL-WIS-MD-000009	Include/ Exclude
2.1.08.10.00	Desaturation/dewatering of the repository	ANL-WIS-MD-000009	Include
2.1.09.01.00	Properties of the potential carrier plume in the waste and EBS	ANL-WIS-MD-000009	Include/ Exclude
2.1.09.02.00	Interaction with corrosion products	ANL-WIS-MD-000009	Exclude/ Include
2.1.09.03.00	Volume increase of corrosion products	ANL-WIS-MD-000009	Include/ Exclude

Table 1. Primary Waste-Form FEPs (Gray shading indicates Miscellaneous WF FEPs in this AMR) (Continued)

YMP FEP Database Number	FEP NAME	Document Identifier (DI)	Include/ Exclude
2.1.09.04.00	Radionuclide solubility, solubility limits, and speciation in the waste form and EBS	ANL-WIS-MD-000009	Include
2.1.09.05.00	In-Package sorption	ANL-WIS-MD-000009	Include/ Exclude
2.1.09.06.00	Reduction-oxidation potential in waste and EBS	ANL-WIS-MD-000009	Include
2.1.09.07.00	Reaction kinetics in waste and EBS	ANL-WIS-MD-000009	Include/ Exclude
2.1.09.08.00	Chemical gradients / enhanced diffusion in waste and EBS	ANL-WIS-MD-000009	Exclude
2.1.09.09.00	Electrochemical effects (electrophoresis and galvanic coupling) in waste and EBS	ANL-WIS-MD-000009 ANL-EBS-PA-000002	Exclude
2.1.09.10.00	Secondary phase effects on dissolved radionuclide concentrations at the waste form	ANL-WIS-MD-000009	Exclude
2.1.09.11.00	Waste-rock contact	ANL-WIS-MD-000009	Exclude
2.1.09.12.00	Rind (altered zone) formation in waste, EBS, and adjacent rock	ANL-WIS-MD-000009	Include/ Exclude
2.1.09.13.00	Complexation by organics in waste and EBS	ANL-WIS-MD-000009	Exclude
2.1.09.14.00	Colloid formation in waste and EBS	ANL-WIS-MD-000012	Include
2.1.09.15.00	Formation of true colloids in waste and EBS	ANL-WIS-MD-000012	Exclude
2.1.09.16.00	Formation of pseudo-colloids (natural) in waste and EBS	ANL-WIS-MD-000012	Include
2.1.09.16.01	Colloidal phases are produced by coprecipitation (in waste and EBS)	ANL-WIS-MD-000012	Include
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and EBS	ANL-WIS-MD-000012	Include
2.1.09.18.00	Microbial colloid transport in the waste and EBS.	ANL-WIS-MD-000012	Exclude
2.1.09.19.00	Colloid transport and sorption in the waste and EBS.	ANL-WIS-MD-000012	Include/ Exclude
2.1.09.20.00	Colloid filtration in the waste and EBS	ANL-WIS-MD-000012	Exclude
2.1.09.21.00	Suspensions of particles larger than colloids	ANL-WIS-MD-000012	Exclude
2.1.09.22.00	Colloidal Sorption at the groundwater interface	ANL-WIS-MD-000012	Exclude
2.1.09.23.00	Colloidal Stability and concentration dependence on aqueous chemistry	ANL-WIS-MD-000012	Include
2.1.09.24.00	Colloid diffusion	ANL-WIS-MD-000012	Include
2.1.10.01.00	Biological activity in waste and EBS	ANL-WIS-MD-000012	Exclude
2.1.11.01.00	Heat output / temperature in waste and EBS	ANL-WIS-MD-000009	Include
2.1.11.03.00	Exothermic and other thermal reactions in waste and EBS	ANL-WIS-MD-000009	Exclude
2.1.11.04.00	Temperature effects / coupled processes in waste and EBS	ANL-WIS-MD-000009	Include
2.1.11.05.00	Differing thermal expansion of repository components	ANL-WIS-MD-000009 ANL-EBS-MD-000015	Include
2.1.11.07.00	Thermally induced stress changes in waste and EBS	ANL-WIS-MD-000009 ANL-WIS-MD-000007	Include/ Exclude
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and EBS	ANL-WIS-MD-000009	Include/ Exclude
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS	ANL-WIS-MD-000009	Include/ Exclude
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and EBS	ANL-WIS-MD-000009	Exclude
2.1.12.01.00	Gas generation	ANL-WIS-MD-000009	Exclude
2.1.12.02.00	Gas generation (He) from fuel decay	ANL-WIS-MD-000009	Exclude
2.1.12.03.00	Gas generation (H_2) from metal corrosion	ANL-WIS-MD-000009	Exclude
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation	ANL-WIS-MD-000009	Exclude
2.1.12.06.00	Gas transport in waste and EBS	ANL-WIS-MD-000009	Exclude
2.1.12.07.00	Radioactive gases in waste and EBS	ANL-WIS-MD-000009	Exclude
2.1.12.08.00	Gas explosions	ANL-WIS-MD-000009	Exclude
2.1.13.01.00	Radiolysis	ANL-WIS-MD-000009	Exclude
2.1.13.02.00	Radiation damage in waste and EBS	ANL-EBS-MD-000015 ANL-WIS-MD-000009 ANL-WIS-MD-000010	Exclude
2.1.13.03.00	Mutation	ANL-WIS-MD-000009	Exclude
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	Include
3.1.01.01.00	Radioactive decay and ingrowth	ANL-WIS-MD-000009	Include
3.2.07.01.00	Isotopic dilution	ANL-WIS-MD-000009	Include/ Exclude

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 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 (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 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 included the FEP 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 86 WF FEPs, Table 1 presents the screening decision as either *Include* or *Exclude*. 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 *Include* and some that are *Exclude*. In these situations the screening decision specifies which elements are *Include* and which are *Exclude*. For FEPs assigned to this AMR, No. ANL-WIS-MD-000009, the disposition of the individual FEP is included in Section 6 in this AMR. Each discussion provides documentation of both the screening arguments and the TSPA disposition. Two other WF AMRs provide documentation for the remaining WF FEPs.

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 Process Model Reports (PMRs). 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, ANL-WIS-MD-000009, provide summaries and dispositions for the 57 miscellaneous WF FEPs.

2. QUALITY ASSURANCE

The Quality Assurance (QA) program applies to the development of this analysis documentation. The Performance Assessment Operations responsible manager has evaluated this document development activity in accordance with QAP-2-0, *Conduct of Activities*. The QAP-2-0 activity evaluation, *Conduct of Performance Assessment* (CRWMS M&O 1999a), has determined that the preparation and review of this technical document is subject to *Quality Assurance Requirements and Description* (QARD) DOE/RW-0333P (DOE 2000) requirements. Preparation of this analysis did not require the classification of items in accordance with QAP-2-3, *Classification of Permanent Items.* This AMR has been developed in accordance with procedure AP-3.10Q, *Analyses and Models*. The results of this AMR do not affect the design or performance of any permanent items.

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 approved software (Microsoft® *Word 97*) for word processing, which is exempt from qualification requirements in accordance with AP-SI.1Q, *Software Management*. There were no additional applications (Routines or Macros) developed using this commercial software.

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 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 include/exclude FEPs. The guidance provides a low probability argument to exclude a FEP 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

Both DOE's Interim Guidance (Dyer 1999) 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 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 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 (LADS) Report* (CRWMS M&O 1999c, pp. 0-21 to 0-26 and 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 strengthens the *Exclude* screening decision.

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 edict (Stroupe 2000).

Assumption 5.3: The assumptions provided by the interim guidance Dyer (1999) will be used within this analysis. The basis for using these assumptions is management edict. 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. If the only significant consequences of the FEP are improved repository performance, and the 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 57 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) and by the EPA in 40 CFR Part 197 (64 FR 46976). The criteria are used to determine whether or not each FEP should be included in the TSPA.

For FEPs that are *Exclude* 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 *Exclude* Screening Decision based solely on proposed regulatory requirements or regulatory-specified assumptions.

For FEPs that are *Exclude* based on 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 *Include*, 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.

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. 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).

Screening Decision: *Include*

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 1×10^{-8} per year adopted by DOE as the level of concern. Therefore, this FEP must be considered as a possible disruptive event (Attachment I).

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.

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 nondisturbed 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).

Basis for Screening Decision:

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 unlikely. 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.

Attachments:

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.

This FEP discusses only the radioactive waste inventory.

Screening Decision: *Included.* However, only a limited number of radionuclides are shown to be important to repository performance.

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.

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 recommendations for 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, ^{243}Am should be 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 should 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 will be 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, will be adequately represented in the TSPA-SR (CRWMS M&O 2000j).

Human-intrusion calculations should include ^{14}C, ^{99}Tc, ^{129}I, ^{227}Ac, ^{229Th}, ^{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 should be included for the human-intrusion scenario because a human-intrusion event could occur as early as 100 years (CRWMS M&O 2000j). The recommended averagen per-package inventory for CSNF, DSNF, and HLW is shown in Table 2.

Table 2. 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
^{229Th}	0.00E+00	2.46E-02	3.79E-03
^{230Th}	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

DTN: SN0003T0810599.009

Basis for Screening Decision:

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 the Radionuclide Screening for the Total Systems 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 assumptions, 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) that was submitted to the TDMS under DTN: SN0003T0810599.009.

6.2.3 Codisposal/co-location of waste – YMP No. 2.1.01.02.00

***YMP Primary FEP Description**:* Codisposal and co-location refer to the disposal of CSNF, DSNF, HLW, and possibly other wastes in close proximity within the repository. Codisposal and co-location 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**: Include* co-location and codisposal. The discussion argues that one of these issues, i.e., the chemical interactivity between DSNF and HLW glass within a codisposal package, must be included in TSPA analyses.

Include chemical interactivity between DSNF and HLW glass affects both DSNF degradation and radionuclide mobilization.

Exclude – no credit for DSNF cladding.
Exclude – no credit is taken for any beneficial effects of DSNF and glass-pour canisters as barriers to DSNF degradation, to HLW-glass dissolution, or to radionuclide release.
Exclude DSNF geometry area dependence based on low consequence.
Exclude dependence of radionuclide release on DSNF surface area based on low consequence.
Exclude chemical interactivity between waste packages based on low probability.
Exclude preferential condensation based on (1) thermal shielding caused by the near-field averaging of the thermal field renders preferential condensation a process of low consequence and (2) occurrence of repository condensation and added uncertainty regarding the occurrence of preferential condensation render preferential condensation a process of low probability.

***Screening Argument**:* Co-location issues consider interactions between waste packages, while codisposal issues consider interactions between components of a single waste package. Co-location issues include the effects of other waste forms, chemical interactivity, and preferential thermal interactivity between waste packages. Currently, other waste forms such as chemical, low-level nuclear, and intermediate-level nuclear wastes are scheduled for disposal in the Yucca Mountain repository. Chemical and preferential thermal interactivity, i.e., interactivity that would cause preferential condensation at a DSNF or codisposal package, is excluded because of low probability.

Codisposal issues include the effects of DSNF and glass-pour canisters, DSNF geometry, and chemical interactivity between DSNF and HLW glass. The Yucca Mountain Project has elected not to take credit for the canisters as barriers to radionuclide release. Dependence of radionuclide release on DSNF (surface area) is of low consequence. Finally, the discussion identifies chemical interactivity between DSNF and HLW glass as an important factor, which must be considered by TSPA analyses.

TSPA Disposition: Geochemistry in the TSPA-SR model that accounts for chemistry effects.

Basis for Screening Decision:

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; and
- 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 include a number of issues relative to codisposal and co-location, issues with contrasting dispositions.

Co-Location Issues

The co-location issues identified by this FEP may be excluded from the TSPA analyses.

Co-Location with Other Waste Forms: Secondary FEPs 2.1.01.02.01 and 2.1.02.02.03 focus on the possible effects of co-location 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 Dyer (1999) stipulates that future human behavior in the region should be assumed to be the same as that occurring at the present time. These issues therefore are moot and should be excluded from the TSPA.

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 unlikely. Interactions are unlikely 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 unlikely. The issue of chemical interactivity between waste packages is therefore excluded based on low probability.

Thermal Interactions between Waste Packages: FEP 2.1.02.09.00 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&0 2000aa and CRWMS M&0 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 time period of concern begins whenever 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.

Codisposal Issues

In contrast to co-location 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 base case model.

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

1999e, 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 1999e, 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 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.

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.

DSNF Geometry: Secondary FEP 2.1.02.08 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 1999e, 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 1999e, Section 6.4.3.2).

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.

Other aspects of waste heterogeneity are addressed in the following FEPs: "Spatial Heterogeneity of Emplaced Waste" (YMP No. 2.1.01.04.00) and "Waste Inventory" (YMP No. 2.1.01.01.00). This FEP includes information developed for secondary FEPs addressing "DOE SNF Geometry" (YMP No. 2.1.01.03.01) and "DOE SNF Structure" (YMP No. 2.1.01.03.02).

Screening Decision: *Include*

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 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.

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.

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

The CSNF 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	Number of Fuel Assemblies 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 are planned to be disposed together in codisposal packages (CDSP). 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	Canisters-per-Package 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.

Basis for Screening Decision:

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. Waste categories of CSNF, DSNF, and HLW, at a minimum, 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 site 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 1999b, Section 1; 2000d). One fuel type, Uranium metal, comprises most (~85%) of the DSNF

(CRWMS M&O 1999b, Table 3; 2000d). 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 1999b, Section 7.1; 2000d).

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 1999b, Section 6.3.1; 2000d).

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 radiologic 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.

Other aspects of waste heterogeneity are addressed in the following FEPs: "Heterogeneity of Waste Forms" (YMP No. 2.1.01.03.00), "Waste Inventory" (YMP No. 2.1.01.01.00).

Screening Decision: *Exclude* – based on low consequence

Screening Argument: Heterogeneity in the mass of radionuclides available for transport from individual waste packages can be ignored in total system performance assessment. Containers of CSNF, DSNF, and HLW are likely to be emplaced randomly throughout the repository. Canister-to-canister and package-to-package variations will be spatially averaged by random placement. Any heterogeneity that remains after spatial averaging is expected to have a negligible impact on total system performance.

TSPA Disposition: N/A

Basis for Screening Decision:

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. Any heterogeneity that remains after spatial averaging is expected to have a negligible impact on total system performance.

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 (DID) calculations for the license application (LA) design (CRWMS M&O 1999f). The DID 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 colocated 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 internal can (HIC) on DSNF degradation.

Screening Decision: *Include.*

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 DSNfF 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.

- ***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 expeimentally 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. 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 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).

Further Basis for TSPA Disposition:

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 1.75×10^6 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 solubilities 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.

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 analysis as they become available.

Influence of Pu Ceramic

Pu is incorporated in a discrete crystalline phase which 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 assuming 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 1999e, 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 1999e, 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 1999e, 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 WP), 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: *Include* (See other FEPs on specific phenomenon included and excluded)

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 in 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 is 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).

TSPA Disposition: Degradation and dissolution of the CSNF matrix is included in the Waste Form Degradation Model as a specific component, the CSNF Matrix Degradation Component as summarized in the *Waste Form Degradation Process Model Report* (CRWMS M&O 2000n). The component consists of a kinetic rate equation (also known as an intrinsic dissolution rate or

forward reaction rate equation that assumes 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).

Basis for Screening Decision:

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: *Include*: in package chemistry-dependent corrosion rates and congruent dissolution

Exclude: phase separation (low probability [credibility]), selective leaching (low consequence), and precipitation of silicate and other minerals (conservatively bounding).

Screening Argument: The glass-dissolution model used in TSPA is derived from a reference glass 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. It is reasonably concluded that radionuclides are to be released congruently as the glass is degraded.

A summary of excluded processes follows:

Phase Separation. 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.

Selective Leaching. Selective leaching of alkali metals and alkaline-earths is expected to occur upon initial contact of waste-glass with groundwater. However, this is a transient and unimportant process for radionuclide release. The basis for the 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.

Co-precipitates/Solid solutions in Glass Waste Form. 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 radionuclides.

Precipitation of Most Silicates. Precipitation of most silicates will be slower than glass dissolution. Consequently the removal of silica by precipitation of secondary solids is overcompensated 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.

TSPA Disposition: HLW degradation and dissolution is included in the total-system analysis as a source term for the mobilization of contaminants. 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 degradation, alteration and dissolution model for HLW in the TSPA can be summarized as follows (see attached figures):

- 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, p. 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.

Basis for Screening Decision:

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 TSPA-SR will be similar to that in 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 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, b). 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, b).

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, b, 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 WP); 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: *Exclude* – The effects of alpha-recoil from performance-assessment calculations on the basis of low consequence to the performance of the disposal system.

Screening Argument: Alpha decay of radionuclides could result in an additional pathway for movement of the radionuclides from the surface of a degraded waste form into groundwater flow. When radionuclides decay by alpha emission, the remaining nucleus (decay daughter) will recoil with a momentum equal in magnitude, but opposite in direction, of the emitted alpha particle. If the daughter nucleus is very close, within several atom mono-layers, to the surface of the waste form, it may be directly released to groundwater in the immediate vicinity. This effect would increase the degradation and dissolution rate for radionuclides from the fuel material (fuel meat) into the groundwater flow.

TSPA Disposition: N/A

Basis for Screening Decision:

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 (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*

n(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):

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

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):

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

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.

$$Disp(t_m) = \begin{cases} \int_{E_d}^{T_m} \frac{T}{2E_d \quad m - 2E_d^2} dT & if \quad T_m \leq E_i \\ \int_{E_d}^{E_i} \frac{T}{2E_d \quad m - 2E_d^2} dT + \int_{E_i}^{T_m} \frac{E_d}{2E_d \quad m - 2E_d^2} dT & if \quad T > E_i \end{cases}$$

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

For the example involving ^{238}U decaying to ^{234}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(a)*. 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 (3.0×10^{-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, we can express it 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 3, 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 3 into proper perspective, comparison to Figure 1 needs to be made. Figure 1 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.

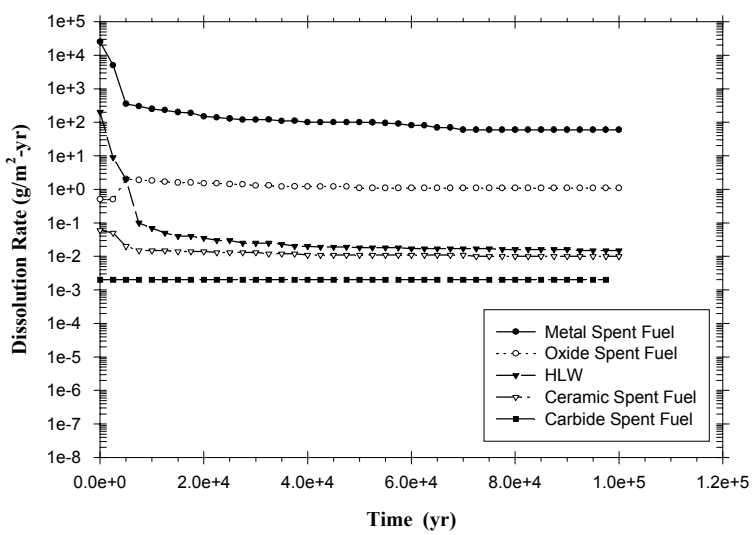
Table 3. Alpha-Recoil Enhanced (from Both α and α-Recoil Atom) Dissolution Rates Due to the Major Mass-Based Constituents of SNF and HLW to be Disposed in the Yucca Mountain Repository

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	4.47×10^{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) 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 $\lambda=\ln(2)/\iota_{½}$, where $\iota_{½}$ 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]).



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

Figure 1. Comparison of Dissolution Rates for High-Level Waste, Metallic Carbide, and Ceramic Spent Nuclear Fuel

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: *Include*

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.

TSPA Disposition: The effect of using an area-enhancement factor will appear in the characterization of mobilization for HLW.

Basis for Screening Decision:

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 an area-enhancement factor, as currently employed by 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 unlikely to occur below 400°C.

Screening Decision: *Exclude*

Screening Argument: 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.

TSPA Disposition: N/A

Basis for Screening Decision:

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, Vol. 2, Section 2.2.2).

6.2.12 Gap and Grain Release of Cs, I – YMP No. 2.1.02.07.00

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

Screening Decision: *Include* gap and grain-boundary inventory produced while in repository. *Exclude* Additional gap and grain-boundary inventory potentially produced while in repository, and any reactions which would mitigate the gap and grain-boundary inventory and, thereby, releases.

Screening Argument: This FEP and its associated secondary FEPs present three issues as discussed below

Gap-and-Grain-Boundary Release

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

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

and I radionuclides in the inventory are assumed released immediately upon perforation of the cladding in the CSNF Cladding Degradation Component of the Waste Form Degradation Model.

I and Cs Migration

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

Pb-I Reactions

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

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

Basis for Screening Decision:

Prior discussion provides adequate basis.

6.2.13 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: *Exclude* – based on low consequence.

Screening Argument*:* An analysis has shown that a pyrophoric event (PE) would have only a minimal impact on 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 based on low consequence.

TSPA Disposition: N/A

Basis for Screening Decision: This analysis, pyrophoricity is defined as the self-sustaining oxidation of SNF. The issue 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 extremely small. First, a sufficient ignition source has not been identified. The presence of U-hydrides in N-reactor fuel has been identified as a possible ignition source. However, the existing hydrides observed so far are present only in small quantities (less than 2%) and tend to be located either inside the fuel matrix or deep within cracks in the fuel surface (Marschman et al. 1997). It has not been demonstrated that hydrides would be capable of initiating an event that would cause the bulk fuel to oxidize.

A second issue is the presence of sufficient oxygen to support a full-scale PE. The waste packages are planned to be filled with an inert gas prior to emplacement in the repository. The inert environment will displace water and oxygen, thus not allowing or at least greatly reducing the environment for chemical reactions (DOE 1998d, Section 5.1.2.1). After 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. Also, a sufficient flow would have to be established (oxygen into and inert gas out of a breached waste package) to preclude the build up of sufficient inert gas inside the waste package to interrupt the oxidation process.

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 assume that PEs are possible and estimates the impact they would have on repository performance if they were to occur.

In terms of pyrophoricity, the DOE Office of Environmental Management/Office of Civilian Radioactive Waste Management (EM/RW) Repository Task Team (DOE 1997, pp. 20-21]) has identified three fuel types: 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 a small part of the total waste inventory.

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 tested 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 dominated by UO_2 (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. Hydrides tend to be a finely divided particulate (having high specific surface area) with a much lower ignition temperature than the bulk uranium metal. Examination and testing of damaged N-reactor zirconium-clad U-metal fuel revealed the presence of only small amounts of uranium hydride as precipitates within the metal and in thin coatings on the surfaces of internal cracks (Marshman et al. 1997, Section 3.4.2). The possibility exists for additional U-hydride growth during interim storage (Reilly 1998, p. 30) This amount of hydride could potentially act as an ignition source if concentrated in a small area. TRIGA fuels, which are predominantly U-Zr-hydride, are an exception. Zr-hydride is more stable than Zr metal.

Aluminum-clad uranium-metal fuel underwent similar storage treatment as the zirconium-clad U-metal fuel. It should produce about the same relative amount of hydrides as the zirconium-clad fuel. The U-Mo fuels should also behave similarly. Aluminum-clad uranium/aluminum-alloy matrix fuel has little potential for forming hydrides in the alloy fuel. The electrochemical potential for this occurring 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 mol% and would not be pyrophoric because of the oxide mass.

Carbide fuels are not a pyrophoric hazard in a dry atmosphere except as high surface area powders. For long-term storage considerations, dust should not be a problem. 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 elements of discussion 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 radionuclide inventory in the affected packaged.
- A PE would cause the two adjacent waste-packages to breach.
- This analysis will be limited to N-reactor SNF because the N-reactor fuel (zirconium-clad uranium-metal fuel) has the greatest potential for being pyrophoric. Also, N-reactor 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, it was assumed that a waste-package 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, by assuming the various types of waste packages (WPs) are evenly distributed throughout the repository, 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 WPs

N_{nr} = number of N-reactor WPs

F_b = fraction of WPs breached

N_a = number of adjacent WPs breached per-PE

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

Eq. 6 predicts 1.1, 0.8, 0.5, 0.3 and 0 percent increases in the number of WPs failing due a PE, assuming normal failure rates of 60, 70, 80, 90, and 100 percent, respectively. Because an even distribution of waste-package types was assumed, 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 terms of the percent increase in radionuclide release, the worst case a PE involving a single N-reactor waste package can have would be if it involved the repository's first waste-package failure. In this case the initial release would be approximately three waste-package equivalents (WPEs) of radionuclides rather than one WPE. However, the probability that an N-reactor waste package is the first waste package to fail is equal to:

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

Therefore, on average, the worst-case increase in radionuclide release attributable to a single PE can be calculated as the probability that it does not involve an N-reactor waste package times 1 WPE plus the probability it does involve an N-reactor waste-package times three WPEs.

$$(0.987)(1\text{WPE}) + (0.013)(3\text{ WPEs}) = 1.03\text{ WPEs}. \qquad \text{(Eq. 8)}$$

This represents a 3% 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.

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 3% increase in the dose resulting from that clustered event (see Eq. 8). This conservatively assumes that the waste packages adjacent to the waste packages undergoing PEs were not failed during the clustering event. It also assumes 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 would have only a minimal impact on 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 in the FEP process.

6.2.14 Flammable Gases Generation from DSNF– YMP No. 2.1.02.08.04

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, acetylene is generated. If the acetylene ignites, localized increases in temperature can occur which might affect fuel degradation, and the area around the ignition point may be mechanically and/or thermally perturbed, affecting the waste container or host rock in the area of the EBS.

Screening Decision: *Exclude* – based on low consequence

Screening Argument: DSNF has a small percent (~1%) of carbide fuels (DOE 1999b, p. D-2). The chemical reaction of U/Th carbide and water produces acetylene. Only DSNF from the Peach-Bottom Core 1 has a potential to generate acetylene gas and, thereby, consequences from a gas explosion. Also, relatively good connection between the repository and land surface through fractures would eventually disperse any acetylene generated in the repository before explosive concentrations could be obtained.

TSPA Disposition: N/A

Basis for Screening Decision:

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 (CRWMS M&O 1999b, Section 1; 2000d), 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 1999; 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. Because the repository is relatively well-connected to the land surface through fractures in the overlying rock formations, any gases generated will disperse and be diluted with air. Acetylene is 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 conditions in a repository in which DSNF canisters are breached are akin to commercial safe-handling conditions for acetylene—low pressures, with a high dilution of inert gases (nitrogen in the air, water vapor, and CO_2 and CH_4 from carbide reactions). An analysis of carbide fuels (Propps 1998) indicated that hydrocarbon production is not a problem.

In the case of backfilled drifts, any gases produced gases will still diffuse away, albeit more slowly than in drifts without backfill. Higher concentrations of flammable gases might accumulate in waste packages or beneath drip shields. However, in this case, the backfill would similarly impede oxygen diffusion into the waste packages, so sufficient quantities of oxygen to support combustion would be unlikely.

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 1995, 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, so only about (0.35)(7%) = 2.5% of the carbide fuel is readily susceptible to reaction with water. In the total repository, (2.5%)(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 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 dispersed over a large area of the repository and co-disposed in 648 waste packages with CSNF, so the probability of generating problematical amounts of flammable gases is very remote.

A screening argument can be based on four 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 which would disperse the flammable gases and dilute potentially autocatalytically explosive acetylene in drifts without backfill (consequence); (3) an insufficient flow of oxygen to sustain hydrocarbon fires in backfilled drifts (consequence); and (4) insufficient pressures to cause acetylene to decompose autocatalytically, because of the high permeability of overlying strata giving good connection to the atmosphere (consequence).

6.2.15 Void Space (in Disposal Container) – YMP No. 2.1.02.09.00

YMP Primary FEP Description: If handling containers ("canisters") and disposal containers are not completely filled, but rather, any remaining space filled with an inert gas or air, then this unfilled volume could influence water-chemistry calculations.

Screening Decision: *Include* concept of unfilled void volume in TSPA-SR/LA calculations.

Screening Argument: The concept of an unfilled void volume is acknowledged and included in some aspects of the TSPA-SR/LA. Two sources of void volume are considered: the unfilled volume and the pore space of rind or corrosion products that forms as the waste form is altered.

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 disposal container 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 TSPA-VA and TSPA-SR/LA, after canister 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 container is reasonably concluded 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.)

Basis for Screening Decision:

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

6.2.16 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: *Exclude* – based on low probability.

Screening Argument: According to preliminary waste acceptance criteria, no cellulosic materials will be included as part of the waste in the potential Yucca Mountain repository.

TSPA Disposition: None

Basis for Screening Decision:

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 effects 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.

6.2.17 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: *Exclude* based on conservatism and low consequence.

Screening Argument: No protective performance is taken for DSNF cladding because more than 80 percent of the DSNF is from the N-Reactor and is in poor condition; up to 70 percent may be already perforated. Unzipping of the cladding is conservatively neglected since the inventory is small.

TSPA Disposition: N/A

Basis for Screening Decision:

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. 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, TSPA takes no credit for radionuclide retardation by the cladding. Also, 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 from the CRWMS M&O (1999b, Section 3.5.3; 2000d) and 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."

6.2.18 Internal Corrosion of Waste Container – YMP No. 2.1.03.06.00

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

Screening Decision: *Exclude* – prior to waste package breach. Based on low consequence. *Include* – after waste package breach.

Screening Argument: The waste package could be corrosively attacked from inside as well as outside. Prior to the breach of the waste packages, there should be minimal corrosion, because of the inert environment in the package. Radiolysis products could contribute to the interior corrosion, but they are ignored based on low consequence.

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. Because no credit will be taken for the canisters, mechanisms that might enhance canister degradation do not impact the predicted consequences from the DSNF.

TSPA Disposition: Included as part of In-Package Chemistry Component.

Basis for Screening Decision:

The waste package could be corrosively attacked from inside as well as outside. 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 environment for chemical reactions. Prior to the breach of the waste packages, there should be no or minimum corrosion due to the inert environment in the package from normal chemical and water-vapor mechanisms (CRWMS M&O 2000n, Section 1.6; 1999h).

Prior to WP breach, internal corrosion is excluded because of low consequence. There will be no or minimum corrosion because of the specified inert gases in the package and limited water 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 itself (Gray and Einziger 1998, Section 4). Other DSNF forms would have low water content.

Once the waste package has been breached and water enters the package, the CSNF dominates the process, due to its having the majority of the volume, for the waste-package degradation. 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 WP, 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 Rockfall (Large Block) – YMP No. 2.1.07.01.00

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

Screening Decision: *Exclude* – based on low consequence.

Screening Argument: The use of backfill and a drip shield significantly reduces the kinetic energy and velocity of falling rocks impacting EBS components. Also, a backfill consisting of granular materials will be highly effective at absorbing or spreading the energy/impulse-load, thus reducing or eliminating damage to EBS components.

TSPA Disposition: N/A

Basis for Screening Decision:

A combination of drip shield backfill is now being included as an EBS design feature. The use of backfill significantly reduces the drop distances of falling rock blocks, and thus reduces the

kinetic energy and impact velocity of all falling rocks hitting backfill (instead of waste packages/waste forms). Also, a backfill consisting of granular materials with dimensions orders of magnitude smaller than the size of the falling blocks of rock will be a highly effective energy/impulse-load absorbers, reducing and preventing damage to HLW glass and other waste forms. Thus, the glass-waste-form surface-area increase in response to falling rock blocks which could theoretically cause form fracture and fragmentation, is expected to be nil.

Bounding static analyses performed in CRWMS M&O 2000f (Section 6.5.3) showed that the combined load from rockfall and backfill (27 psi) was much less than the estimated collapse pressure of a drip shield idealized as a cylindrical shell (375 psi). Further analyses of the drip shield, idealized as either a cylindrical shell or flat plate, bounded the maximum deflection of the crown of drip shield between 19 mm and 77 mm. These analyses indicate that it is very unlikely that rockfall will ever come in contact with waste.

6.2.20 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: *Include* through the use of a series of linked one dimensional flowpaths and mixing cells through the EBS, drip shield, waste package and into the invert.

Exclude preferential pathways within the EBS, waste form and invert based on beneficial consequence.

Screening Argument: 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 itself 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 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.

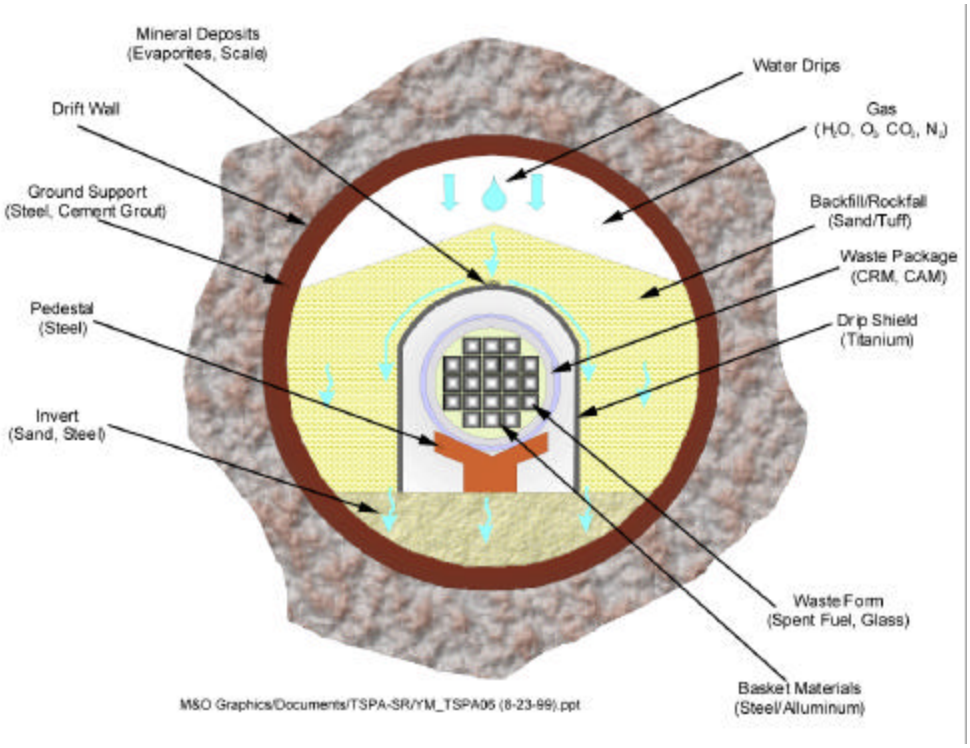
The drip shield and its surrounding backfill can form a preferential pathway if adjacent drip shields separate during the lifetime of the repository. The possibility of enhanced flow through a separation between adjacent drip shields is included in the computational models for corrosion of the drip shield 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 flow path 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 unlikely 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 (Figure 2). 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 flow paths onto the waste package.



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

Figure 2. Schematic of Drift Conditions for Engineered Barrier System (EBS)

Basis for Screening Decision:

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, it is unlikely that discrete fractures will 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.

6.2.21 Waste-Form and Backfill Consolidation – YMP No. 2.1.08.07.05

YMP Primary FEP Description: Physical and chemical degradation of the drip shield, 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: Excluded based on low consequence or possible slight beneficial consequence which is conservatively ignored.

Screening Argument: Waste form and backfill consolidation can be excluded based on low or slight beneficial consequence, because backfill and consolidation would tend to decrease the available reactive surface area (pore area) and permeability of the waste. The potential deleterious effect of maintaining water contact with the waste form is already conservatively bounded by assuming that the waste package contents are fully accessible once the waste package has been breached.

TSPA Disposition: N/A

Basis for Screening Decision:

Inconsequential gravity induced settlement of the backfill is expected. Rockfall and collapse of the drift walls is 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.

Current designs call for the use of a drip shield surrounded by backfill. The backfill will not come into contact with the containers or waste form until the drip shield fails due to corrosion from contact with water. Both the use of backfill and any consolidation would tend to decrease the available reactive surface area, 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 unlikely for 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.

6.2.22 Induced Hydrological Changes in the Waste and EBS – YMP No. 2.1.08.08.00

YMP Primary FEP Description: Thermal, chemical, and mechanical processes related to the construction of the repository and the emplacement of waste may induce changes in the hydrologic behavior of the system.

Screening Decision: *Include* induced hydrological changes (flow areas) from corrosion for the waste package and drip shield
Include induced hydrological changes (exposed fuel area) for the waste form

Exclude changes to hydrological properties for the waste form based on low consequence
Exclude changes to hydrological properties for the invert based on low consequence

Screening Argument: 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 waste package. 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 impact of thermal, chemical and mechanical effects on hydrological properties of the waste form is excluded from the EBS model for the TSPA. The waste form is represented as a single mixing cell in the EBS system. This mixing cell provides instantaneous access of groundwater to all the waste form. This reasonable, very conservative conclusion is independent of changes in hydrological properties from thermal, chemical, and mechanical processes in the waste.

The impact of thermal, chemical, and mechanical effects on the hydrological response of the invert is also excluded from the TSPA models. 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 small, on the order of 0.5 meters, and (3) there will be no sorption for any radionuclide expected to be released.

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. However, thermal effects in the repository are not anticipated to further change the physical state of the waste because spent fuel experiences much greater extremes of temperature in the reactor than in the repository, and because vitrified waste is formed at much higher temperatures than will occur in the repository. In this situation, the temperature variations in the repository will have at most a minor impact on the physical state of these materials.

- 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.

- 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 not included in the TSPA models because waste-form temperatures will be below 350°C, the temperature at which unzipping is initiated.

N/A for the waste form
N/A for the invert

Basis for Screening Decision:

The impact of thermal, chemical, and mechanical effects on hydrological properties of the waste form is excluded from the EBS model for the TSPA. The waste form is represented as a single mixing cell in the EBS system. This mixing cell provides instantaneous access of groundwater to all the waste form. This reasonable, very conservative conclusion is independent of changes in hydrological properties from thermal, chemical, and mechanical processes in the waste.

The effect of induced hydrological changes on the invert is excluded from the TSPA. The invert will be filled with quartz sand or crushed tuff that is represented as a homogeneous, porous medium with no retardation of radionuclides. Thermal and chemical processes will have minimal effect on the invert because retardation is already at its minimum value and because the quartz sand or crushed tuff will be relatively inert in any anticipated repository environment.

Precipitation 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 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 small, on the order of 0.5 meters, and (3) the 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.

6.2.23 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: *Include*

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.

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.

Basis for Screening Decision:

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 postoperational period. In response to waste heat, vapor movements will rapidly redistribute both water and vapor, thus removing relic effects due to dewatering.

This FEP is relevant to YMP No. 2.1.08.10.01, YMP No. 2.1.08.10.01, YMP No. 2.1.08.11.00, and to models described in the following AMRs: Mountain-Scale Coupled Processes (TH) Models (CRWMS M&O 1999n), Multi-Scale 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).

6.2.24 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: *Include* potential effects of carrier plume by evaluating the influence of steel corrosion on the water chemistry in order to establish an uncertainty band.

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

Screening Argument: The current design of the tunnel support system uses a steel liner. The WP 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 has eliminated concrete; consequently, the amount of cementitious material that can influence water chemistry has been decreased 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 neglected because of low beneficial 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 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 effects 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 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) it is assumed to have no sorption for any radionuclide.

Also note that any changes in invert permeability induced by the carrier plume 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.

The effects of a carrier plume on the unsaturated zone are discussed in FEPs 2.2.07.11.00 and 2.2.08.02.00.

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 WP, only the WP steel is used establish the uncertainty band. Therefore, water entering the waste package is assumed to have 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 WP, waste type, waste degradation rate, cladding failure, and seepage flux.

Basis for Screening Decision:

The basis for the screening decision is sufficiently discussed in the above screening argument.

6.2.25 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 solubilities.

Screening Decision: *Include* the presence of a rind around the fuel pellets on the availability of water for radionuclide dissolution;
Include the interaction between the expanding rind and the cladding: both in sealing of the gap and in unzipping of the cladding;
Include selected chemical effects in the integrated source term for each waste form.

Exclude the potential effects from corrosion products on advective or diffusive transport of water and radionuclides;
Exclude the potential sorptive effects from corrosion products (see YMP No. 2.1.09.05.00);

Screening Argument: The presence of a rind of corrosion products around the fuel pellets is included in determining the volume of water available for dissolution of radionuclides. The volume of water is estimated by reasonably concluding that the pore space of the corrosion products is saturated with water and that this pore fluid is the fluid available to dissolve radionuclides up to their solubility limits.

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 close to the cladding breach. This continued reaction is assumed to exert force on the cladding, which splits the cladding open at a rate that is faster than the forward dissolution rate.

No credit is taken for (1) the potential of low-permeability corrosion products to limit transport of radionuclide species within the waste package (other than within intact portions of fuel rods), and (2) the potential retardation of radionuclide species on the corrosion products within the waste package. These are conservative conclusions for the TSPA-SA and TSPA-LA because they maximize transport from the source (waste form) to the EBS. However there is one exception when zero retardation is nonconservative. This exception is for near-field criticality,

where retention of radionuclides within the waste package may be quite important. In the evaluation of near-field criticality, sorption is therefore considered within the waste package. See YMP No. 2.1.14.11.00 for details.

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.

TSPA Disposition*: Include* the presence of a rind around the fuel pellets on the availability of water for radionuclide dissolution. This is included in the time dependent performance assessment numerical model calculation of the water volume within the WP.

Include the interaction between the expanding rind and the cladding: both in sealing of the gap and in unzipping of the cladding. This is included in the fast release fraction and wet unzipping model.

Exclude the potential effects from corrosion products on advective or diffusive transport of water and radionuclides. N/A

Exclude the potential sorptive effects from corrosion products. N/A

Include selected chemical effects in the integrated source term for each waste form. This is included in the chemistry-dependent solubilities, colloid concentrations, and waste form corrosion rates.

Basis for Screening Decision:

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 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 TSPA-VA (CRWMS M&O 1998c, p. 6-132)). Instead, it is assumed that water entering the waste package is instantly mixed with all resident water, which is assumed to have 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.

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 Zircalloy 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 the *Clad Degradation – Summary Abstraction* (CRWMS M&O 2000b).

- 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. (see *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-WP 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* (CRWMS M&O 2000z).

6.2.26 Volume Increase of Corrosion Products – YMP No. 2.1.09.03.00

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

Screening Decision*: Included*: clad unzipping due to wet oxidation of CSNF
Excluded: dry oxidation of CSNF based on low consequence.

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

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

TSPA Disposition: Fuel rods with perforated cladding are expected to remain intact until the WP fails and permit air and moisture to enter. Wet unzipping is modeled to start at WP failure for rods that are pre-breached or when rod perforation occurs, if after WP failure. In the model, the fuel matrix is dissolved at the intrinsic dissolution rate that is evaluated at the local temperature and in-package chemistry. The dissolved UO_2 is assumed to form a precipitate. This secondary phase isolates most of the fuel from the moisture and increases volume compared to UO_2. In time, the clad in the reaction region is torn as the reaction continues. This reaction region is assumed to be cone shaped based on experimental observations of dry unzipping. The unzipping propagates along the rod at a rate approximately 9 times (range 1 to 200 times) faster than the intrinsic dissolution rate. It is assumed that the perforation is in the center of the rod. This maximizes the release rate. The time to unzip a rod as a function of temperature, local chemistry and pH. In TSPA, the unzipping velocity and fraction of fuel exposed is evaluated at each time step because of the evolution of in-package chemistry and temperature (CRWM M&O 2000b; CRWM M&O 2000w).

Basis for Screening Decision*:*

Dry Oxidation

The clad that has failed during reactor operation, or from creep rupture, and delayed hydride cracking (DHC) generally has small perforations and very small areas of fuel exposed to the near-field environment. When the WP fails, the inert atmosphere inside the WP is replaced with the in-drift environment. In dry oxidizing conditions and with perforated cladding, the UO_2 phase in the spent fuel can oxidize in two stages, first to U_4O_9 and then to the U_3O_8 phase (see CRWMS M&O (1998c, Section 6.3.1.2), causing an increase of the spent-fuel-matrix volume. The stress from the volume increase can unzip the clad, causing a gross rupture of the clad and

exposing the fuel inside. The fuel also has an increase in surface area and dissolution rate, the combined factor being 150 (Gray and Wilson 1995).

However, dry oxidation is not expected to be a significant contributor to the results and is important only for the limited case of early container and clad failure. This is due to the expected long life of the waste package, the expected aqueous conditions on the waste surface exposed to the drift environment, and the expected deliquescent properties of the fuel. In general, by the time a WP fails, the fuel temperature will be low enough that U_3O_8 conversion is not expected. (CRWMS M&O 1998c, Section 6.3.1.2,). Dry fuel oxidation has been experimentally studied by Einziger et al. (1982) and Hanson (1998). McEachern and Taylor (1997) summarize the work of others (CRWMS M&O 1998c).

Wet Oxidation

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

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

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

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

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

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

6.2.27 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: *Include*

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 solubilities of RN-bearing solids formed by the interaction between RN-bearing waste (spent nuclear fuel) and the solution phase.

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).

Basis for Screening Decision:

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 solubilities 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, we need 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 proposed 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.

6.2.28 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: *Include* sorption on mobile colloidal material

Exclude sorption within the waste form/waste package based on beneficial consequence (conservative).

Screening Argument*:*

The WP, 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, this assumption of zero retardation is 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 1.1.01.09.14.00.

TSPA Disposition: N/A

Basis for Screening Decision:

A more complete discussion of the internal scenarios for fuel degradation is discussed in YMP (1998, Section 3.1). Both internal and external scenarios are discussed.

6.2.29 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: *Include.*

Screening Argument: 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 WP 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.

TSPA Disposition: The regression equation of the in-package chemistry 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 assumed equilibrated with the atmosphere to ensure maximum plausible oxygen and carbon dioxide conditions.

Basis for Screening Decision:

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 WP 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 WP, corrosion of the outer package steel will produce a reducing environment in the vicinity of the WP. 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 WP.

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).

6.2.30 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 equilibrium in the drift and waste environment.

Screening Decision: *Include* – reaction kinetics in the equilibrium model.

Exclude – reaction transients based on low consequence.

Screening Argument: In general, redox equilibrium does not exist in the waste environment, and it is not significant, because transport calculations using an equilibrium model were based on systems with multiple oxidation states, and, therefore, results are averaged over all oxidation states.

TSPA Disposition: Included reaction kinetics through quasi-equilibrium analysis in the in-package chemistry component of the waste form degradation model.

Basis for Screening Decision:

In general, redox equilibrium does not exist in the waste environment, and it is not significant, because transport calculations using an equilibrium model were based on systems with multiple oxidation states (CRWMS M&O 2000m), and, therefore, results are averaged over all oxidation states. The calculations of (CRWMS M&O 2000m) are part of the in-package chemistry program.

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.

The reaction and transient component of this FEP is excluded based on low consequence the reaction transients between time steps.

6.2.31 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: *Exclude* – based on low consequence.

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. Oxidizing conditions will not increase actinide solubility outside the WP 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.

TSPA Disposition: N/A

Basis for Screening Decision:

This FEP is excluded based on low consequence.

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. Oxidizing conditions will not increase actinide solubility outside the WP 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.

It is documented that 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 WP where salt buildup can occur in the early post closure stage of the repository. Heat generated by the WP can evaporate groundwater and concentrate salts near the WP. However, once radionuclides are transported a short distance away from the WP, ionic strength will decrease to groundwater levels and radionuclides will precipitate. Therefore, the possibility of a gradient to facilitate transport is very low.

Because of glass in the WP dissolving in any water entering the WP, the pH of the water will increase and therefore CO2 concentration can increase. CO2 complexes with actinides and can increase total-dissolved-species concentrations (Felmy and Rai 1999). Outside the WP, CO2 concentrations are set by equilibrium with CO2 in the atmosphere (CRWMS M&O 2000m). Higher CO2 concentrations in the WP than outside the WP will not create a chemical gradient to facilitate actinide transport outside the WP. If anything, the reverse will occur.

Because glass dissolution will increase the solution pH inside the WP, a pH gradient will be established going from high to lower outside the WP. Radionuclide concentrations are higher at high pH. (Felmy and Rai 1999). Therefore, this gradient will not facilitate actinide transport outside the WP.

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

In the “In-Package Chemistry Component” (CRWMS M&O 2000m) of the TSPA the WP is assumed to be a mixing cell without chemical gradients. 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 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 assuming cladding unzipping for CSNF; 100% failed cladding for DSNF; and very high reactive surface area for HLW glass.

6.2.32 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: *Exclude* – based on low consequence.

Screening Argument: The flow rate through a failed container was set a minimum value of 15 l/yr in performing the equilibrium-model calculations for release. These calculations are part of the in-package chemistry program. This rate is much greater than any effect that can be created by electrophoresis or electro-osmosis.

TSPA Disposition: N/A

Basis for Screening Decision:

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.

The flow rate through a failed container was set a minimum value of 15 l/yr in performing the equilibrium-model calculations for release (CRWMS M&O 2000m). This rate is much greater than any effect that can be created by electrophoresis or electro-osmosis (Soderman and Jonsson 1996).

6.2.33 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: *Exclude.* Low probability due to uncertainty in amount of radionuclide actually being chemically bound, reasonably conclude complete release of radionuclides.

Screening Argument: 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 proposed 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 appears to control their releases in the Argonne National Laboratory (ANL) unsaturated experiments. In fact, Tc release may depend on the corrosion rate of insoluble metallic particles (CRWMS M&O 2000c, Sec. 6.5, p. 74) 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.

The dose contribution from ^{237}Np becomes significant only beyond 10,000 years. 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.

U(VI) solids formed during corrosion of both unirradiated UO_2 and spent fuel are closely 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.

Experimental evidence from the Argonne National Laboratory (ANL) unsaturated experiments suggest 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 is apparently not released congruently with other matrix elements. 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 best defensible approach.

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 potentially mobile, available radionuclides.

TSPA Disposition: N/A

Basis for Screening Decision:

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 proposed 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 we need 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.

6.2.34 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: *Exclude* based on low consequence

Screening Argument: This FEP applied to emplacement of containers 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). The rock-water interactions and the Fe of the container is expected to be more controlling on U solubility than the interactions suggested in this FEP.

Because of the drip shield and the long-lived waste disposal container, rock is not expected to come directly in contact with the waste in the first 10,000 yr after repository closure. 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. The direct influence of rock fall is also excluded in cladding perforation in the first 10,000 yr (FEPs 2.1.02.24.00 and 2.1.07.01.00)

The indirect influence of the rock on in-package chemistry is included through the use of the J-13 well water (FEP 2.2.08.12.00).

TSPA Disposition: N/A

Basis for Screening Decision:

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 uranium 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.

These interactions have been considered in aspects of TSPA. The in-package chemistry calculations considered reaction of waste with J-13 groundwater, which is the groundwater that would be encountered by waste in the drift.

For these reasons, the consequences of waste-rock contact can be excluded from consideration in the TSPA.

6.2.35 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 re-dissolution 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: *Included* in radionuclide mobilization

Excluded in adjacent rock

Screening Argument:

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. 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 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 unsaturated zone (UZ) through increased sorption and reduced permeability and porosity.

TSPA Disposition: The waste-form-degradation conceptual model uses a single mixing cell to represent the waste form within the WP. This cell is scaled in size to the number of packages that have been breached within that WP sub-group. As more packages are breached, the size of the single WP 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 (CRWMS M&O 1998c, Section 6.5.2.2).

Basis for Screening Decision:

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 2000b, Section 4.4). 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.

6.2.36 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: *Exclude* – based on low probability.

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. These arguments suggest that complexation by organics is a minor consideration.

TSPA Disposition: N/A

Basis for Screening Decision:

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 maybe 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.

6.2.37 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. See following FEPs for more detailed discussion:

YMP No.	Topic
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

Screening Decision: *Include*

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, 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.

Basis for Screening Decision:

In-depth discussions can be found in the FEPs referenced above under YMP Primary FEP Description.

6.2.38 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: *Exclude* – based on low consequence.

Screening Argument: 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).

TSPA Disposition: N/A

Basis for Screening Decision:

The amount of heat produced by radioactive decay is substantial. As shown in 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; CRWMS M&O 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 1.3×10^{13} J or

1/100th of the heat energy produced in one year by all DSNF and HLW (assuming DSNF and HLW is ~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-Reator by an order of magnitude (see FEP 2.1.02.01.00) and thus bound any thermal effects on waste degradation.

6.2.39 Temperature Effects/Coupled Processes in Waste and EBS – YMP No. 2.1.11.04.00

YMP Primary FEP Description: This FEP broadly encompasses coupled-process effects of temperature changes within the waste and in the EBS. Technical discussions relevant to this FEP are provided individually for each relevant process. For details see other primary FEPs under 2.1.11, including:

YMP No.	Topic
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

Screening Decision: *Include*

Screening Argument: The heat released by the waste will increase the temperatures in the repository. These higher temperatures will affect the thermal, hydrological, chemical, and mechanical behavior of the waste and EBS.

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) As 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 (NFGE) models to other aspects within the NFGE 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.

Basis for Screening Decision:

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 as is a generic restatement of the 2.1.11 category. Some 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 (YMP No. 2.1.11.04.01), and coupled processes (YMP No. 2.1.11.04.06). Detailed discussions of more focused temperature/coupled effects are included in other 2.1.11 FEPs.

6.2.40 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: *Include*

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 and 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 2.1.11.07.00 and 2.1.11.01.00, respectively.

TSPA Disposition: Thermally induced stresses 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).

Basis for Screening Decision:

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, it 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 and 2.1.11.01.00, respectively.

6.2.41 Thermally Induced Stress Changes in Waste and EBS – YMP No. 2.1.11.07.00

YMP Primary FEP Description: Thermally induced stress changes in the waste and EBS may affect performance of the repository. Relevant processes discussed elsewhere include rockfall (YMP No. 2.1.07.01.00); drift stability (YMP No. 2.1.07.02.00); changes in physical properties of the disturbed rock zone around the repository (YMP No. 2.2.01.02.00); changes in physical properties of the surrounding rock (subentries under YMP No. 2.2.06.00.00); and stress corrosion cracking of cladding (YMP No. 2.1.02.21.00).

Screening Decision: *Include* thermally induced stress changes in near-field environment. *Exclude* thermally induced stress changes in the waste and packaging based on low consequence (by design).

Screening Argument: Repository heat at Yucca Mountain will drive the mechanical and chemical evolution of the repository and the mountain, 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.

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 considered.

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: Thermally induced stresses are included in the near-field modeling (CRWMS M&O 2000a). Their effects are implicitly included in 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. Thermally induced stress changes in the waste are not considered because waste packaging is designed to withstand expected repository temperature and temperature induced stresses.

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.

Basis for Screening Decision:

Repository heat at Yucca Mountain will drive the mechanical and chemical evolution of the repository and the mountain, 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 that could alter and/or enhance existing pathways.

Changes in the *in situ* stress field that will affect the near-field environment are discussed in the secondary FEP YMP No. 2.1.11.07.01. 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 WP 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. (See FEP concerning stress corrosion cracking of cladding, YMP No. 2.1.02.21.00).

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.

6.2.42 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. See FEP YMP No. 2.1.10.01.00 for a more specific discussion of microbial processes and sub-entries under 2.1.09 for a discussion of chemical processes and conditions.

Screening Decision: *Include.* Thermal effects in waste/repository chemistry models where data are available.

Exclude Based on low consequence: For many radionuclides, temperature specific solubility data is not available. Thermal effects on microbiological activity excluded since microbiological FEP is excluded

Screening Argument: The thermal load at Yucca Mountain results from an distributed, exponentially decaying heat source that is inserted over 30–50 years. The hydrogeologic response and evolution of Yucca Mountain are driven by the thermal load. Thermal effects will alter the local environment inducing chemical and microbiological changes. These changes are a function of temperature.

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). See FEPs in category 2.1.09 for details on these strongly coupled thermal-geochemical processes and conditions and references CRWMS M&O (1998a, 1998b) for extensive discussions on thermal hydrology and near-field geochemical environment.

Two key thermal constraints on microbial growth are relative humidity (RH) and temperature thresholds that limits the start of microbial activity until the boiling period is over. Even though microbes could be sterilized out of the drifts during the highest temperature period, because they are present in the water-rock system they will return as water drips back into potential drifts (CRWMS M&O 1998a, Section 4.2.3.2.4).

The evolution of microbial communities within the drift is intended to be a bounding assessment on the masses of microbes produced. 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).

Basis for Screening Decision:

The thermal load at Yucca Mountain results from an areally distributed, exponentially decaying heat source that is inserted over 30–50 years. The hydrogeologic response and evolution of Yucca Mountain are driven by the thermal load. Thermal effects will alter the local environment inducing chemical and microbiological changes. These changes are a function of temperature.

The temperature of the subsurface environment will greatly affect or limit the type of bacteria present, based on the optimum growth band of the microbe. There are five temperature classifications of bacteria: psychrophiles, facultative psychrophiles, mesophiles, thermophiles, and hyperthermophiles. During elevated temperatures in the Yucca Mountain repository, the microbial population would be dominated by thermophiles and hyperthermophiles; later, as the repository cooled, the repository would be dominated by mesophiles (CRWMS M&O 1998a, Section 4.2.1.3.3).

It is expected that in Yucca Mountain the concrete components 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 reaching the waste packages. However, current design does not include extensive use of concrete.

6.2.43 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: *Include* – The amount of water reaching the EBS and eventually the waste is based on hydrologic calculations that consider the effects of temperature.

Exclude – Two-phase flow within the waste is not included based on low consequence. *Exclude* – Thermally driven single-phase flow within the waste is not included based on low consequence.

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). A number of Yucca Mountain FEPs are devoted to the complexities of two-phase flow.

Also see YMP Nos. 2.2.10.10.00 and 2.2.10.11.00 for a discussion of thermally driven flow in the surrounding rocks of the unsaturated zone.

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) ‘Ab*straction of Near Field Environment Drift Thermodynamic Environment and Percolation Flux*”. (See also FEPs 2.2.10.10.00 and 2.2.10.11.00 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 WP was not discretized to account for differences in water chemistry caused by two-phase flow and local temperature differences.

Basis for Screening Decision:

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 WP environment. The temperature variations that could occur inside the waste package prior to 1000 yr (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 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. (See FEPs 2.1.11.10.00 and 2.1.09.08.00 on thermal and chemical diffusion processes, which are also excluded in the In-Package Chemistry Component.)

6.2.44 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: *Exclude* – based on low consequence

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.

Using the bounding argument detailed below, expected temperature gradients in the repository will be small enough that the effects of Soret diffusion are expected to 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: N/A

Basis for Screening Decision:

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 9 and 10, respectively, are given by Bird et al. (1960, pp. 574 and 575):

$$j_{Az}^{(T)} = -\frac{c^2}{\rho} M_A M_B D_{AB} \frac{k_T}{T} \frac{dT}{dz} \qquad \text{(Eq. 9)}$$

$$j_{Az}^{(x)} = -\frac{c^2}{\rho} M_A M_B D_{AB} \frac{dx_A}{dz} \qquad \text{(Eq. 10)}$$

Here, $j_{Az}^{(T)}$ is the diffusive mass flux driven by temperature gradients, and $j_{Az}^{(x)}$ is the diffusive 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):

$$k_T = \sigma\, x_A x_B T \qquad \text{(Eq. 11)}$$

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 9, 10, and 11, the following ratios can be derived:

$$\frac{j_{Az}^{(T)}}{j_{Az}^{(x)}} = \frac{\sigma\, x_A x_B \frac{dT}{dz}}{\frac{dx_A}{dz}} = \sigma\, x_A x_B \frac{dT}{dx_A} = \sigma\, x_A (1 - x_A) \frac{dT}{dx_A} \qquad \text{(Eq. 12)}$$

where, for a binary system, $x_B = 1 - x_A$.

We use as a bounding scenario, diffusion through the EBS. 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 12 between T_{WP}, $x_{AWP,}$ and T_{DW}, x_{ADW} gives Equation 13:

$$\sigma\,(T_{DW} - T_{WP}) = \ln\left[\frac{x_{ADW}}{1 - x_{ADW}}\frac{(1 - x_{AWP})}{x_{AWP}}\right] \qquad \text{(Eq. 13)}$$

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 13 to solve for σ, we find Equation 14 that the Soret diffusion effect will be negligible when:

$$\sigma << \ln\left[\frac{x_{AWP}}{1 - x_{AWP}}\frac{(1 - x_{ADW})}{x_{ADW}}\right](T_{WP} - T_{DW})^{-1} \qquad \text{(Eq. 14)}$$

If we let component A be 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 we can approximate this result by using Equation 15:

$$\sigma << \ln\left[\frac{x_{AWP}}{x_{ADW}}\right](T_{WP} - T_{DW})^{-1} \qquad \text{(Eq. 15)}$$

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 11. The values of σ range 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:

$$x_{AWP} >> 1.25 x_{ADW} \qquad \text{(Eq. 16)}$$

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.

6.2.45 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 of the repository, and could affect radionuclide transport. This FEP aggregates all types of gas generation into a single category. Technical discussions specifically addressing unique groups of gases are presented in separate YMP FEPs:

YMP No.	Topic
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

Screening Decision: *Exclude*–based on low consequence.

Screening Argument: Because the repository is located in the unsaturated zone (UZ), which is connected to the surface via fractures, fault zones, and a highly porous matrix, gas produced by any mechanism is expected to escape through diffusion and/or advective transport. Therefore, gas escaping from the waste packages to the repository will not noticeably affect repository pressures. In particular, hydrogen, which may be the most likely gas to be produced (by means of anoxic corrosion) and is extremely mobile and buoyant, would diffuse easily up through the overlying rocks. Furthermore, gas diffusion alone is rapid enough to ensure homogeneity, which reduces the chances of problematic build up of high concentrations of reactive gases.

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.

TSPA Disposition: None.

Basis for Screening Decision:

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 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. Technical discussions for individual gas types are presented separately:

YMP No.	Topic
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

Because the repository would be in the UZ, which is connected to the surface (DOE 1998b, Section 11.4.3) gas produced by any mechanism is expected to escape. Gas diffusion alone is rapid enough to ensure homogeneity (CRWMS M&O 1999h, Section 3.3.4), thereby preventing corrosive gases from accumulating and exacerbating degradation of waste packages and other components. 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.

Where gas could be produced, gas permeability, as measured in pneumatic tests (Weeks 1978), is believed to be adequate to allow enough escape through diffusion and/or direct gaseous transport so as not to permit significant build up of gas at any period of time. Gaseous diffusion alone is rapid enough to ensure homogeneity at the drift scale (CRWMS M&O 1999h, Section 3.3.4). Additionally, hydrogen, which is an extremely mobile and buoyant gas, would diffuse easily up through the overlying formations via the matrix, fractures, or fault zone.

6.2.46 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. He production might cause local pressure buildup in cracks in the fuel and in the void between fuel and cladding, leading to cladding failure.

Screening Decision: *Exclude* – The effects of He gas generation from performance-assessment calculations on the basis of low consequence to behavior of the waste package as part of overall repository performance assessment.

Screening Argument: Additional He production due to alpha (nuclear) decay of waste (actinides) may add sufficient pressure within the waste package to increase internal stresses and result in increased rates of package degradation and failure with release of radionuclides into the surrounding media and, ultimately, the groundwater.

TSPA Disposition: N/A

Basis for Screening Decision:

Radioactive decay of alpha-emitters (actinides such as Pu, U, Th, etc.) results in the creation of He gas within the waste form. The number of alpha particles resulting from the decay of these actinides can be upper bounded by determining the integrated decays measured in activity (curies) of each of these radionuclides over 10,000 years. Each result is then multiplied (from each radionuclide) by the number of alpha particles per curie (i.e., 3.7×10^{10}), and finally

summed over all pertinent radionuclides (e.g., ^{234}U, ^{230}Th, ^{226}Ra, etc.). The molar inventory is calculated by dividing by Avogadro's number (~6.02×10^{23} [atoms/mole]).

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, "Since the temperature is down to near background temperature, the He production will not contribute to cladding failure. In earlier cladding code He pressures were modeled and removed because of the small pressure increase." Quoting a previous write-up drawn from an early cladding model that addresses helium production, CRWMS M&O (1998c) continues:

> Both DHC [delayed hydride cracking] and strain cladding failures are driven by the cladding stress which may be caused by the internal gas (including fission product gases) pressure buildup. The gas pressure will slowly increase over time by the production of helium (He) which is produced in nuclear decay as an alpha particle.

Manaktala (1993, p. 3-12, Fig. 3-4) presents the helium pressure change as a function of time. That pressure-change profile (after being adjusted for temperature) was used in this analysis. The pressure-change curve (in log-log coordinates) was approximated by two linear equations, one for below 9,000 years and one for above that time. The helium partial pressure is added to the fission-gas pressure at each time step and a new stress is calculated. Incorporation of the helium gas produced does not affect the cladding failure rates because the clad is sufficiently cool by the time any significant amount of helium is produced. In addition, the reader is referred to Table 6-20 in CRWMS M&O (1997b).

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.

6.2.47 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: *Exclude* – based on low consequence.

Screening Argument: A repository in the UZ in Yucca Mountain is expected to be connected to the atmosphere and to be operating under oxic conditions. Some gases generated by metal corrosion could interact with the containers or escape from the drifts. Related FEPs are discussed in:

YMP No.	Topic
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

The design of the waste canisters is currently 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 1999e). 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 "Hydride Embrittlement of Cladding" (FEP 2.1.02.22.00) which discusses this issue.

TSPA Disposition: N/A

Basis for Screening Decision:

A repository in the UZ in Yucca Mountain is expected to be connected to the atmosphere and to be operating under oxic conditions. Some gases generated by metal corrosion could interact with the containers or escape from the drifts. Related FEPs are discussed in:

YMP No.	Topic
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

The design of the waste canisters is currently 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).

Because the repository would be in the UZ, which is well connected to the land surface, 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 1998b, 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.

6.2.48 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: *Exclude* – based on low consequence.

Screening Argument: Gas generation from microbial degradation has been eliminated on the basis of low consequence. 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.

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 and, because of the air permeability of Yucca Mountain, be diluted before it could accumulate in quantities of concern.

TSPA Disposition: N/A

Basis for Screening Decision:

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 and. 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).

Gas generation from microbial degradation has been eliminated on low consequence. 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.

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. 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 be diluted before it could accumulate in quantities of concern (DOE 1998b, Section 11.3.2).

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, vol. 3, Section 3.3.1.3). 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).

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 Document (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.

6.2.49 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 flow paths, and two-phase flow conditions may be important.

Screening Decision: *Exclude* – based on low consequence.

Screening Argument: Only $^{14}CO_2$ is considered as significant enough to be of concern and it is not expected to be part of the standard.

Also, atmospheric transport of radionuclides has been shown to be insignificant and, therefore, can be excluded from the TSPA on the basis of low consequence. Those radionuclides, such as ^{14}C, that have a potential to be transported in the aqueous phase as well as in the gaseous phase can be conservatively concluded to be transported entirely in the aqueous phase.

TSPA Disposition: N/A

Basis for Screening Decision:

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. For a repository in the UZ, most gases escape to the atmosphere. Usually only $^{14}CO_2$ is considered as significant enough to be of concern (CRWMS M&O 2000j). Bubbles and other interactions with the buffer do not apply at this site. It is reasonably concluded that any form of CO_2 goes into aqueous solution. The models do not consider two-phase flow conditions.

Based on the preliminary screening decision for YMP No. 3.2.10.00.0, "Atmospheric Transport of Contaminants," it is conservative to conclude that all ^{14}C remains in the aqueous phase. YMP No. 3.2.10.00.00, addressed in the "TSPA System-Level FEPs AMR," (CRWMS M&O 2000r), concludes that if all gaseous radionuclides, including all ^{14}C incorporated in CO_2 gas, were to be released as a gas through the atmospheric pathway, the additional contribution to expected annual dose at the location of the critical group would be insignificant (EPA 1999, Section 9.2.4). Thus, atmospheric transport of radionuclides has been excluded from the TSPA on the basis of low consequence. Those radionuclides, such as ^{14}C, that have a potential to be transported in the aqueous phase as well as in the gaseous phase can be conservatively concluded to be transported entirely in the aqueous phase.

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).

6.2.50 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: *Exclude* – The effects of noble (Ar, He, Kr, Rn, Xe) and CO_2 and CH_4 gas generation can be excluded from performance-assessment calculations on the basis of low consequence to waste package behavior within the overall repository performance assessment.

Screening Argument: The gases trapped within the waste packages due to fission (in reactor) and radioactive decay processes (both pre- and post-closure), consisting of noble isotopes (Ar, Kr, Rn, Xe), CO_2, and CH_4, may add sufficient pressure within the waste package to increase internal stresses, resulting in increased rates of package degradation and failure, with release of radioisotopes into the surrounding media and, ultimately, the groundwater. Post-closure noble gases are generated as the result of (nuclear) spontaneous and induced fissions of certain actinides. Carbon dioxide (CO_2) may be generated as a result of an oxidizing environment and methane (CH_4) produced from organics.

Post-closure noble gasses are generated as the result of (nuclear) spontaneous and induced fissions of certain actinides. Carbon dioxide (CO_2) may be generated as a result of an oxidizing environment and methane (CH_4) produced from organics. Sufficient pressure may be generated within the waste package (increasing internal stresses) to result in increased rates of package degradation and failure.

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 (CRWMS M&O 2000j).

TSPA Disposition: Exclude

Basis for Screening Decision:

CRWMS M&O (1997a) demonstrated in an analysis that the maximum pressure of 1.01 MPa was of no concern for the TSPA-VA waste-package design. However, previous studies (Sanchez et al. 1998) have shown that the number of additional fissions that occur over 10,000 years as the result of potential criticalities is insignificant when compared to the fission gases produced from fuel burnup.

Studies cited in chapter 6 of the TSPA-VA Technical Basis Document (CRWMS M&O 1998c, pp. 6-17 through 6-26, sections 6.3.1 through 6.3.1.1.2.6) documented the stresses produced as a function of burnup on spent-nuclear-fuel clad. In the same study for TSPA-VA, internal partial pressures from fission products were plotted as a function of time out to 1,000,000 years (Table 6-20). The peak partial pressure for fission products is about 7.1 MPa at 10 years and then decreases as the repository temperature drops. In the CRWMS M&O study (1997a), an internal clad pressure of 8.4 MPa was reasonably concluded (given an added conservative safety factor).

Because of burnup, the makeup of the fission product gases consists predominantly of stable (end of decay chain) isotopes of the noble elements Ar, Kr, and Xe (due to relatively short half-lives of radioactive fission-product gases [DOE 1999b]). In general, the other radioactive gases in the repository disappear rapidly or quickly become negligible in terms of mass and thus volume. Specifically:

- Radon gas (^{222}Rn and ^{219}Rn) is short-lived (~3.8 days and ~4 seconds, respectively) as are any gaseous daughters (^{218}Po, half-life ~3.1 minutes and ^{215}Po, half-life ~microseconds). Thus, radon will not build up over regulatory time periods but will remain in secular equilibrium with its parent, radium.

- CO_2 will not be created in an anoxic environment (with He fill gas). When the package is breached, any CO_2 produced will be released fairly rapidly. Note, however, the amount of ^{14}C remaining in the inventory and the package-failure rate would limit the potential creation of CO_2 to very small amounts. Further, the amounts of ^{14}C transported to the receptor are extremely small so that the consequence for risk is negligible.

- CH_4 should not be produced because there are essentially no organics in the waste (DOE 1999a). Of the noble gases, ^{85}Kr has a significant initial inventory but because of its half-life (~10 yr) its concentration rapidly becomes insignificant. The major gas constituents trapped inside a waste package will be stable noble gases produced before disposed (CRWMS M&O 1998c).

The internal pressures on the waste package interior from the predominantly noble gases inside the waste packages may be bounded by reasonably concluding all of the gas inside the free volume of the fuel elements was released. The pressures inside the commercial packages were bounded by applying the ideal gas law to the free volumes of the waste package and fuel elements (CRWMS M&O 1997a).

If it is concluded that pressures inside commercial fuel elements bound the pressures created within any of the DOE-owned spent nuclear fuel (DSNF) fuel elements and that N-reactor fuel represents the worst case of all DSNF fuel types, a similar calculation shows that for DSNF the internal pressure caused by fission gases is bounded by:

$$P_{max,DSNF} \leq P_{max,CSNF} \times V_{free,CSNF}/V_{free,DSNF} \qquad \text{(Eq. 17)}$$

where:

$P_{max,DSNF}$ and $P_{max,CSNF}$ are the internal pressures generated by the gas inside the DOE-owned-fuel and commercial-fuel assemblies, respectively, and $V_{free,CSNFi}$ and $V_{free,DSNF}$ are the free (void) volumes within the commercial and DOE-owned waste packages, respectively.

Calculations for free volumes of commercial and DOE-owned waste packages have been done, and they give 4.55 m^3 and 6.19 m^3, respectively (the latter number for N-reactor fuel packaging configuration) (DOE 1998d, Figure 3-16). Since N-reactor fuel represents the worst case of any DOE-owned fuel categories (using the codisposal configuration for DSNF/HLW), pressure build-up in DOE-owned fuel is less than in commercial fuel.

Another aspect of this problem is the impact of internal gas pressure on clad degradation. Most clad failure (in reactors) occurs due to pitting and "unzipping," rather than from gradual rupture because of internal gas pressure or pressure buildup (CRWMS M&O 1998c, Section 6.3.1.1.2.6). If no credit is taken for DSNF cladding, then it may be assumed that the DSNF will have an immediate influence on the chemistry of the waste stream plume chemistry. But because DSNF is such a small percent of the total waste designated for storage, 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" (CRWMS M&O 2000e). The commercial fuel clad is designed with a low failure rate for a reactor environment and thus will not be significantly degraded by the pressures from these gases. Although this is not true, in general, for DSNF and HLW, not only is the burnup for these fuels relatively small, but no credit will be taken for cladding on DSNF.

For He gases, see FEP YMP No. 2.1.12.02.00, "Gas Generation (He) from Fuel Decay."

6.2.51 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: *Exclude* - based on low probability

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. 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 H_2O as described in YMP No. 2.1.13.01.00. But there is virtually no H_2O 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.

TSPA Disposition: N/A

Basis for Screening Decision:

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. 2.1.02.08.04). 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, p. 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.

Possibly gases, as well as water, could accumulate within the drift area, if there was a condensation cap or reduced permeability, (CRWMS M&O 1999b, Section 3.2.3; 2000b). 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 1999b, Section 3.2.1; 2000d). 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.

6.2.52 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: *Exclude* – based on low consequence

Screening Argument: Alpha, beta, gamma, and neutron irradiation of water leads to formation of highly reactive excited and ionized species which in turn can undergo various reactions. In pure water, the final products are hydrogen and oxidants. 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 alpha radiolysis have become negligible. Shielding calculations can show that intact clad 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 clad can be predicted based on semi-empirical methods to have minor consequence.

TSPA Disposition: N/A

Basis for Screening Decision:

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:

$$R = \int_0 dE/S(E) \qquad \text{(Eq. 18)}$$

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.

6.2.53 Radiation Damage in Waste and EBS – YMP No. 2.1.13.02.00

YMP Primary FEP Description: This category contains features, events, and processes (FEPs) related to nuclear radiation effects upon the waste and EBS in the Yucca Mountain Project (YMP) repository. 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. This FEP screening argument will estimate the radiolysis-enhanced corrosion of spent nuclear fuel due to emitted alphas based on micro-scale experiments, and compare them to dissolution rates predicted by YMP models. These radiolysis corrosion rates are based on commercial spent fuels which bound effects expected from U.S. Department of Energy (DOE) spent fuels since commercial fuel

receive substantially greater burnup rates. This FEP will identify that the radiolysis rate much less than the expected dissolution rates are thus insignificant.

Screening Decision: *Exclude* – The effects of radiolysis-enhanced dissolution of spent nuclear fuel are excluded from performance-assessment calculations on the basis of low consequence to the performance of the disposal system during the regulatory timeframe.

Screening Argument: 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. The extent to which they enhance the degradation is not significant, as identified below. However, for evaluation of the solubility of the Pu, Pa, and Cm, where the controlling solid is unknown, radiation damage to crystal lattice was assumed and an amorphous solid conservatively used (CRWMS M&O 2000y).

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.

DSNF and HLW are being evaluated with regard to their effect or contribution to the technical issues discussed in this FEP screening arguments.

TSPA Disposition: N/A

Basis for Screening Decision:

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 3. 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 19. This equation is superimposed upon the experimental data in Figures 3, 4, and 5.

$$CR = C_1 A^{C_2} e^{-C_3/A} \qquad \text{(Eq. 19)}$$

where

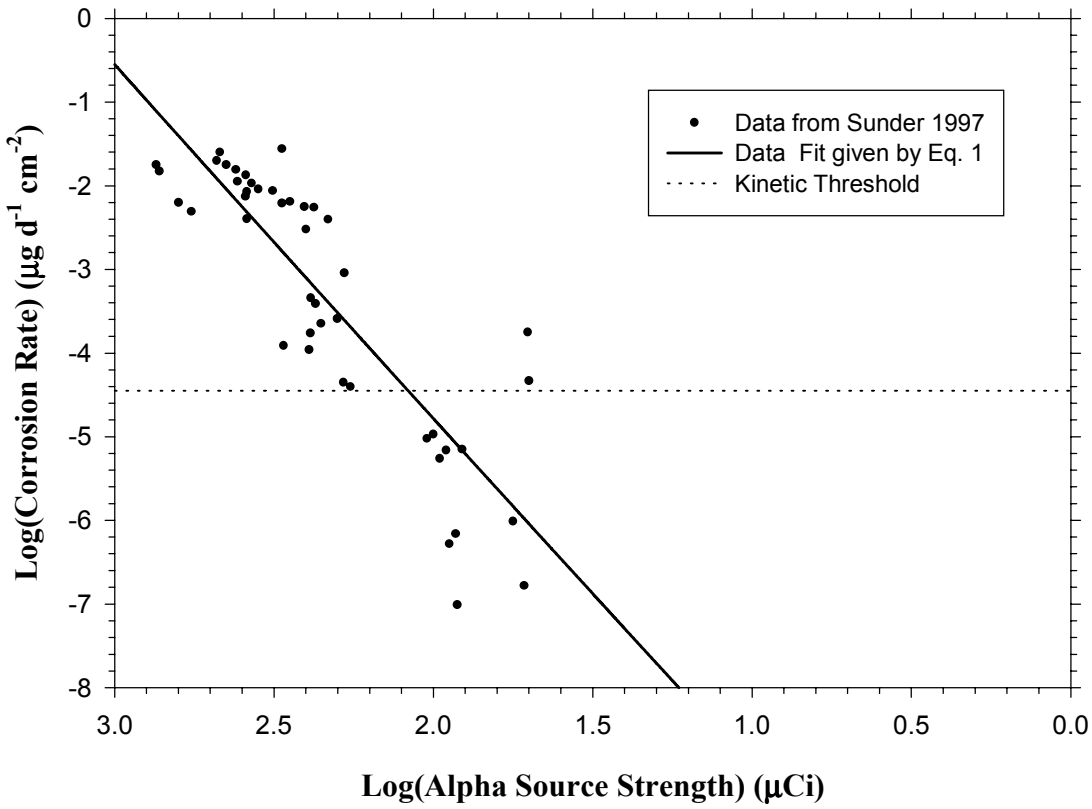
CR = *corrosion rate due to* α *radiolysis* (*gm/yr-m*^{2})
A = *alpha source strength* (*Ci/m*^{2})
C_1 = 4.0168303×10^{-1}
C_2 = 4.2341650×10^{0}
C_3 = $-2.8349524 \times 10^{-3}$

Tables 4 and 5 were generated to apply Equation 19. Table 4 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 5, 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 5 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 6 (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.

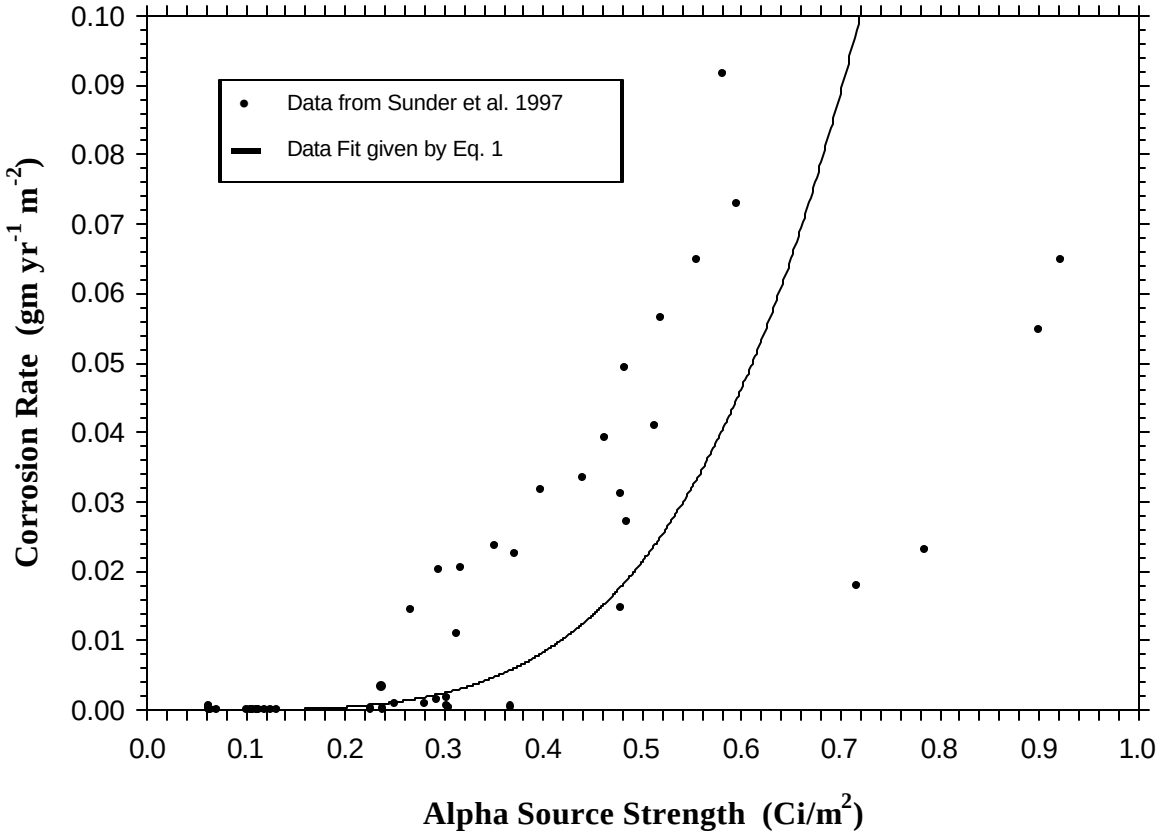
2. While high-level waste (HLW), metallic carbide, and ceramic spent nuclear fuels exhibit lower dissolution rates than that due to metal and oxide spent fuels, these DOE spent fuels/waste have corresponding burnups that are typically one-twentieth or less than that from commercial fuels. This can be 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.

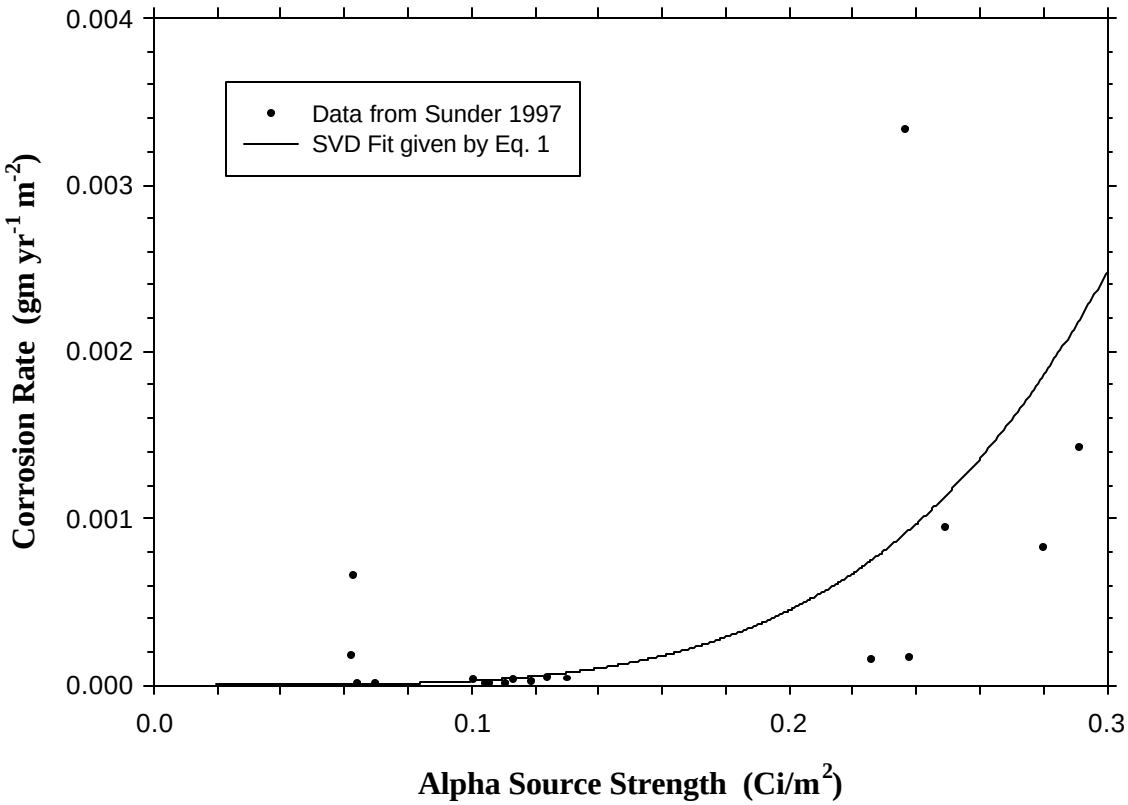
Note: The horizontal dashed line corresponds to the threshold above which kinetically controlled oxidative dissolution (corrosion) of UO_2 fuel occurs (after Sunder et al. 1997).

Figure 3. 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).



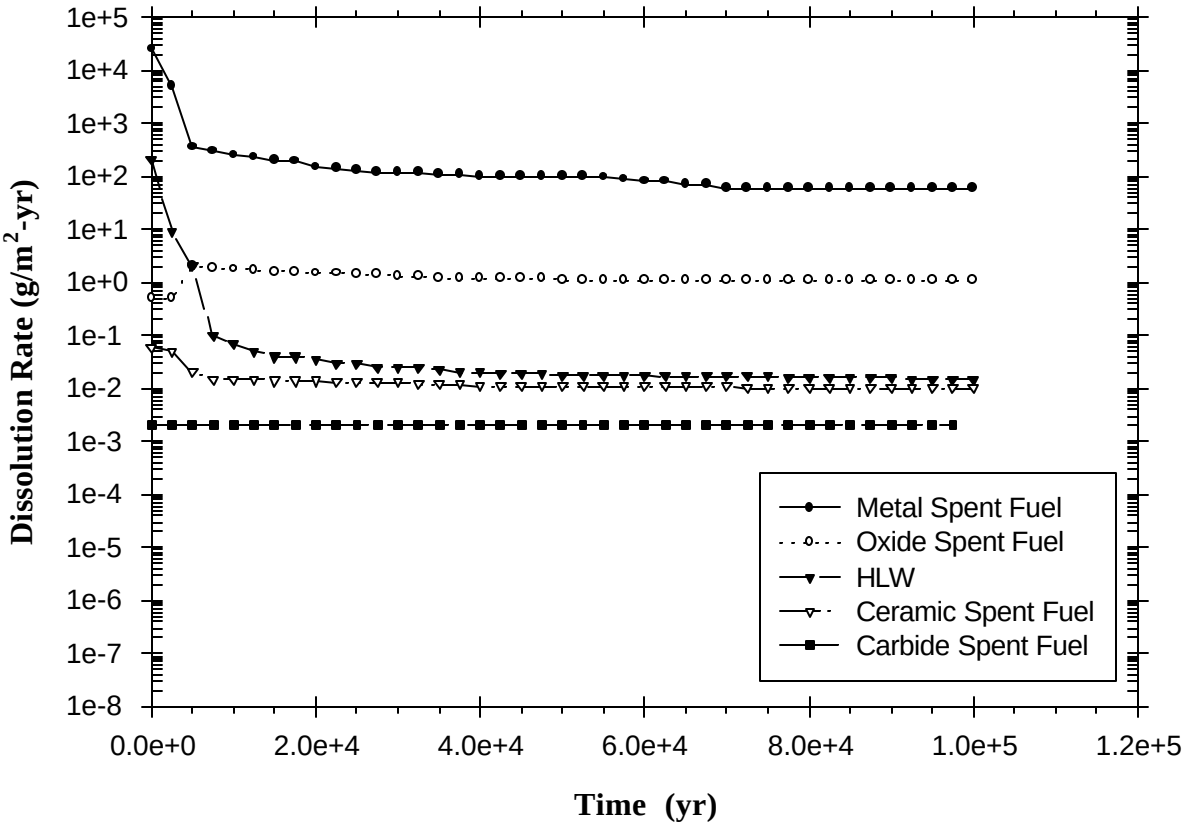
Note: Data taken from Figure 3, converted to conventional units, and plotted in linear-linear coordinates (after modifying data from Sunder et al. 1997).

Figure 4. 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 5. Expanded Range (alpha-source strength $\leq$ 0.3 Ci/m^2) of Corrosion Rates for Uranium Oxide using Alpha Radiolysis Rates from Figure 4



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

Figure 6. Dissolution Rates for High-Level Waste, Metallic Carbide, and Ceramic Spent Nuclear Fuel

Table 4. 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, 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 5. 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 4.

(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 19 on the data from Column 5. (Estimated corrosion rate values can be visualized from Figure 3.) Comparison of these data values with expected dissolution rates for high-level waste, metallic carbide, and ceramic spent nuclear fuel (see Figure 4) 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 FEP's 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.

6.2.54 Mutation – YMP No. 2.1.13.03.00

YMP Primary FEP Description: Radiation fields could cause mutation of microorganisms, leading to unexpected chemical reactions and impacts.

Screening Decision: *Exclude* – based on low consequence

Screening Argument: Microbes can affect the mobility of colloidal material as well as influence the rate of waste-package corrosion. Given present knowledge, estimates of the effects of microbes on corrosion processes are highly uncertain; the potential effects of mutated microbes are more uncertain. No analyses or experimental research have been performed to investigate this problem specifically. However, 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. Thus, exclusion of effects of mutated microbes from TSPA is probably conservative.

TSPA Disposition: N/A

Basis for Screening Decision:

The colloid-associated radionuclide concentrations from the repository may be sensitive to the presence of microbes, which could affect the amount of mobile colloidal material. The effects of microbes in the repository on the colloid-associated radionuclide concentrations can include: (1) changes in near field chemistry in "micro-environments" in and around waste packages, and/or canisters within the waste packages and/or canisters within the waste packages, (2) microbial oxidation of metallic iron (Fe) to produce Fe oxide colloids and aggregates, (3) microbially influenced corrosion (MIC), accelerating radionuclide release into the surrounding environment, and (4) decrease in the concentration of stable colloids by aggregating colloidal material used as a food source.

Considerable uncertainty surrounds the degree to which small, isolated populations of microbes affect chemical systems around the repository. In the absence of much organic material, microbial effects are believed to be small in comparison to changes induced in chemistry by global variables such as heat and radiation.

Regarding the process of mutation (Wilson and Bossert 1971, pp. 30–33) populations of microorganisms not destroyed by heat and radiation in the post-closure repository could in time *evolve* phenotypic traits very different from traits of the pre-closure populations; and it is possible that these altered phenotypes could be associated with unusual manifestations of effects (items 1–4 noted above). The process of mutation, however, is not sufficient to drive evolution in a definite direction; most mutations are either neutral or deleterious with regard to an organism's phenotype. Thus, mutational processes alone are highly unlikely to produce the aforementioned effects. The process dominating directional evolution of isolated populations of prokaryotes is *natural selection* which acts over successive generations to maintain the fitness of an organism for life in its environment; but the manner and direction in which selective forces "move" an organism's phenotype is not predictable except under the most closely controlled conditions (*artificial selection).*

In any case, if mutated microbes were to occur, it is unlikely that sufficient quantities will be available to accelerate corrosion rates significantly or have unpredictable effects. Also, as mentioned above, microbial action tends to increase colloid size, which would result in decreased colloid stability. Therefore, the potential effects of mutated microbes are excluded from TSPA.

6.2.55 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: *Include.*

Screening Argument: There is the possibility that fluids compositionally different from 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.

TSPA Disposition: N/A

Basis for Screening Decision:

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).

6.2.56 Radioactive Decay and Ingrowth – YMP No. 3.1.01.01.00

YMP Primary FEP Description: This category contains FEPs related to radioactive decay and ingrowth.

Screening Decision: *Include.*

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).

Basis for Screening Decision:

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 a conservative, reasonable conclusion is implemented. 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).

6.2.57 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: *Include* in waste package

Exclude outside waste package based on low consequence. Excluding isotopic dilution is conservative (bounding).

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.

Isotopic dilution during transport away from the waste package due to stable and/or naturally occurring isotopes of the same element would only tend to minimally decrease radiological effects and therefore can, conservatively, be excluded.

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.

Basis for Screening Decision:

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 Waste Inventory YMP No. 2.1.02.02.00 and Radioactive Decay and Ingrowth YMP No. 3.1.01.01.00 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 57 Miscellaneous WF FEPs.
- For 57 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 6 through 13 provide a summary of the Miscellaneous Waste-Form FEPs screening decisions and the basis for *Exclude* 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 proposed geologic repository (Sagar 1997). At present, NRC staff developed 10 Key Technical Issues (KTIs), nine of which relate to post-closure performance assessment (Table 14).

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 WF FEPs relate to a variety of KTIs, they relate dominantly to the first, second, and third issues listed in Table 14. Tables 15, 16, and 17 associate WF 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 6. 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	Include	WF	ANL-WIS-MD-000009 ANL-WIS-MD-000006	Average and bounding inventories of 27 isotopes for 3 waste categories (CSNF, DSNF, HLW) have been developed (different subsets of isotopes used for volcanic release, groundwater release, and human intrusion release); 24 isotopes from screening arguments based on human dose and 3 isotopes mandated by the Groundwater Protection Requirement of the proposed 40 CFR 197. This FEP is the topic of AMR, *Inventory Abstraction* (CRWMS M&O 2000j).
2.1.01.02.00	Codisposal/ collocation of waste	Include	WF	ANL-WIS-MD-000009	Codisposal/collocation of DSNF and HLW is included in the TSPA-SR analysis by separately modeling waste form degradation of both categories within a codisposal package.

Table 6. FEPs Related to the Radioisotope Inventory Component - (continued)
(Gray shading indicates Miscellaneous WF FEPs in this AMR.)

2.1.01.03.00	Heterogeneity of waste forms	Include	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	Exclude	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. The influence of the physical aspects of gas has generally been excluded since they would readily dissipate from the repository. Pressurization of cladding is discussed in cladding FEP 2.1.02.20.00.
3.2.07.01.00	Isotopic dilution	Include/ Exclude	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 based on beneficial consequences (conservative): 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.
3.1.01.01.00	Radioactive decay and ingrowth	Include	WF, UZ, SZ	ANL-WIS-MD-000009	Radioactive decay and ingrowth is either incorporated into the computer codes used for TSPA-SR analysis modeling, or the inventory of a daughter is artificially increased when generating the inventory in cases that are not explicitly modeled (conservative assumption).

Table 7. 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
2.1.01.04.00	Spatial heterogeneity of emplaced waste	Exclude	WF	ANL-WIS-MD-000009	Excluded based on low consequence. For evaluating the in-package chemistry, the time scale of convective transport and diffusion of aqueous species in the fully flooded waste package is rapid in relation to the alteration of the contents of the disposal package (waste form, cladding, and support material); consequently, the contents are assumed to be uniformly mixed. Spatial heterogeneity of the waste form throughout the repository is also excluded. WPs of CNSF and codisposed DSNF and HLW are assumed to be closely packed together so that variations of individual package temperatures (which can influence aspects of waste form degradation) will not vary significantly.
2.1.02.09.00	Void space (in disposal container)	Include	WF	ANL-WIS-MD-000009	Included: In the process-model for evaluating the In-Package Chemistry Component, the mixing cell volume is equal to the entire void volume in the disposal container.
2.1.02.10.00	Cellulosic degradation	Exclude	WF	ANL-WIS-MD-000009	Excluded based on low probability (credibility). 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	Include/ Exclude	WF, WP	ANL-WIS-MD-000009	After breach of the WP, the corrosion of inner structural stainless steel is important to determining the in-package chemistry and is, thus, included in the process model and as part of the uncertainty of the In-Package Chemistry Component. Prior to WP breach, internal corrosion is excluded because of low consequence. There will be no or minimum corrosion because of the specified inert gases in the package and limited water 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 itself. Other DSNF forms would have low water content.

Table 7. FEPs Related to the In-Package Chemistry Component - (continued)
(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
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and EBS	Include/ Exclude	WF, EBS	ANL-WIS-MD-000009	Include through the use of a series of linked one-dimensional flowpaths and mixing cells through the EBS, drip shield, waste package, and into the invert. Exclude based on beneficial consequence (conservative): transport pathways inside container excluded; 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	Include/ Exclude	WF, EBS	ANL-WIS-MD-000009	Included: Changes in flow volume into the WP as it degrades are included. Excluded based on low consequence: changes to hydrological properties of the waste form. Since the uncertainty caused by these changes are small in relation to the overall uncertainty of the in-package chemistry predictions.
2.1.08.10.00	Desaturation/ dewatering of the repository	Include	WF, NFE	ANL-WIS-MD-000009	Included through unsaturated flow modeling of near field environment (NFE), 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	Include/ Exclude	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: 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 7. FEPs Related to the In-Package Chemistry Component - (continued)
(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
2.1.09.02.00	Interaction with corrosion products	Include/ Exclude	WF, EBS	ANL-WIS-MD-000009	Included: The presence of corrosion product rind in included in determining the availability of water for radionuclide dissolution; the interaction between the volume of the rind and the unzipping of the cladding; chemical influence on the in-package chemistry; and sorption on mobile colloids. Excluded based on beneficial consequence (conservative): The potential beneficial effects from corrosion products which include the decreased advective and diffusive transport of radionuclides and the sorptive effects of immobile corrosion products have been conservatively excluded.
2.1.09.06.00	Reduction-oxidation potential in waste and EBS	Include	WF, EBS	ANL-WIS-MD-000009	The regression equation of the in-package chemistry 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 assumed equilibrated with the atmosphere to ensure maximum plausible oxygen and carbon dioxide conditions.
2.1.09.07.00	Reaction kinetics in waste and EBS	Include/ Exclude	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 of the reaction transients between time steps. The in-package chemistry process model, EQ3/6, assumes instantaneous equilibrium between changes in amounts of corrosion products available. Small enough time steps are taken to avoid errors exceeding uncertainty in thermodynamic parameters.

Table 7. FEPs Related to the In-Package Chemistry Component - (continued)
(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
2.1.09.08.00	Chemical gradients / enhanced diffusion in waste and EBS	Exclude	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, 100% failed cladding for DSNF, and very high reactive surface area for HLW glass).
2.1.09.09.00	Electrochemical effects (electrophoresis, galvanic coupling) in waste and EBS	Exclude	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.
2.1.09.11.00	Waste-rock contact	Exclude	WF, EBS	ANL-WIS-MD-000009	Excluded based on low consequence. Because of the drip shield and the long-lived waste disposal container, 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 reactions due to their dissolution kinetics. The direct influence of rock fall is also excluded in cladding perforation in the first 10,000 yr (cladding FEPs 2.1.02.24.00 and 2.1.07.01.00). The indirect influence of the rock on in-package chemistry is included through the use of the J-13 well water (in-package chemistry FEP 2.2.08.12.00).
2.1.11.04.00	Temperature effects / coupled processes in waste and EBS	Include	WF, EBS	ANL-WIS-MD-000009	Temperature effects on in-package chemistry such as pH are not 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). See more specific FEPs for details.

Table 7. FEPs Related to the In-Package Chemistry Component - (continued)
(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
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and EBS	Include/ Exclude	WF, NFE, EBS	ANL-WIS-MD-000009	Included: Thermal effects are included in waste/repository chemistry models that determine water and gas composition, precipitation/dissolution, mineral stability, phase equilibrium, and reaction rates when thermodynamic data are available as a function of temperature (i.e., uranium solubility). Excluded, based on low consequence: For many radionuclides, temperature specific solubility data is not available. Thermal effects on microbiological activity excluded since microbiological FEP excluded (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	Include/ Exclude	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 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	Exclude	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 Gap and Grain Release FEP 2.1.02.07.00).
2.1.12.01.00	Gas generation	Exclude	WF, EBS, UZ	ANL-WIS-MD-000009	Excluded based on low consequence. Any gas produced is expected to escape the repository because it is connected to the surface via fractures, fault zones, and a highly porous matrix. Gas generation within an unbreached WP is expected to be insignificant. Gas generation and pressurization of unbreached cladding is discussed in cladding FEP 2.1.01.20.00.

Table 7. FEPs Related to the In-Package Chemistry Component - (continued)
(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
2.1.12.02.00	Gas generation (He) from fuel decay	Exclude	WF, EBS	ANL-WIS-MD-000009	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.
2.1.12.03.00	Gas generation (H_2) from metal corrosion	Exclude	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	Exclude	WF, EBS	ANL-WIS-MD-000009	Excluded based on low probability (credibility) and consequence. 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.
2.1.12.06.00	Gas transport in waste and EBS	Exclude	WF, EBS	ANL-WIS-MD-000009, ANL-WIS-MD-000006	Excluded based on low consequence. Screening studies (waste inventory FEP 2.1.01.01.00) conclude that if all gaseous radionuclides, including all C-14 incorporated in CO_2 gas, were to be released through an atmospheric pathway, the expected annual dose at the location of the critical group would be insignificant. In TSPA-SR screening analysis, those radionuclides, such as C-14, that had a potential to be transported in the aqueous phase as well as in the gaseous phase were conservatively assumed to be transported entirely in the aqueous phase (waste inventory FEP 2.1.12.07.00).

Table 7. FEPs Related to the In-Package Chemistry Component - (continued)
(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
2.2.08.12.00	Use of J-13 Well water as a surrogate for water flowing Into the EBS and waste	Include	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 8. 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	Include	WF, Tec, WP	ANL-WIS-MD-000009	This FEP sets plausible particle sizes for the fragmented waste. Direct volcanic effects (i.e., radionuclides carried by ash plumes from volcanic eruptions) are modeled as described in the Disruption Events Report.
2.1.02.02.00	CSNF alteration, dissolution, and radionuclide release	Include	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	Exclude	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).
2.1.13.01.00	Radiolysis	Exclude	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. It is, therefore, not necessary to include radiolysis explicitly in the models.

Table 8. FEPs Related to the CSNF Matrix Degradation Component - (continued)
(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
2.1.13.02.00	Radiation damage in waste and EBS	Exclude	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*. 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.
2.1.11.05.00	Differing thermal expansion of repository component	Include	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*; influence of temperature is also discussed in FEPs 2.11.11.07.00 on cladding and 2.1.11.01.00 on HLW.
2.1.08.07.05	Waste-form and backfill consolidation	Exclude	WF	ANL-WIS-MD-000009	Excluded based on beneficial consequence since backfill and consolidation would tend to decrease the available reactive surface area. The potential deleterious effect of maintaining water contact with the waste form is already conservatively bound by assuming the waste package is fully flooded with water.

Table 9. 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
2.1.02.07.00	Gap and grain release of radionuclides after cladding perforation	Include/ Exclude	WF	ANL-WIS-MD-000009; ANL-WIS-MD-000007	Included: gap and grain boundary released when cladding first perforated or when WP fails for initially perorated cladding. Exclude Additional gap and grain-boundary inventory potentially produced while in repository, and any reactions which would mitigate the gap and grain-boundary inventory and, thereby, releases.

Table 9. FEPs Related to the CSNF Cladding Degradation Component - (continued)
(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
2.1.02.11.00	Waterlogged rods	Exclude	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.
2.1.02.12.00	Cladding degradation before YMP receives it	Include	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*.
2.1.02.13.00	General corrosion of cladding	Exclude	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	Exclude	WF	ANL-WIS-MD-000008	Excluded based on low consequence. 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	Exclude	WF	ANL-WIS-MD-000008	Excluded based on low probability. 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	Exclude	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000012	Excluded based on low consequence. The conditions for pitting corrosion such as $FeCl_3$ enhanced corrosion of the cladding are not present inside the WP. This FEP is the topic of AMR, *Clad Degradation Local Corrosion of Zirconium and its Alloys Under Repositiory Conditions.* Thus, pitting corrosion is, in general, excluded except for localized corrosion from fluorides, which is included as discussed in cladding FEP 2.1.02.26.00.

Table 9. FEPs Related to the CSNF Cladding Degradation Component - (continued)
(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
2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding	Exclude	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000012	Excluded based on low probability (credibility). 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 Repositiory Conditions.*
2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding	Exclude	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 (cladding FEP 2.1.02.16.00).
2.1.02.19.00	Creep rupture of cladding	Include	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*.
2.1.02.20.00	Pressurization from He production causes cladding failure	Include	WF	ANL-WIS-MD-000008	Both delayed hydride cracking (DHC) and creep perforation are 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	Include	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 2.1.02.16.00).
2.1.02.22.00	Hydride embrittlement of cladding	Exclude	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*.

Table 9. FEPs Related to the CSNF Cladding Degradation Component - (continued)
(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
2.1.02.23.00	Cladding unzipping	Include	WF	ANL-WIS-MD-000008; ANL-EBS-MD-000013; ANL-EBS-MD-000014	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. "Wet unzipping" is the topic of AMR, *Clad Degradation - Wet Unzipping: Release Rates from Breached Cladding and Potential Unzipping Velocity*. Unzipping of cladding in a dry environment is excluded based on low probability and 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*.
2.1.02.24.00	Mechanical failure of cladding	Include	WF	ANL-WIS-MD-000008	The CSNF Cladding Degradation Component includes perforation of cladding from severe earthquakes (once per million years).
2.1.02.27.00	Localized corrosion perforation of cladding by fluoride	Include	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*.
2.1.02.28.00	Diffusion-controlled cavity growth (DCCG) concerns	Exclude	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.
2.1.07.01.00	Rockfall (large block)	Exclude	WF	ANL-WIS-MD-000009, 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 (see also FEP 2.1.02.24.00).
2.1.09.03.00	Volume increase of corrosion products	Include/ Exclude	WF, WP	ANL-WIS-MD-000009	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.

Table 9. FEPs Related to the CSNF Cladding Degradation Component - (continued)
(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
2.1.11.07.00	Thermally induced stress changes in waste and EBS	Include/ Exclude	EBS, WF	ANL-WIS-MD-000009; ANL-WIS-MD-000007	Temperature influences creep perforations, and SCC failures, in the CSNF Cladding Degradation Component. Specifically, temperature differences through the cross-section of the WP are considered. Degradation of waste matrix of CSNF and HLW is also a function of temperature (FEPs 2.1.11.05.00 and 2.1611.01.00).

Table 10. 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	Include	WF	ANL-WIS-MD-000009	DSNF and its degradation is included in TSPA-SR analysis. 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 in TSPA-SR analysis for modeling of all DSNF. The inventory, however, is the weighted mass average of all the ~250 types of DSNF (see waste inventory FEP 2.1.01.01.00).
2.1.02.08.00	Pyrophoricity	Exclude	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.
2.1.02.08.04	Flammable gas generation from DSNF	Exclude	WF	ANL-WIS-MD-000009	Excluded based on low probability (credibility) and consequence. Only DSNF from the Peach-Bottom Core 1 has a potential to generate acetylene gas and, thereby, consequences from gas explosion. However, the Peach bottom core is only in ~100 Wps out of ~3900 DSNF WPs and of ~11,800 total WPs. Also, relatively good connection with the surface through fractures would eventually disperse any acetylene in the repository before explosive concentrations could be obtained.

Table 10. FEPs Related to the DSNF Degradation Component - (continued)
(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
2.1.02.25.00	DSNF cladding degradation	Exclude	WF	ANL-WIS-MD-000008	Excluded based on beneficial consequence (conservative). No protective performance is taken for DSNF cladding because more than 80 percent of the DSNF is from the N-Reactor and is in poor condition; up to 50 percent may be already perforated. Unzipping of the cladding is also conservatively neglected .
2.1.11.03.00	Exothermic reactions and other thermal effects in waste and EBS	Exclude	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 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	Exclude	EBS WF	ANL-WIS-MD-000009	Excluded based on low probability. Because of the absence of oxygen and water, there is a very low probability of explosion in an inert gas environment within the canister until the WP and DSNF canister is breached. After the canister is breached, any gas produced will disperse into the drift. The permeability of Yucca Mountain provides adequate connect to the surfaces such that flammable/explosive gases would be diluted and/or dispersed before they could reach explosive concentrations.

Table 11. 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	Include/ Exclude	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. Degradation model includes the minor phase separation permitted during production since parameters derived directly from dissolution experiments. Excluded: Extensive phase separation (precipitation/coprecipitates/solid solution phase) is excluded based on low probability (credibility argument) because phase separation is controlled during production. Extensive selective leaching is excluded based on low additional consequence; the rate of degradation used for the glass matrix is an upper bound on the actual rate of release and, therefore, bound any additional consequence due to selective leaching.
2.1.02.05.00	Glass cracking and surface area	Include	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 recrystalli-zation	Exclude	WF	ANL-WIS-MD-000009	Extensive recrystallization, excluded based on low probability (credibility). Controls are to be implemented as part of the waste production to ensure that extensive glass recrystallization and phase separation will not occur. Degradation model includes minor effects of recrystallization since parameters derived directly from dissolution experiments.
2.1.11.01.00	Heat output / temperature in waste and EBS	Include	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 12. 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	Include	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 particular 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	Exclude	WF	ANL-WIS-MD-000009	Excluded based on beneficial consequences (conservative). 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. These effects have been conservatively excluded; 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	Include/ Exclude	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 (conservative) based on beneficial consequence. Rind or altered zone may tend to prevent advective flow past the waste and, thus, reduces release of radionuclides; however, this effect has been conservatively excluded.

Table 12. FEPs Related to the Dissolved Radioisotope Concentration Component - (continued)
(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
2.1.09.13.00	Complexation by organics in waste and EBS	Exclude	EBS WF	ANL-WIS-MD-000009	Excluded based on low probability (credibility). Based on the preliminary Waste Acceptance Criteria, organic material will be excluded from the radioactive; hence, large sources of organic material will not be present in waste; only small amounts could occur 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 13. 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	Include/ Exclude	WF	ANL-WIS-MD-000009	Sorption is included on mobile colloidal material. Sorption on stationary material is excluded based on beneficial consequence.
2.1.09.14.00	Colloid formation in waste and EBS	Include	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	Exclude	WF, EBS	ANL-WIS-MD-000012	Excluded based on the low consequence: 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	Include	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.
2.1.09.16.01	Colloidal phases produced by coprecipitation in the waste package or EBS.	Include	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.

Table 13. FEPs Related to the Colloidal Radioisotope Concentration Component - (continued)
(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
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and EBS	Include	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.
2.1.09.18.00	Microbial colloid formation in the waste and transport in EBS.	Exclude	WF, EBS	ANL-WIS-MD-000012	Excluded based on beneficial consequence. Microbial activity has been screened out as a FEP (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.
2.1.09.19.00	Colloid transport and sorption in the waste and EBS.	Include/ Exclude	WF, EBS	ANL-WIS-MD-000012	The Colloidal Radioisotope Concentration Component assumes colloids reversibly sorb or irreversibly incorporate radionuclides; however, the modeling component only acts as a source-term and does not evaluate the transport (or filtration) of these colloids inside the WP.
2.1.09.20.00	Colloid filtration in the waste and EBS	Exclude	WF, EBS	ANL-WIS-MD-000012	Exclude based on beneficial consequence (conservative). 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.
2.1.09.21.00	Suspensions of particles larger than colloids	Exclude	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 groundwater interface	Exclude	WF	ANL-WIS-MD-000012	Excluded based on beneficial consequence. Although colloids may be retained at the air-water interface in unsaturated conditions, neglecting this

Table 13. FEPs Related to the Colloidal Radioisotope Concentration Component - (continued)
(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
					phenomenon is conservative; furthermore, the conceptual model of waste package is a fully saturated mixing cell.
2.1.09.23.00	Colloidal Stability and concentration dependence on aqueous chemistry	Include	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	Include	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.
2.1.10.01.00	Biological activity in waste and EBS	Exclude	WF, EBS, WP	ANL-WIS-MD-000012	Excluded based on low probability and low consequence. Preliminary analysis shows that sufficient quantities of microbes will not be available to beneficially affect colloid mobility or adversely accelerate corrosion rates significantly. Since organic matter is excluded from the waste based on the preliminary waste acceptance criteria.
2.1.13.03.00	Mutation	Exclude	EBS	ANL-WIS-MD-000009	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 14. 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 15. Mapping between Waste-Form FEPs and TSPAI Key Technical Issues
(See NRC 1998b, Sections 4.3 and 4.4 for discussion of KTIs)

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 assumptions are incorporated into the radionuclide release rates and solubility limits abstraction.	CLST 3 and CLST 4 correspond to Tables 7-13.	
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 6-13.	

Table 16. Mapping between Waste-Form FEPs and CLST Key Technical Issues
(See NRC 1998a, Sections 4.1, 4.3 and 4.4 for discussion of KTIs)

3. The rate at which radionuclides in Spent Nuclear Fue 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 proposed YM repository.	See Table 6	
	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 WPs.	See Table 7.	
	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 WP emplacement drifts.	See Tables 8-10, 12-13.	

Table 16. Mapping between Waste-Form FEPs and CLST Key Technical Issues
(See NRC 1998a, Sections 4.1, 4.3 and 4.4 for discussion of KTIs) (Continued)

	6. DOE has considered the compatibility of SNF and the internal components of the WP such as the basket Material in the evaluation of radionuclide release. Specifically the SNF should not compromise the performance of the WP.	2.1.03.06.00	Internal corrosion of waste container.
4. The rate at which radionuclides in High Level Waste glass are leached and released from the Engineered Barrier System.	1. DOE has taken into account all types of HLW glass planned for YM disposal.	See Table 6.	
	3. DOE has identified the range of environmental conditions to be expected inside breached WPs containing HLW glass and eventually certain types of SNF, as in the co-disposal WPs.	See Table 7.	
	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 WP emplacement drifts.	See Tables 11-13.	
	6. DOE has assessed the compatibility of HLW glass with internal components of the WP in the evaluation of radionuclide release, taking into consideration co-disposal with DOE-owned SNF in the same WP. Specifically, HLW glass should not compromise the performance of the WP.	2.1.03.06.00	Internal corrosion of waste container

Table 17. Mapping between Waste-Form FEPs and ENFE Key Technical Issues
(See NRC 1997, Section 4.4 for discussion of KTIs)

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 6, 7.	
	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 7.	
	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 7-13.	
	4. A nutrient and energy inventory calculation should be used to determine the potential for microbial activity that could impact radionuclide release.	See Tables 13.	
	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 13.	

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8.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

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8.3 SOURCE DATA, LISTED BY DATA TRACKING NUMBER

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.

SN0003T0810599.009. Revised Waste Package Radionuclide Inventory Approximations for TSPA-SR. Submittal date: 03/14/2000.

ATTACHMENT I -
AN ESTIMATE OF FUEL-PARTICLE SIZES FOR PHYSICALLY DEGRADED SPENT FUEL FOLLOWING A DISRUPTIVE VOLCANIC EVENT THROUGH THE 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 proposed 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 ATM103 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 ATM103 results alone. We 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 ATM103 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 (1998) 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).

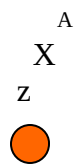
(b) Mass inventory values calculated using half-lives from the Decay Libraries from ORIGEN2 (Croff 1980).

(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.

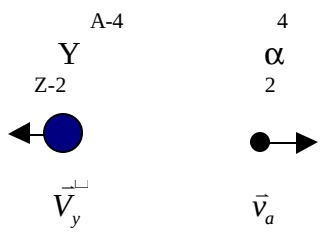
ATTACHMENT III -
ALPHA-RECOIL MECHANICS

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).



z=Atomic Number
A=molecular wt.

Figure III-1. Initial Center-of-Mass (COM) Frame Conditions for Alpha Emitting Radionuclide



$\vec{V}$ =Velocity of nuclide Y

$\vec{v}$ =Velocity of α particles

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.

Momentum Before = Momentum After (Eq. III-1)

M_y=Mass Nuclide Y
m_α=Mass of α particles

$$0 = M_y \vec{V}_y + m_a \vec{v}_a \qquad \text{(Eq. III-2)}$$

Thus the velocity of the recoil nucleus, in terms of the velocity of the alpha particle, is given by Equation III-3:

$$V_y = -v_a \left(\frac{m_a}{M_y} \right) \qquad \text{(Eq. III-3)}$$

where

m_a = 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 Equation B-4:

$$KE(Y) = \frac{1}{2} M_Y V_Y^2 = \frac{m_a}{M_Y}\left(\frac{1}{2} m_a V_a^2\right) = \frac{m_a}{M_Y} KE(a)$$

(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, we have: $KE(\alpha)$ = 4.196 MeV (Lederer and Shirley, 1978), $m_a \cong 4.0$, $M_{Th} \cong 234.0$. 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. These commercial fuels dominate the 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). These commercial fuels have considerable higher burnup than that expected for DSNF (e.g., DOE N-reactor fuels typically have less than one-twentieth of the burnup for commercial reactor fuels). 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 (assuming that 100% of the radionuclides are alpha emitters, which are the worst-case radiolysis generators). The only numerical values from these tables that are used in Table IV-3 are 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 4, the radiolysis rates are very insignificant. This means that should future estimates for the radionuclides in Tables IV-1 and IV-2 results in slightly different inventories, their differences will not significantly impact the finding of this FEPs screening argument.

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]	Radioactivity Inventory [Ci/Assembly]			
		(yr)	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	β⁻,γ,εβ⁺	2.39m	2.23E-04	1.29E-04	7.48E-05	4.34E-05
Ag108m	γ,εβ⁺,*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	β⁻,γ,εe⁻	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	β⁻,β⁺,ε	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	α,γ,*SF*,ε	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	α,γ,*SF*	8.5E3 a	1.03E-01	1.01E-01	9.98E-02	9.83E-02
Co60	β⁻,γ	4.76E3 a	6.06E-04	1.17E-09	2.27E-15	4.40E-21
Cs134	β⁻,γ,ε	5.271 a	6.33E-14	1.59E-28	0.00E+00	0.00E+00
Cs135	β⁻	2.065 a	3.50E-01	3.50E-01	3.50E-01	3.50E-01
Cs137	β⁻,γ	2.3E6 a	4.08E+03	4.05E+02	4.01E+01	3.98E+00
Eu150	γ,ε	36. a	2.99E-06	4.31E-07	6.23E-08	8.98E-09
Eu152	β⁻,γ,εβ⁺	13.48 a	7.23E-03	3.98E-05	2.20E-07	1.21E-09
Eu154	β⁻,γ,εγ	8.59 a	2.10E-01	6.57E-05	2.06E-08	6.44E-12
Eu155	β⁻,γ	4.71 a	1.90E-05	7.04E-12	2.60E-18	9.61E-25
Fe55	ε	2.73 a	3.25E-11	3.04E-22	0.00E+00	0.00E+00
H3	β⁻	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	β⁻,γ	10.73 a	1.76E+00	2.74E-03	4.26E-06	6.63E-09
Mo93	γ,ε,*e*⁻	~3.5E3 a	4.06E-02	3.98E-02	3.90E-02	3.83E-02
Nb91	β⁺,ε	7E2 a	1.77E-05	1.60E-05	1.44E-05	1.30E-05
Nb93m	*ITe*⁻	16.1 a	1.09E+00	9.29E-01	9.27E-01	9.26E-01
Nb94	β⁻,γ	2.0E4 a	8.36E-01	8.33E-01	8.30E-01	8.28E-01
Ni59	ε	7.6e4 a	2.09E+00	2.09E+00	2.08E+00	2.08E+00
Ni63	β⁻	100. a	1.26E+02	6.30E+01	3.15E+01	1.58E+01
Np237	α,γ	2.14E6 a	3.26E-01	3.98E-01	4.60E-01	5.12E-01
Np238	β⁻,γ	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	β^-,γ,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	α,γ,SF	2.87 a	8.86E-07	8.85E-07	8.85E-07	8.84E-07
Pu238	α,γ,SF	87.7 a	1.04E+03	4.73E+02	2.16E+02	9.83E+01
Pu239	α,γ,SF	2.410E4	1.77E+02	1.77E+02	1.76E+02	1.76E+02
Pu240	α,γ,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	α,γ,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	$\gamma,ITe-$	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	β^-,γ,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	$\gamma,ITe-$	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)]	Radioactivity Inventory [Ci/Assembly]			
		(yr)	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	$\alpha,\gamma,SF,\varepsilon$	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	1.17 m	6.32E-02	6.32E-02	6.32E-02	6.32E-02
Pb212	β^-,γ,IT	10.64 h	1.77E-03	6.55E-04	2.43E-04	9.01E-05
Pd107	β^-,γ	6.5E6 a	2.65E-02	2.65E-02	2.65E-02	2.65E-02
	β^-					

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	α,γ,ε	17.7 a	5.20E-06	1.03E-07	2.06E-09	4.10E-11
Pm146	β^-,γ,εγ	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	β^-,γ,εβ^+	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 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	4.47×10^{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)] 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 $\lambda=\ln(2)/\iota_{½}$, where $\iota_{½}$ 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]).