Results of Re-evaluation of FEPs Related to Implementing the ABD Glass Program

Spent Fuel and Waste Disposition

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APPENDIX E
NFCSC DOCUMENT COVER SHEET [1]

Name/Title of Deliverable/Milestone/Revision No.	Results of Re-Evaluation of FEPs Related to Implementing the ABD Glass Program
Work Package Title and Number	GDSA-FEPs Reevaluation for ABD Glass - SNL SF-22SN01030412
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This deliverable was prepared in accordance with Sandia National Laboratories *(Participant/National Laboratory Name)*

QA program which meets the requirements of
☒ DOE Order 414.1 ☐ NQA-1 ☐ Other

This Deliverable was subjected to:

☒ Technical Review	☐ Peer Review
Technical Review (TR)	**Peer Review (PR)**
Review Documentation Provided	**Review Documentation Provided**
☐ Signed TR Report or,	☐ Signed PR Report or,
☐ Signed TR Concurrence Sheet or,	☐ Signed PR Concurrence Sheet or,
☒ Signature of TR Reviewer(s) below	☐ Signature of PR Reviewer(s) below
Name and Signature of Reviewers	**Name and Signature of Reviewers**
Michael Wallace Michael Wallace 2-24-2022	

NOTE 1: *Appendix E should be filled out and submitted with each deliverable. Or, if the PICS: NE system permits, completely enter all applicable information in the PICS: NE Deliverable Form. The requirement is to ensure that all applicable information is entered either in the PICS: NE system or by using the NFCSC Document Cover Sheet.*

- *In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package documentation or the documented review of the issued document through the document control process) of the completion of the activity, along with the Document Cover Sheet, is sufficient to demonstrate achieving the milestone.*

NOTE 2: *If QRL 1, 2, or 3 is not assigned, then the QRL 4 box must be checked, and the work is understood to be performed using laboratory QA requirements. This includes any deliverable developed in conformance with the respective National Laboratory / Participant, DOE or NNSA-approved QA Program.*

NOTE 3: *If the lab has an NQA-1 program and the work to be conducted requires an NQA-1 program, then the QRL-1 box must be checked in the work Package and on the Appendix E cover sheet and the work must be performed in accordance with the Lab's NQA-1 program. The QRL-4 box should not be checked.*

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SUMMARY

One of the objectives of the United States Department of Energy Office of Nuclear Energy's Spent Fuel and Waste Science and Technology Campaign is to better understand the technical bases, risks, and uncertainties associated with the safe and secure disposition of spent nuclear fuel and high-level radioactive waste. Domestic defense and research activities have generated a few thousand metric tons of spent nuclear fuel and high-level radioactive waste, much of which has been or will be vitrified into high-level waste glass. The disposal of these materials is the responsibility of the Department of Energy (*Nuclear Waste Policy Act 1982)*.

In 2008, the Department of Energy submitted a license application to the United States Nuclear Regulatory Commission to authorize construction of a repository for spent nuclear fuel and high-level waste at Yucca Mountain, Nevada; the Department of Energy decided it would no longer pursue the license application process in 2010. The technical basis for the license application was consistent with plans for managing defense spent nuclear fuel and high-level waste as they existed in 2008. However, some of those plans have changed since 2008. This report describes one such change to the waste management planning for some Department of Energy-managed spent nuclear fuel under consideration and addresses how that change might affect the future repository licensing.

The Savannah River Site plans to reprocess defense spent nuclear fuel currently stored in their L-Basin via the Accelerated Basin Deinventory (ABD) Program**.** The previous plan for the L-Basin spent nuclear fuel was to dispose of it directly in the federal repository without reprocessing. Implementing the ABD Program will result in final disposal of approximately 900 fewer canisters of defense spent nuclear fuel and the production of approximately 521 more canisters of vitrified high-level waste glass with some specific differences from the planned high-level waste glass. Because the ^{235}U in the L-Basin spent nuclear fuel is not intended to be recovered, the fissile mass loading of the vitrified high-level glass waste form to be produced must be increased above the current value of 897 g/m^3 to a maximum of 2,500 g/m^3**.** Therefore, implementing the ABD Program would produce a variant of high-level waste glass—the ABD glass—that needs to be evaluated for future repository licensing, which includes both preclosure safety and postclosure performance.

This report describes the approach to and summarizes the results of an evaluation of the potential effects of implementing the ABD Program at the Savannah River Site on the technical basis for future repository licensing for a generic repository that is similar to Yucca Mountain and for one that is fully generic. This evaluation includes the effects on preclosure safety analyses and postclosure performance assessment for both repository settings. The license application for the proposed Yucca Mountain repository (DOE 2008), which is serving as a framework for this evaluation, concluded that the proposed Yucca Mountain repository would meet all applicable regulatory requirements. The evaluation documented in this report found that implementing the ABD Program is not expected to change that conclusion for a generic repository similar to Yucca Mountain or for a generic repository with respect to the preclosure safety analyses. With respect to the postclosure performance of a generic repository, no concerns were identified.

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ACRONYMS

2D, 3D	two-dimensional, three-dimensional
ABD	Accelerated Basin Deinventory
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
BET	Brunauer–Emmett–Teller
BWR	boiling water reactor
CDSP	codisposal (waste package)
CFR	Code of Federal Regulations
CSNF	commercial spent nuclear fuel
CTM	canister transfer machine
DFA	driver fuel assembly
DHLW	DOE high-level radioactive waste
DHLW-L	DOE high-level radioactive waste long (shorthand version of 5-DHLW/DOE Long)
DHLW-S	DOE high-level radioactive waste short (shorthand version of 5-DHLW/DOE Short)
DOE	(United States) Department of Energy
DSNF	DOE spent nuclear fuel
DWPF	Defense Waste Processing Facility
EA	Environmental Assessment
EBS	engineered barrier system
FEP	feature, event, and/or process
FFTF	Fast Flux Test Facility
GM	ground motion
GROA	geologic repository operations area
HAZOP	hazard and operability
HF	high fissile (glass)
HFIR	High-Flux Isotope Reactor
HLW	high-level radioactive waste
HM	high aluminum
LWBR	Light Water Breeder Reactor
MCNP	Monte Carlo N-Particle
MCO	multicanister overpack
MTHM	metric tons of heavy metal

MTR	Materials Test Reactor
NG	nominal glass
NRC	Nuclear Regulatory Commission
PA	performance assessment
PCSA	preclosure safety analysis
PCT	product consistency test
PGV	peak ground motion
PSHA	probabilistic seismic hazard analysis
PUREX	plutonium and uranium extraction
PVHA	probabilistic volcanic hazard analysis
PWR	pressurized water reactor
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SRS	Savannah River Site
SSC	structure, system, and/or component
std dev	standard deviation
TAD	transportation, aging, and disposal
TEDE	total effective dose equivalent
TEV	transport and emplacement vehicle
TMI	Three Mile Island
TMI-2	Three Mile Island Unit 2
TRIGA	Training, Research, Isotope, General Atomics
TSPA	total system performance assessment
TSPA-LA	total system performance assessment for the license application
U.S.	United States
USL	upper subcritical limit
UZ	unsaturated zone
UZrH	uranium-zirconium hydride
WAPS	Waste Acceptance Product Specifications
WP	waste package

RESULTS OF RE-EVALUATION OF FEPS RELATED TO IMPLEMENTING THE ABD GLASS PROGRAM

1. INTRODUCTION

One of the objectives of the United States (U.S.) Department of Energy (DOE) Office of Nuclear Energy's Spent Fuel and Waste Science and Technology Campaign is to better understand the technical basis, risks, and uncertainty associated with the safe and secure disposition of spent nuclear fuel (SNF) and high-level radioactive waste (HLW). Domestic defense and research activities have generated a few thousand metric tons of DOE spent nuclear fuel (DSNF) and HLW, much of which has or will be vitrified into HLW glass. The disposal of these materials is the responsibility of the DOE (*Nuclear Waste Policy Act 1982)*. Any repository used to dispose of the DSNF and HLW must meet requirements regarding both preclosure safety and long-term performance. Analyses used to determine whether a repository meets the long-term performance requirements must include features, events, and processes (FEPs) that could affect repository performance after repository closure. The data, design information, models, and analyses used to support both the preclosure safety assessment and the postclosure performance assessment (PA) are to be part of the license application for the repository.

In 2008, the DOE submitted a license application to the U.S. Nuclear Regulatory Commission (NRC) to authorize construction of a repository for SNF and HLW at Yucca Mountain, Nevada (DOE 2008); DOE decided it would no longer pursue the license application process in 2010. The technical basis for the license application was consistent with DOE's plans for managing DSNF and HLW as they existed in 2008. However, some of those plans have changed since 2008. This report describes one such change to the waste management plans for some DSNF that is being considered and addresses how that planned change might affect future repository licensing both for the preclosure safety analysis (PCSA) and the postclosure PA.

1.1 Background

One of the missions of the DOE's Savannah River Site (SRS) is to reprocess DSNF. The major waste stream from reprocessing is a vitrified glass waste form containing the actinides that were not removed for other purposes and most of the fission products; this glass waste form is classified as HLW.

SRS plans to reprocess DSNF currently stored in the L-Basin at SRS via the Accelerated Basin Deinventory (ABD) Program**.** The previous plan for this DSNF was to dispose of it directly in the federal repository without reprocessing. Implementing the ABD Program will result in repository disposal of approximately 900 fewer canisters of DSNF and the production of approximately 521 more canisters of HLW glass with specific differences from the planned HLW glass for disposal (Section 3.1.1**)**. Thus, the ABD program represents a change in DOE's waste management plans. Because the ^{235}U in the DSNF is not intended to be recovered, the fissile mass loading of the HLW glass waste form to be produced under the ABD Program is being increased above the current maximum value of 897 g/m^3 to a maximum of 2,500 g/m^3 **(**DOE 2019**).** Therefore, implementing the ABD Program is producing a variant of HLW glass—ABD glass—that needs to be evaluated for future repository licensing, which includes both the PCSA and the postclosure PA.

To keep the fissile mass load within acceptable limits established by the DOE for vitrified high-level waste forms (DOE 2012), the liquid waste from reprocessing the highly enriched L-Basin SNF will be combined with liquid waste from reprocessing SNF with lower enrichments before being vitrified to form the ABD glass waste form. Once SRS begins reprocessing the highly enriched L-Basin SNF, waste from reprocessing this SNF will be included in every batch of glass waste produced thereafter (DOE 2019; SRNS 2020). In total, between 3,600 and 4,500 canisters of glass waste forms that could have the higher fissile mass loading are planned to be produced.

1.2 Objectives

The objective of this report is to describe the approach and summarize the results of evaluating the potential effects of implementing the ABD Program at SRS on future repository licensing both for a generic repository that is similar to Yucca Mountain and for one that is fully generic. This evaluation includes the effects on the PCSA and the postclosure PA for both repository settings.

In this document, the term “effect” means something that causes a change or a difference in the configuration and/or technical bases of repository design, models/analyses, or modeling cases used to support the Yucca Mountain license application (DOE 2008), which was used as a framework for the evaluation described herein. For example, implementing the ABD Program would reduce the number of codisposal (CDSP) waste packages to be disposed by about 900 (Section 3.1.1). Changing the number of waste packages would, among other things, affect the inventory model used to support the Yucca Mountain license application in that the number of waste packages and their contents will be different compared to what was used previously in the original inventory model (DOE 2008).

1.3 Scope

The evaluation described in this report examines only the effects of implementing the ABD Program at SRS on future repository licensing; no other potential changes in DOE nuclear waste management plans (e.g., transportation) are addressed. Furthermore, the technical bases examined in this evaluation include the Yucca Mountain license application (DOE 2008), the evaluation of FEPs (SNL 2008a) for the Yucca Mountain total system performance assessment (TSPA[1]) (SNL 2010a), the preclosure nuclear safety design bases (BSC 2008h), and a screening analysis of criticality FEPs (SNL 2008d). The review also includes relevant supporting technical documents that form the bases of the detailed analyses underlying the primary documents. Note that this report uses a combination of English and metric units rather than a single system of units to maintain consistency with the evaluation framework provided by the Yucca Mountain license application and related documents.

1.4 Assumptions

The following assumptions were made in conducting this evaluation:

- There is no change to the waste packages that are intended to dispose of the ABD glass produced by SRS. That is, the waste packages that will be used to dispose of the canisters of ABD glass that result from implementing the ABD Program at SRS are the same as those that were previously planned to be used to dispose of HLW glass waste forms. See Section 3.1.1 for an explanation of the relationship between canisters of glass waste and waste packages.

[1] The TSPA supporting the Yucca Mountain license application is also known as the “TSPA-LA.” This report uses “TSPA” for simplicity.

- There are no changes to the ABD glass canister engineering characteristics compared to the HLW glass canisters currently being used at SRS. The engineering characteristics include canister materials of construction, thickness, and total weight (including HLW glass). Therefore, the canister response to potential hazards and associated mechanical and thermal stresses is assumed to remain unchanged for the ABD glass canisters.
- The approach used to evaluate a generic repository that is similar to Yucca Mountain is based on the framework already in place, namely the license application for the proposed Yucca Mountain repository (DOE 2008), with respect to both the PCSA and the postclosure PA.
- The PCSA that was part of the license application for the proposed Yucca Mountain repository (DOE 2008) can be used to evaluate the PCSA for a generic repository.
- The projected DSNF inventory does not include waste generated after 2035.
- Advanced Test Reactor (ATR) elements are disposed of in short canisters, with 20 elements per short canister.
- The DSNF in L-Basin that is to be reprocessed into HLW glass as part of the ABD Project (i.e., ABD glass) would have been disposed of in Yucca Mountain as DSNF in CDSP waste packages.
- The thermal output of the ABD glass to be produced is similar to that of the HLW glass that would otherwise have been produced. Specifically, most of the heat generated in the glass waste is produced by fission products ($^{137}Cs/^{137m}Ba$ and $^{90}Sr/^{90}Y$); based on process knowledge, the concentrations of these fission products would not change significantly because of implementing the ABD Program. Ranges of concentrations of the heat-generating actinides (^{238}Pu, ^{239}Pu, and ^{240}Pu) in samples of high fissile glass that simulate the ABD glass expected to be produced are not significantly different than those in nominal glass (Crawford et al. 2021, Tables C-6 and C-7). Data regarding changes in concentrations of the other two heat-generating radionuclides (^{241}Am and ^{244}Cm) are not available, but based on process knowledge, there is no reason to think that their concentrations would be significantly different as a result of implementing the ABD Program.
- The allocation of waste types between DSNF and DOE HLW (DHLW)[2], as described in DOE (2008), will not change. By law (*Nuclear Waste Policy Act of 1982*), the proposed Yucca Mountain repository was limited to 70,000 metric tons of heavy metal (MTHM): 63,000 MTHM to commercial SNF (CSNF) and commercial HLW (i.e., ~275 HLW canisters from the West Valley Demonstration Project) and 7,000 MTHM to DOE materials. In the license application, the 7,000 MTHM allocated to DOE materials was broken down as follows (DOE 2008, Section 1.2.1 and Table 1.5.1-1):
  - **DSNF**—2,268 MTHM
  - **DHLW**—4,667 MTHM
  - **Naval SNF**—65 MTHM

[2] DHLW is also known as defense HLW. This report sometimes uses the term "HLW" to refer to DHLW in a more general sense or to refer to HLW as a whole. When the difference between DHLW and commercial HLW is important, the report clearly makes the distinction between the two types of HLW.

Based on an assumed loading of 0.5 MTHM equivalence per canister for DHLW, the total allocated number of DHLW canisters is 9,334. The DOE sites that have or plan to have DHLW are SRS, Hanford, and Idaho National Laboratory. Note that the single value for heavy metal loading per DHLW canister was assumed regardless of length (10 to 15 ft), vitrification site (SRS, Hanford, or Idaho), and, for the purpose of this analysis, composition (nominal DHLW glass or ABD HLW glass). The value for West Valley HLW is different (i.e., 2.3 MTHM equivalence per canister), reflecting its classification as commercial HLW. Appendix D provides background on the bases for the assumed DHLW value as well as alternative approaches for calculating the heavy metal equivalence loading of HLW glass canisters. Alternative approaches based on total radioactivity or radiotoxicity result in a significantly lower heavy metal equivalence for HLW.

- The potential future regulatory regime assumed for this evaluation is similar to the regulatory regime that was in place at the time that the Yucca Mountain license application was prepared.

2. APPROACH

The approach to evaluating the effect of implementing the ABD Program on future repository licensing depends on whether the evaluation was being conducted for potential effects on the PCSA or on the postclosure PA. In Section 2.1, the approach to evaluating the effects of implementing the ABD Program on future repository licensing with respect to the PCSA is presented, while in Section 2.2, the approach to evaluating the effects of implementing the ABD Program on future repository licensing with respect to the postclosure PA is presented.

The first step in evaluating the effects of implementing the ABD Program was to meet with the SRS personnel responsible for this program to better understand how the characteristics of the ABD glass to be produced by SRS might affect future repository licensing. The following topics were discussed:

- Changes in the quantity of uranium isotopes (besides ^{235}U) in the glass waste form
- Changes in the quantity of fission products in the glass waste form
- How homogeneity in the glass waste form is ensured
- The number of glass waste canisters expected to have the higher fissile loading
- The reduction in the number of canisters of DSNF in need of disposal

2.1 Preclosure Safety Evaluation

To support identification of the effects of implementing the ABD Program on preclosure safety for a generic repository and for a generic repository that is similar to Yucca Mountain, a brief background on applicable regulatory requirements and the structure of the Yucca Mountain preclosure safety is provided. The background is followed by a description of the approach.

2.1.1 Requirements and Performance Objectives

In 10 Code of Federal Regulations (CFR) 63.2, the PCSA is defined as "a systematic examination of the site; the design; and the potential hazards, initiating events and event sequences and their consequences (e.g., radiological exposures to workers and the public). The analysis identifies structures, systems, and components important to safety." The following are key requirements of the PCSA of the geologic repository operations area (GROA):

- During normal operations, and for Category 1 event sequences, the annual dose to any real member of the public located beyond the boundary of the site may not exceed 15 mrem (10 CFR 63.111(a)(2) and 63.204). Category 1 event sequences are event sequences that are expected to occur one or more times before permanent closure of the GROA (10 CFR 63.2).
- "The geologic repository operations area must be designed so that, taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on, or beyond, any point on the boundary of the site will receive, as a result of the single Category 2 event sequence, …a TEDE [total effective dose equivalent] of 0.05 Sv (5 rem)…" (10 CFR 63.111(b)(2)). Category 2 event sequences are event sequences (besides Category 1 event sequences) that have at least one chance in 10,000 of occurring before permanent closure (10 CFR 63.2).

- "An analysis of the performance of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes the controls that are relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems. The analysis…must include…consideration of…(6) Means to prevent and control criticality…" (10 CFR 63.112(e)).

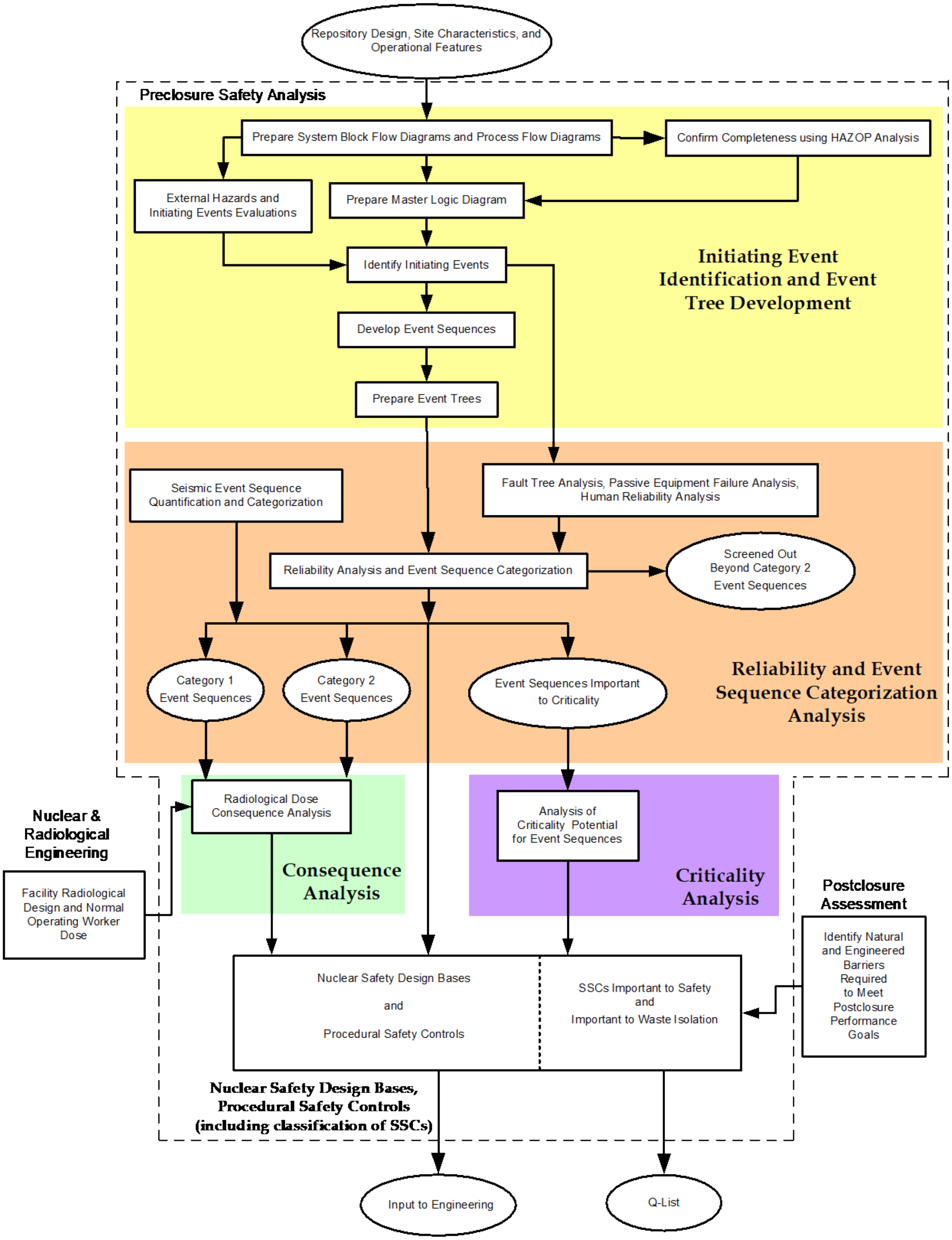
2.1.2 Preclosure Safety Process and Document Structure

Preclosure safety includes two components: PCSA and radiological engineering. The preclosure safety process is depicted in Figure 2-1 (USARS 2010). The PCSA covers the following areas, which form the foundation for deriving nuclear safety design bases, procedural safety controls, and Q-List:

- Initiating event identification and event tree development
- Reliability and event sequence categorization analysis
- Consequence analysis
- Criticality analysis

Radiological engineering covers the following areas:

- Source term development
- Surface and subsurface facilities shielding design support
- GROA worker dose calculations



NOTE: HAZOP = hazard and operability
SSC = structure, system, and/or component

Figure 2-1. Preclosure Safety Process

The *Yucca Mountain License Application Safety Analysis Report* (SAR) (DOE 2008) sections pertaining to the PCSA and radiological engineering are the following:

- Section 1.1 Site Description as It Pertains to Preclosure Safety Analysis
- Section 1.2 Surface Facility Structures, Systems, and Components and Operational Process Activities
- Section 1.3 Subsurface Structures, Systems, and Components and Operational Process Activities
- Section 1.4 Infrastructure Structures, Systems, Components, Equipment, and Operational Process Activities
- Section 1.5 Waste Form and Waste Package
- Section 1.6 Identification of Hazards and Initiating Events
- Section 1.7 Event Sequence Analysis
- Section 1.8 Consequence Analysis
- Section 1.9 Structures, Systems, and Components Important to Safety; Natural and Engineered Barriers Important to Waste Isolation; Safety Controls; and Measures to Ensure Availability of Safety Systems
- Section 1.10 Meeting the As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences
- Section 1.11 Plans for Retrieval and Alternate Storage of Radioactive Wastes
- Section 1.12 Plans for Permanent Closure, Decontamination, and Dismantlement of Surface Facilities
- Section 1.13 Equipment Qualification Program
- Section 1.14 Nuclear Criticality Safety

The NRC evaluation of the Yucca Mountain preclosure safety is documented in Volume 2 of the *Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada* (NRC 2015a). The specific safety evaluation report (SER) sections summarizing the NRC evaluation of the PCSA and radiological engineering are the following:

- Section 2.1.1.1 Site Description as it Pertains to Preclosure Safety Analysis
- Section 2.1.1.2 Description of Structures, Systems, Components, Equipment, and Operational Process Activities
- Section 2.1.1.3 Identification of Hazards and Initiating Events
- Section 2.1.1.4 Identification of Event Sequences
- Section 2.1.1.5 Consequence Analysis
- Section 2.1.1.6 Identification of Structures, Systems, and Components Important to Safety, and Measures to Ensure Availability of the Safety Systems
- Section 2.1.1.7 Design of Structures, Systems and Components Important to Safety and Safety Controls

- Section 2.1.1.8 Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences
- Section 2.1.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes
- Section 2.1.3 Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement, of Surface Facilities

Table 2-1 provides mapping between the SAR, SER, and preclosure safety area.

Table 2-1. Mapping between the SAR, SER, and Preclosure Safety Area

SAR Section(s)	SER Section(s)	Preclosure Safety Area
1.1	2.1.1.1	Site Description (input to preclosure safety)
1.2, 1.3, 1.4, 1.5	2.1.1.2, 2.1.1.7	Designs (surface, subsurface, structures, systems, and components, waste form, and waste package) and Operations (iteratively developed with preclosure safety)
1.6	2.1.1.1, 2.1.1.2, 2.1.1.3	Identification of Initiating Events
1.7	2.1.1.4	Reliability and Event Sequence Categorization
1.8	2.1.1.5	Consequence Analysis
1.9, 1.13	2.1.1.6, 2.1.1.7	Safety Systems
1.10	2.1.1.8	Radiological Engineering
1.11	2.1.2	Waste Retrieval (not a preclosure safety area)
1.12	2.1.3	Repository Closure (not a preclosure safety area)
1.14	2.1.1.2, 2.1.1.3, 2.1.1.4, 2.1.1.6, 2.1.1.7	Criticality Analysis

NOTE: SAR = safety analysis report
SER = safety evaluation report

2.1.3 Evaluation Approach

A systematic and comprehensive top-down approach is taken to evaluate the potential effects of ABD glass on the PCSA and on the radiological engineering components of the license application; this approach is comprised of the following steps:

1. Perform a high-level evaluation to screen the areas that could be affected by a change to HLW glass composition or canister throughput.
2. For the areas that could not be screened out in the first step, develop a listing of the specific technical disciplines that support each specific area. Perform an evaluation with an appropriate level of detail to determine which technical discipline could be affected by a change to HLW glass composition or canister throughput.
3. Provide perspective on anticipated results based on qualitative or semiquantitative evaluations.

The screening steps described above are based on one or a combination of the following:

- Defensible technical rationale that a change in HLW glass composition could not affect a specific issue addressed in the safety analysis
- An assumption related to the composition change that must be verified
- Design changes or operational requirements that could be implemented

The results of the evaluation are summarized in Section 3.2.

2.2 Postclosure Performance

One purpose of the license application for the proposed Yucca Mountain repository was to demonstrate that the repository system barriers would function such that SNF and HLW could be disposed of without exceeding regulatory dose limits (i.e., that they could be disposed of safely) for up to 10,000 years and 1,000,000 years after closure. Quantitative information supporting this demonstration was developed using the TSPA (SNL 2010a), which incorporated other models and abstractions of models. FEPs that were found to be important to repository performance were included in the postclosure PA, while FEPs that were not found to be important to repository performance were not included in the postclosure PA. A short summary description of the TSPA model is given in Section 2.2.1.

As discussed further in Section 3.1.1, the HLW glass canisters were to be disposed of in waste packages that also contained DSNF. In the TSPA model, waste packages were modeled as containing only SNF (either CSNF or naval) or as waste packages containing both HLW glass and DSNF. DSNF was assumed to dissolve instantaneously as soon as the waste package failed, such that radionuclides were immediately available for transport out of the waste package. The degradation of HLW was not instantaneous but followed a rate law. On average, the radionuclide inventory and thermal output of waste packages containing only SNF are higher than those waste packages containing both HLW and DSNF (DOE 2008, Section 1.5).

In conducting the evaluations, a decision with respect to whether a particular component of the license application technical basis would be affected (i.e., change) by implementing the ABD Program fell into one of two categories:

- **Yes**—The component is potentially affected by implementing the ABD Program. For example, the number of waste packages containing DSNF will decrease, compared to the number that was planned for the proposed Yucca Mountain repository, thereby affecting the FEP 2.1.01.01.0A "Waste Inventory." Components in this category could require additional analyses.
- **No**—The component is not affected by implementing the ABD Program or the change in the component clearly fits within the uncertainty and variability envelope already analyzed. For example, the FEP 2.1.03.01.0A "General Corrosion of Waste Packages" is not affected by implementing the ABD Program because there is no change to the waste packages and the proposed compositional changes to the ABD glass are not significant enough to affect general corrosion of the waste package. Another example is the FEP 2.1.02.03.0A "HLW Glass Degradation…" because the anticipated degradation rate of the proposed ABD glass is within the range of degradation rates already considered in the license application.

For a generic repository that is similar to Yucca Mountain, evaluating the effects of implementing the ABD Program on future repository licensing with respect to postclosure performance consisted of (1) identifying design changes that might result from omitting the highly enriched SNF from the inventory to be disposed of and the increased fissile material content of ABD glass to be disposed (Section 2.2.1), and (2) evaluating FEPs analyses for design, modeling, and FEPs inclusion/exclusion decisions that could be affected by implementing the ABD Program (Section 2.2.2). These different components are discussed below.

For a generic repository, the effects of implementing the ABD Program are more challenging to evaluate at this time because much of the information needed to complete a meaningful evaluation is not available. The postclosure performance of a future repository will be considered as a part of a postclosure PA developed for that specific disposal concept. The future PA will include the design of the repository, which will necessarily consider the characteristics of the host rock and the waste projected to be emplaced in that repository. As appropriate, these characteristics will be provided by site-specific data deemed representative of the proposed repository site and design. This PA will consider all the FEPs that could affect the repository and will include the characteristics of all HLW glass projected to be emplaced in the repository, be it the glass produced by the ABD Program, produced at SRS prior to implementing the ABD Program, or produced by other DOE programs. The PA will evaluate the fissile content of the HLW glass as well as how the HLW affects and is affected by the host rock and the design of the waste packages and repository. Thus, FEPs concerning criticality, waste form degradation, and thermal loading will be evaluated as part of the postclosure PA, along with any other FEPs that affect or are affected by the ABD and other HLW glass.

While conceptual ideas for repositories that are not similar to Yucca Mountain have been developed and lists of applicable FEPs have also been developed (Mariner et al. 2011; Mariner et al. 2017; Mariner et al. 2018; Freeze et al. 2011), none contains enough detail to do a PA similar to the PA described in this report for a repository that is similar to Yucca Mountain. However, based on the information available regarding the projected characteristics of the ABD glass, no concerns regarding disposal of that waste in a generic repository have been identified. The remainder of this subsection and the results in Section 3.1 and Section 3.3 pertain only to the technical basis for postclosure performance of a repository similar to that proposed at Yucca Mountain.

2.2.1 Summary Description of the TSPA Model

The TSPA model is a system-level description of a nuclear waste repository. A system-level model, by its very nature, includes a variety of components that represent a wide range of physical and chemical processes. These physical and chemical processes are included in the system-level model through submodels and/or abstraction of results from process-level models. A detailed description of the architecture of the TSPA model, including components, submodels, and abstractions, is provided in SNL (2010a, Volume 1).

The representation of the engineered barrier system (e.g., waste packages, drip shields) and natural barrier system (e.g., host rock) includes models for the thermal-hydrologic processes in the rocks surrounding the emplacement drifts as well as thermal-hydrologic and chemical processes in the drift. Waste package and drip shield degradation models simulate general corrosion of the waste packages and drip shields, stress corrosion cracking of the waste packages, microbially influenced corrosion of the waste package outer surface, and the possibility of localized corrosion of the waste package outer surface. Waste form degradation and mobilization models simulate the degradation of CSNF, DSNF, and HLW glass and the subsequent dissolution of their radionuclide inventories into the liquid phase present in the degraded waste. Flow and transport models calculate the rate of radionuclide release from the engineered barrier system to the unsaturated host rock below the repository, which is determined by seepage into the emplacement drifts, condensation on the drift walls, waste package and drip shield degradation, the presence of water films on in-package internals, waste form degradation, and the thermal-hydrologic environment of the engineered barrier system (DOE 2008, Section 2.4.1.2.1). Implementing the ABD Program would change some aspects of the technical basis for several of the models of the engineered barrier systems used in the TSPA. Differences would be introduced into several of the engineered barrier system models; these differences are discussed below.

A review of the models used by the postclosure PA led to the identification of design parameters that were used as direct inputs in the PA. The review led to the assignment of controls on important design parameters (BSC 2008e). Three of these are important for the evaluation of changes related to ABD glass (DOE 2008, Section 1.3.1.2.5):

- Maximum linear heat load at emplacement over the length of a seven-waste-package segment: 2.0 kW/m
- Nominal spacing between adjacent waste packages, averaged over a seven-waste-package segment: 10 cm
- Calculated thermal energy density of any seven adjacent as-emplaced waste packages: not to result in exceeding a thermal-energy-density index (temperature) of 96°C at the mid-pillar (between adjacent emplacement drifts), adjusted to account for host-rock thermal conductivity and hydrologic conditions in the host rock

TSPA calculates the total annual dose to a human receptor, as defined by regulation, for two timeframes, i.e., 10,000 years and 1,000,000 years. The model calculates possible future outcomes for multiple realizations using distributions of values for uncertain parameters that may be important to the performance of the system. The model realizations are performed using various combinations of parameter values. Each of the combinations of parameter values is representative of a subset of the full range of potential outcomes. The total annual dose is the sum of the annual doses calculated for several scenario classes, including the nominal scenario class, the early failure scenario class, the igneous scenario class, the seismic scenario class, and a human intrusion scenario. Implementing the ABD Program would lead to differences in all of these scenario classes. The details of the differences vary between scenario classes and are discussed further in Section 3.3.2 and Section 3.3.3. The representation of these scenario classes in the TSPA model is described in Volume I of SNL (2010a) and the results of the calculations are presented in Volume 3 of SNL (2010a).

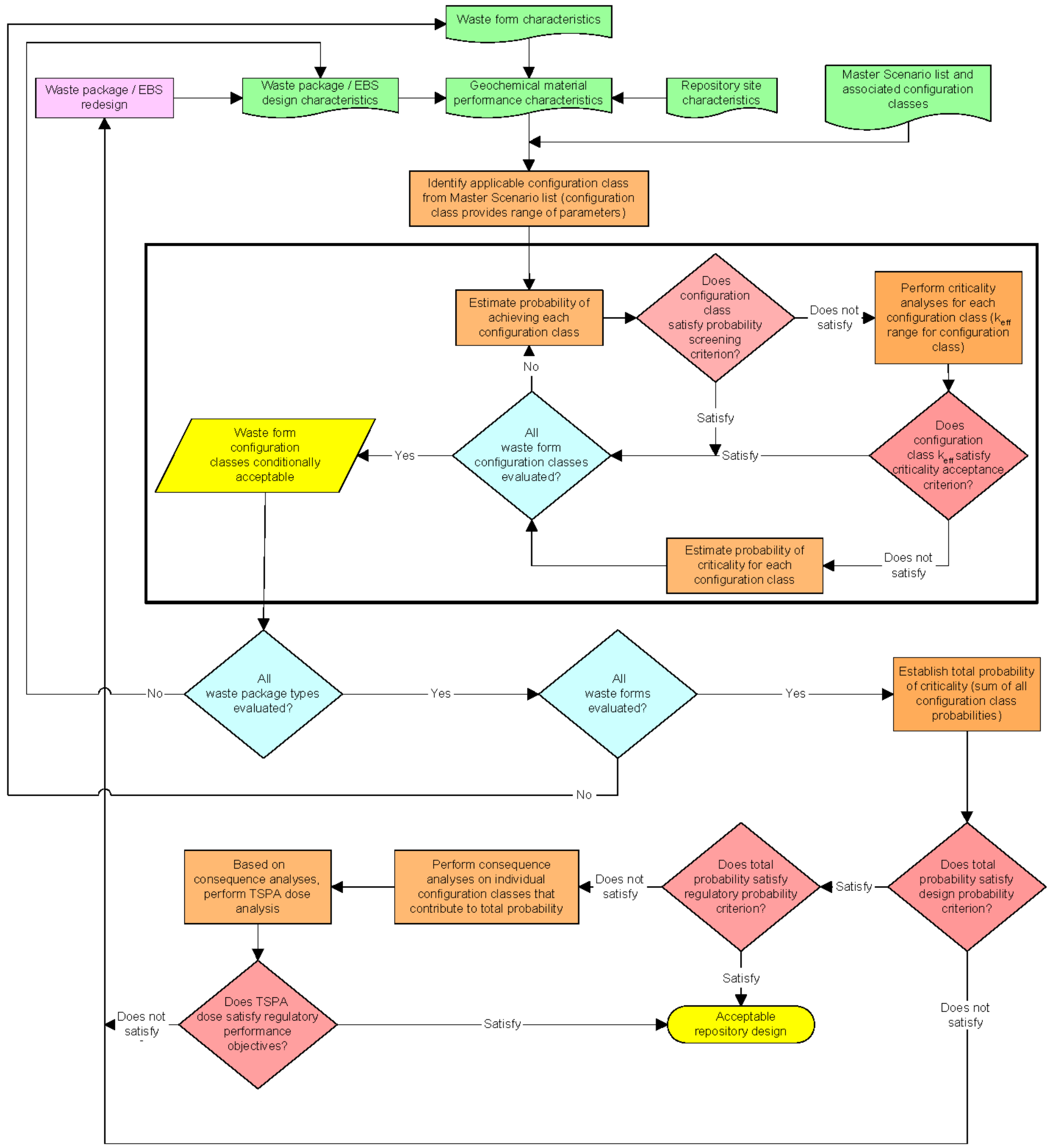
2.2.2 Identify Affected FEPs Justifications and Models

Each of the 374 FEPs presented in *Features, Events, and Processes for the Total System Performance Assessment: Analyses* (SNL 2008a) was evaluated for design, modeling, and inclusion/exclusion justifications that could be affected by implementing the ABD Program. The FEPs report (1) identifies each FEP that is to be included in the postclosure PA and describes how it is included in the PA, and (2) identifies each FEP that is not to be included in the postclosure PA and provides the technical basis for excluding the FEP from the postclosure PA. The approach used in the evaluation of performance-related FEPs depended on the whether the FEP was included or excluded from the PA, as well as the reason for exclusion (for excluded FEPs) and how it was included in the PA (for included FEPs). The approach to evaluating the sixteen criticality FEPs is described first, followed by a description of the approach to evaluating the noncriticality FEPs.

Evaluation of Criticality FEPs—The postclosure criticality FEPs analysis follows the risk-informed, performance-based, disposal criticality analysis methodology as documented in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003) and illustrated in Figure 2-2. Note that the highlights represent the various analysis areas and decision points; the detailed description of the process is provided in Section 3.1 of the *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003). The methodology allows for (1) excluding criticality FEPs from the postclosure PA on the basis of low probability or on the basis of low consequence, or (2) including them in the postclosure PA. However, a decision early in the Yucca Mountain repository design was made to screen postclosure criticality on the basis of low probability. To meet this objective, specific requirements were established on waste form loading and canister designs, including neutron absorbers and basket materials of construction. Specific loading requirements and designs were established for CSNF and DSNF canisters; however, the HLW canister designs and glass waste form loadings were provided as an input from the respective sites.

Sixteen criticality FEPs covering in-package and external configurations under nominal conditions and disruptive events were defined as follows:

- In-package criticality (intact configuration)
- In-package criticality (degraded configurations)
- In-package criticality resulting from rockfall (intact configuration)
- In-package criticality resulting from rockfall (degraded configurations)
- In-package criticality resulting from a seismic event (intact configuration)
- In-package criticality resulting from a seismic event (degraded configurations)
- In-package criticality resulting from igneous event (intact configuration)
- In-package criticality resulting from igneous event (degraded configurations)
- Near-field criticality
- Near-field criticality resulting from rockfall
- Near-field criticality resulting from a seismic event
- Near-field criticality resulting from an igneous event
- Far-field criticality
- Far-field criticality resulting from rockfall
- Far-field criticality resulting from a seismic event
- Far-field criticality resulting from an igneous event



NOTE: EBS = engineered barrier system

TSPA = total system performance assessment

Figure 2-2. Overview of Disposal Criticality Analysis Methodology

Postclosure criticality FEPs are addressed in Section 2.2.1.4.1 of the license application (DOE 2008), which is primarily supported by the criticality FEPs screening analysis documented in *Screening Analysis of Criticality Features, Events, and Processes for License Application* (SNL 2008d). That document covers the postclosure criticality analysis for all waste forms. The combined probability of postclosure criticality is additive based on waste form, scenario, and configuration. Although the postclosure criticality screening analysis is based on low probability, a significant part of the postclosure criticality analysis was deterministic. The only probabilistic components of the postclosure criticality analysis used to evaluate FEPs screening were the probability of neutron absorber and waste form misloads (in combination with the probability of waste package failure). A large number of calculations for potential changes to in-package configurations and releases of fissile material formed the deterministic part of the analysis. These deterministic analyses supported the argument that if the canisters were designed as specified and loaded as specified, in-package postclosure criticality would not be “credible” for the range of potential configurations and compositions associated with potential in-package chemistry and degradation scenarios. Likewise, release and accumulation of sufficient fissile materials in the near field and far field to support an external criticality would also not be “credible.” Note that the external criticality analysis was not performed explicitly for every DSNF type; rather, an argument based on hypothetical conservative analyses and the required processes to result in a critical external configuration was developed (DOE 2008, Sections 2.2.1.4.1.3.3 and 2.2.1.4.1.3.4).

Because HLW glass is to be disposed of in waste packages containing both HLW and DSNF (Section 3.1.1), the in-package and external criticality potential for these waste packages was based on the combined inventory of HLW glass canisters and DSNF canisters. The postclosure criticality analysis was based on nine representative DSNF types. These representative SNF types were conservatively selected based on fissile material enrichment, mass, and geometry. Acceptance of other SNF types is contingent on demonstrating that they are bounded by the analyzed types or the performance of fuel-specific analyses.

Table 2-2 lists the eight DSNF types evaluated in this report, omitting N Reactor SNF, which is to be disposed of in multicanister overpack (MCO) canisters, because the proposed ABD HLW glass would not be codisposed with N Reactor SNF in a 2-MCO/2-DHLW waste package.

Table 2-2. Representative DSNF Fuel Groups for Criticality Analyses

Fuel Group	Representative Fuel Type
Mixed Oxide	FFTF Driver Fuel
UZrH	TRIGA Fuel
Mo and U-Zr Alloys	Enrico Fermi Fast Reactor Fuel
Highly Enriched Uranium Oxide	Shippingport PWR Fuel
^{233}U/Th Oxide	Shippingport LWBR Seed Assembly Fuel
Highly Enriched Uranium-Al	ATR Fuel
U/Th Carbide	Fort St. Vrain Fuel
Low Enriched Uranium Oxide	TMI-2 Fuel

NOTE: ATR = Advanced Test Reactor
FFTF = Fast Flux Fast Reactor
LWBR = Light Water Breeder Reactor
PWR = pressurized water reactor
TMI-2 = Three Mile Island Unit 2
TRIGA = Training, Research, Isotope, General Atomics
UZrH = uranium-zirconium hydride

Unlike the criticality analysis for waste packages containing only CSNF, which relies on a single bounding analysis basis configuration for in-package criticality and a low-enrichment, low-concentration argument for postclosure criticality, the postclosure criticality analysis for waste packages containing both DSNF and HLW is DSNF-type dependent. The process generally included the following steps:

1. A proposed basket design and SNF loading were developed.
2. Intact criticality analyses were performed.
3. In-package geochemistry degradation models and analyses provided in-package chemical compositions for various stages of degradation. These analyses considered wide ranges of degradation rates and order of degradation to provide in-package chemical compositions. The analyses included conservative assumptions for retention of fissile material compounds, hydration, and removal of absorber elements. The primary analyzed stages of degradation were the following:
    a. HLW glass degrades before DSNF
    b. DSNF degrades before HLW glass
    c. DSNF and HLW degrade concurrently
4. Detailed degraded criticality calculations covered the potential configurations and compositions identified in the degradation analyses.
5. If subcriticality is maintained for the wide array of analyzed configurations, then subcriticality during disposal could be concluded; if not, feedback was provided to modify the design and/or loading.
6. Basket design, fuel loading, or addition of absorbers in the form of fixed neutron absorbers or shot (for moderator displacement and absorption) was considered.

7. Material release and accumulation models and analyses were performed based on conservative solubility assumptions that maximized release of fissile material.
8. Parametric criticality calculations for near-field and far-field configurations were performed to determine the required fissile material mass and concentration that could result in a critical configuration.

A systematic and comprehensive approach was taken to evaluate the potential effects of the proposed ABD glass on in-package and external postclosure criticality FEPs; this approach is comprised of the following steps:

1. Summarize the in-package criticality bases for each DSNF type that could be codisposed with ABD glass based on the intact and degraded criticality calculations and supporting geochemistry degradation and release models and analyses.
2. Evaluate the potential effect of the compositional change of the ABD glass on these calculations and analyses and provide perspective on anticipated results.
3. Determine whether there are potential effects on the canister design requirements or loading criteria.
4. Evaluate the potential effects of ABD glass on external postclosure criticality FEPs in a manner consistent with the bounding analyses relied upon to screen out the potential for near-field and far-field criticality for all waste forms.
5. Evaluate the potential effect on the in-package and external criticality FEPs probabilistic screening.

Evaluation of Noncriticality FEPs—The approach to evaluating noncriticality FEPs depended on whether the FEP was excluded by regulation, was excluded by low probability or low consequence, or was included as follows:

- **FEPs Excluded by Regulation**—Categorize these FEPs as having no effect, "N." Regulations are not affected by implementing the ABD Program; therefore, these FEPs are categorized as having no effect.
- **FEPs Excluded by Low Probability or Low Consequence**—Determine whether the justification for the decision to exclude the FEP from the PA is based on anything that might change because of implementing the ABD Program. The screening justifications for FEPs excluded by low probability may rely on specific results of analyses to show the probability of occurrence is below the regulatory threshold for inclusion in the PA. Similarly, the screening justifications claiming low consequence often rely on specific results to show that excluding the FEP would have no significant effect on meeting the various postclosure performance standards. Therefore, the low probability and low consequence analyses and justifications were examined to determine whether implementing the ABD Program could result in differences in the technical basis for these analyses and justifications.
- **Included FEPs**—Determine if the way a particular FEP is included in the postclosure PA might change because of implementing the ABD Program. The screening justifications for included FEPs differ from excluded FEPs in that they explain how the FEP is included in the postclosure PA. Evaluating included FEPs required looking at the model or models in which the FEPs of interest are included to determine whether implementing the ABD Program could change the

technical basis for the model or models. If implementing the ABD Program could potentially change the technical basis for the model or models associated with that FEP, the effects of implementing the ABD Program on the technical basis was further investigated. FEPs that are affected only because of potential changes to inputs handed off from an upstream model are marked "N." For example, a change to the inventory source term because of implementing the ABD Program results in a "Y" being assigned to the Waste Inventory FEP (2.1.01.01.0A). However, marking a "Y" to all FEPs related to models that are downstream from the inventory model in the postclosure PA would mean propagating the potential effect through to the final biosphere FEPs without adding value. Another consideration for included FEPs is that the method of inclusion for some FEPs is provided by regulation. An example is the methodology used to demonstrate compliance with the human intrusion standard. Obviously, any parts of methodology given by regulation would not be affected by ABD glass. For these types of FEPs, the evaluation for potential effects focuses on any parts of methodology not set by regulation and/or on data gathered.

The results of the analysis of the effects of implementing the ABD Program on models and all FEPs are presented in Section 3.1 and Section 3.3.

3. RESULTS OF EVALUATIONS

Results of the evaluations conducted as described in Section 2 are discussed below. Section 3.1 presents the results with respect to the subsurface facility (DOE 2008, Section 1.3), Section 3.2 presents the results with respect to the preclosure safety (multiple parts of DOE 2008, Section 1), and Section 3.3 presents the results with respect to the postclosure PA (DOE 2008, Section 2).

3.1 Subsurface Facility

For a repository that is similar to Yucca Mountain, the primary effect of implementing the ABD Program on future repository licensing is disposal of a reduced number of CDSP waste packages, thereby affecting subsurface repository design and the postclosure PA, as discussed below and in Section 3.3.

3.1.1 Change in Number of Waste Packages Disposed

As presented in the Yucca Mountain license application (DOE 2008), DSNF and HLW canisters for commercial HLW and DHLW were to be disposed of in CDSP waste packages. These CDSP waste packages took one of three forms (DOE 2008, Section 1.5.2.1.3):

- The 2-MCO/2-DHLW waste package, consisting of two MCOs containing DSNF (25 in. diameter) and two canisters of HLW (24 in. diameter)
- The 5-DHLW/DOE Short waste package, consisting of five canisters of HLW (each 24 in. diameter) surrounding one canister of DSNF (18 in. diameter), 10 ft long
- The 5-DHLW/DOE Long waste package, consisting of five canisters of HLW (each 24 in. diameter) surrounding one canister of DSNF (18 in. diameter), 15 ft long

The license application also allowed the 5-DHLW/DOE Short and 5-DHLW/DOE Long waste packages to be loaded with a DSNF canister in a peripheral position, instead of a HLW canister, if the center location was empty, or for the waste package to be loaded with five HLW canisters in peripheral positions with the center position empty (DOE 2008, Section 1.5.2.1.3).

There was a relatively small uncertainty in the total number of each type of waste package to be disposed of. Table 3-1 gives the count of each waste package configuration planned for the proposed Yucca Mountain repository as estimated for seismic calculations, while Table 3-2 gives the count of each waste package configuration planned for disposal in Yucca Mountain as modeled in the TSPA.

Table 3-1. Waste Package Design Basis Inventory for Seismic Calculations

Waste Package Configuration	Nominal Quantity
TAD-Bearing	7,483
Naval Long	310
Naval Short	90
5-DHLW/DOE Short	1,207
5-DHLW/DOE Long	1,862
2-MCO/2-DHLW	210
Total	11,162

NOTE: DHLW = DOE high-level radioactive waste
MCO = multicanister overpack
TAD = transportation, aging, and disposal

Source: DOE 2008, Table 2.3.4-27.

Given the MTHM allocation limits on the quantity of DHLW that could be disposed of (Section 1.4), DHLW cannot “replace” the DSNF that had been slated for disposal but is instead to be reprocessed under the ABD Program. As stated in the Yucca Mountain license application, no more than 4,667 MTHM of DHLW was to be disposed of in the repository (DOE 2008, Table 1.5.1-1). This number is equivalent to 9,334 canisters of DHLW. Since each of the ~200 2-MCO/2-DHLW waste packages contains 2 DHLW canisters (Table 3-1 and Table 3-2), only ~1,800 of the 5-DHLW/DOE (Short and Long) waste packages are required to dispose of the remaining canisters from the DHLW allocation of 9,334 canisters (assumes fully loaded 5-DHLW/DOE waste packages). This estimate does not account for the ~275 canisters of West Valley HLW planned for disposal in the 5-DHLW/DOE Short waste package (SNL 2008, Section 1.5.2.1.3) because, as commercial HLW, the West Valley HLW is not included in the DHLW allocation. Again, assuming fully loaded waste packages, ~55 5-DHLW/DOE Short waste packages would be required to dispose of the West Valley HLW canisters, bringing the total number of 5-DHLW/DOE (Short and Long) waste packages required for commercial HLW and DHLW canister disposal to ~1,855. This estimate is less than the ~3,000 5-DHLW/DOE (Short and Long) waste packages that were planned for disposal (Table 3-1 and Table 3-2), meaning that some of the CDSP waste packages were not going to contain the full complement of HLW canisters. It should be noted, however, that the TSPA model assumed that all CDSP waste packages were fully loaded with waste, thereby conservatively overestimating the quantity of HLW to be emplaced in the repository (SNL 2010a). In any case, the number of CDSP waste packages considered for disposal is driven by the number of DSNF canisters, not the number of HLW canisters.

Table 3-2. Waste Package Configurations and Count of Each Configuration as Modeled in the TSPA

Waste Package Configuration	Waste Package Types	Shorthand Designator	Count in Repository Footprint (Inventory)
21-PWR/44 BWR TAD Canister Waste Package	21 PWR assemblies in TAD canister	21-PWR TAD	4,586
	44 BWR assemblies in TAD canister	44-BWR TAD	3,037
	12 PWR assemblies in long TAD canister	12-PWR-long TAD	173
Naval Long	Naval SNF in long canister	Naval SNF-long	323
Naval Short	Naval SNF in short canister	Naval SNF-short	94
		Total CSNF[a] and Naval SNF Waste Packages	8,213
5-DHLW/DOE Long	1 DSNF and 5 HLW long canisters	CDSP-long	1,940
5-DHLW/DOE Short	1 DSNF and 5 HLW short canisters	CDSP-short	1,257
2-MCO/2-DHLW	2 DSNF MCO and 2 HLW canisters	CDSP-MCO	219
		Total CDSP Waste Packages	3,416
		Total Waste Packages	11,629

NOTE: [a] PWR and BWR SNF are disposed of in CSNF waste packages.
BWR = boiling water reactor
CDSP = codisposal
CSNF = commercial spent nuclear fuel
DHLW = DOE high-level radioactive waste
DSNF = DOE spent nuclear fuel
HLW = high-level radioactive waste
MCO = multicanister overpack
PWR = pressurized water reactor
SNF = spent nuclear fuel
TAD = transportation, aging, and disposal

Source: SNL 2010a, Table 6.3.7-1.

Implementing the ABD Program will reduce the quantity of DSNF requiring disposal by processing that DSNF into HLW glass, thereby reducing the number of DSNF canisters needing disposal and increasing the number of HLW glass canisters needing disposal, compared to what was planned for the Yucca Mountain repository. Under the ABD Program, the waste to be reprocessed that was originally destined for disposal as DSNF at Yucca Mountain consists of 3,500 Materials Test Reactor (MTR) bundles, 200 High-Flux Isotope Reactor (HFIR) cores, Japanese Fast Critical Assembly waste, and all non-aluminum SNF currently stored in L-basin at SRS or projected to be received by 2033 (DOE 2019). This waste is equivalent to about 30 MTHM of SNF (DOE 2019). SRS expects to receive aluminum-clad SNF from domestic research reactors and foreign research reactors (e.g., HFIR cores and MTRs) until 2033 (DOE 2019). These wastes will be reprocessed into HLW glass as part of the ABD Program.

Note that implementing the ABD Program will leave about 75 HFIR cores at Oak Ridge National Laboratory (i.e., they will not be reprocessed as part of the ABD Program). Each HFIR core consists of an inner core and an outer core; they were to be separated and disposed of with three inner cores stacked in an 18 in. diameter canister and three outer cores stacked in a 24 in. diameter canister. Thus, it will take about 50 canisters to dispose of the 75 HFIR cores that will not be reprocessed into HLW glass as part of the ABD Program.

A query of the Spent Fuel Database (Version 7.0.6, 2020) provided information regarding the MTHM and number of DSNF disposal canisters associated with each of these wastes, as shown in Table 3-3. In this table, waste that is to be reprocessed as part of the ABD Program and not disposed of as SNF is indicated in ***bold italics***.

Table 3-3. Waste Quantities

Waste	Canisters	MTHM
Total Current SNF Inventory	3061	2446.878
Current MCO Inventory	412	2115.986
Current SNF Inventory at SRS	***727***	***27***
Projected ATR SNF	68	1.196
Projected HFIR SNF (to be reprocessed)	***30***	***0.36***
Projected HFIR SNF	50	0.60
Projected University SNF	54	0.661
Projected Foreign Research Reactor SNF	***269***	***2.713***

NOTE: ***Bold italics*** is used to identify waste to be reprocessed as part of the ABD program and not disposed of as SNF.
ATR = Advanced Test Reactor
HFIR = High-Flux Isotope Reactor
MCO = multicanister overpack
SNF = spent nuclear fuel
SRS = Savannah River Site

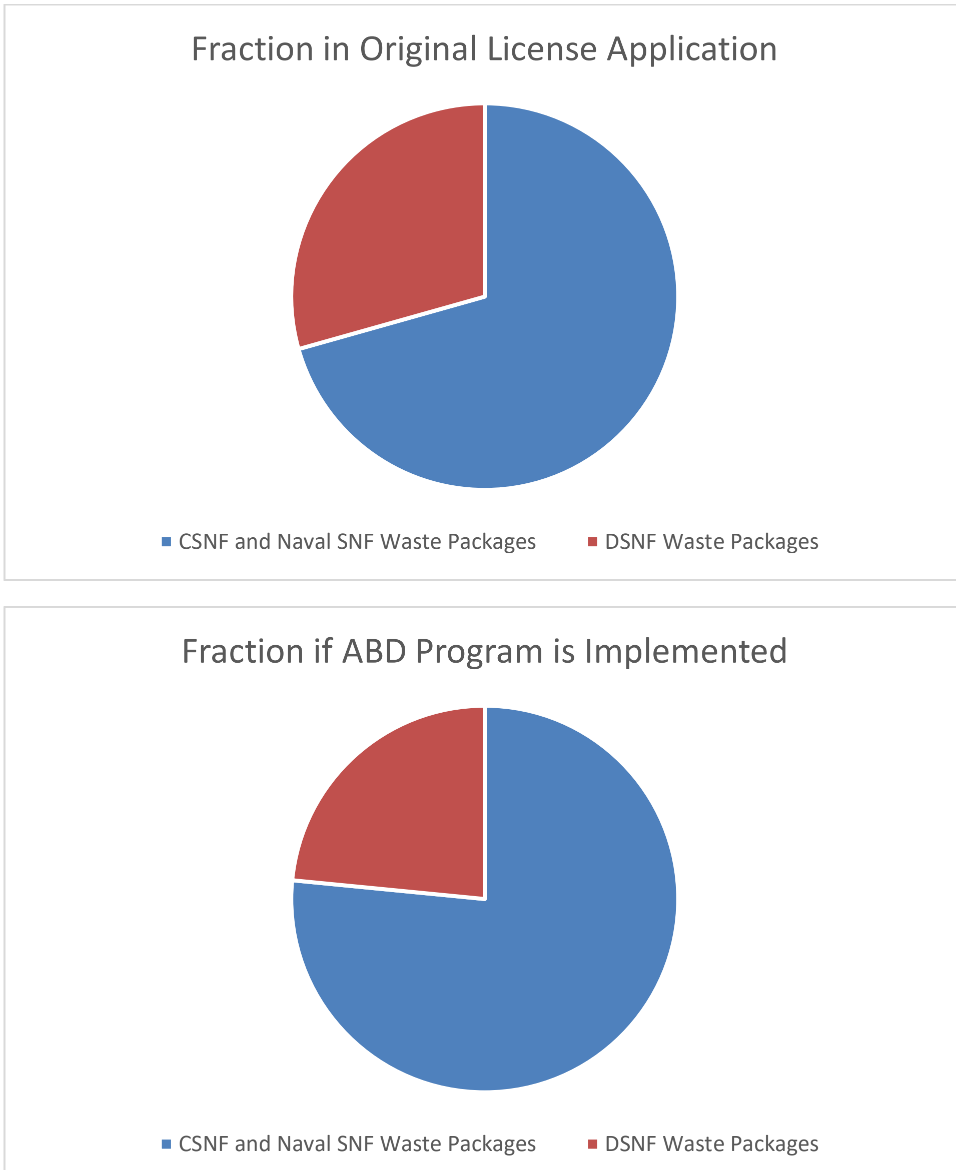
Of the wastes shown in Table 3-3, the "Current SNF Inventory at SRS," some of the "Projected HFIR SNF," and the "Projected Foreign Research Reactor SNF" are slated to be reprocessed into HLW glass as part of the ABD Program and are not going to be disposed of as SNF. Note that because not all HFIR cores are to be disposed of, HFIR is divided into two fractions: 50 canisters to be disposed of as SNF and the equivalent of 30 canisters that will be reprocessed and not disposed of as SNF. Based on the information in Table 3-3, the waste slated to become ABD glass would have originally been disposed of as SNF in 1,026 canisters (727 + 30 + 269) if there was no reprocessing as a part of the ABD Program. These 1,026 canisters represent about 30 MTHM of waste (27 + 0.36 + 2.713).

However, the current total inventory of SNF (2,447 MTHM) exceeds the limit of 2,268 MTHM established by the DOE for the proposed Yucca Mountain repository. Reprocessing, rather than disposing of, the approximately 30 MTHM of SNF indicated in Table 3-3 allows 30 MTHM of different types of SNF to be disposed of in its place. How many canisters this incremental amount would represent depends on the type of SNF to be disposed of in place of the SNF that is to be reprocessed into ABD glass. If the "replacement" SNF is in MCOs (primarily SNF from the N-Reactor), the amount would constitute about 6 MCO canisters, based on an average of about 5 MTHM of SNF per MCO (from Table 3-3). If the "replacement" SNF is not in MCOs, the amount would constitute many more canisters of waste. The existing and projected non-MCO SNF inventory to be disposed of contains an average of 0.118 MTHM per canister ((2446.878 − 2115.986 + 1.196 + 0.60 + 0.661) / (3061 − 412 + 68 + 50 + 54). Disposing of 30 MTHM of this SNF in lieu of the SNF to be reprocessed under the ABD Program would constitute about 254 canisters of DSNF (30 MTHM ÷ 0.118 MTHM per canister).

Implementing the ABD Program will result in reprocessing 1,026 canisters of DSNF which had been slated for disposal; these 1,026 canisters of DSNF correspond to 1,026 fewer CDSP waste packages. If the SNF that "replaces" the SNF reprocessed under the ABD Program is in MCOs, there would be about 1,023 fewer CDSP waste packages (1,026 – 6/2) slated for disposal. If the SNF that "replaces" the SNF reprocessed under the ABD Program is not in MCOs, there would be about 772 fewer CDSP waste packages (1,026 − 254). Taking the average of 1,023 and 772 yields 898, or about 900 fewer CDSP waste packages to be disposed of than originally planned.

Regardless of whether the information in Table 3-1 is used or the information in Table 3-2 is used, the removal of approximately 900 CDSP waste packages from the repository results in a ~26% reduction in the number of CDSP waste packages to be disposed of and a ~8% reduction in the total number of waste packages to be disposed. The removal of approximately 900 CDSP waste packages also reduces the ratio of the number of CDSP waste packages to the number of CSNF and naval SNF waste packages from ~0.4 to ~0.3. This difference is shown in Figure 3-1. Note that in calculating this ratio, naval SNF waste packages, which are part of the DOE waste allotment for inventory purposes (Section 1.4), are grouped with CSNF. This grouping is driven by the fact that naval SNF waste packages, which are similar to CSNF waste packages in terms of contents, thermal output, etc., are modeled as if they were CSNF waste packages in the postclosure PA (SNL 2010a). This reduction in the number of waste packages could affect subsurface repository design, as discussed in Section 3.1.2. The reduction in the ratio of cooler CDSP waste packages to hotter SNF waste packages might also require waste packages to be spaced farther apart to meet thermal requirements, or for the ventilation period to be increased, etc. These changes could also affect multiple models, as discussed in Section 3.3.

It should also be noted that ~21,000 canisters of DHLW were expected to be produced by SRS, Hanford, and Idaho (DOE 2008, Section 1.5.1.2.1.2). This amount is ~12,000 more canisters of DHLW than the allocation limit considered for Yucca Mountain. Implementing the ABD Program will produce approximately 521 additional canisters of DHLW glass (SRNS 2020), thereby increasing the number of DHLW canisters above the allocation limit.



NOTE: ABD = Accelerated Basin Deinventory
CSNF = commercial spent nuclear fuel
DSNF = DOE spent nuclear fuel
SNF = spent nuclear fuel

Figure 3-1. Change in Radionuclide Inventory If ABD Program Is Implemented

3.1.2 Effects on Subsurface Design

Changing the ratio of the number of CDSP waste packages to the number of CSNF and naval SNF waste packages from ~0.4 to ~0.3 would affect subsurface design with respect to meeting thermal requirements because there would be fewer cooler CDSP waste packages available to place between the hotter CSNF waste packages for the purpose of keeping repository temperatures within design constraints. While repository temperatures could still be kept within design constraints, the subsurface design might have to be altered to accomplish this goal. For example, the original design called for waste packages to be placed 10 cm apart; this spacing might have to be increased to keep temperatures within design constraints. If waste package spacing remains the same, reducing the total number of waste packages to be disposed of by ~8% would affect the subsurface design in that fewer drifts would need to be excavated.

The subsurface design is included in some TSPA models and modeling cases. Changes to the design could result in different models and modeling cases. To facilitate a discussion of how these models and modeling cases could be changed, three different examples of possible changes to the subsurface design are presented below and summarized in Table 3-4. It should be noted that these are just examples and do not represent an exhaustive analysis of possible repository designs. Design Examples 1 and 2 assume about 900 fewer CDSP waste packages as discussed in Section 3.1.1. For comparison purposes, Design Example 3 assumes the number of waste packages remains the same. Besides the total number of CDSP waste packages, the design examples discussed below vary with respect to the waste package spacing (i.e., distance between waste packages, originally set at 10 cm) (OE 2008, Section 1.3.1.2.5), drift spacing (i.e., distance between drifts, centerline to centerline, originally set at 81 m) (DOE 2008, Section 1.3.2.4.3.5), and number of drifts (originally set at 91) (DOE 2008, Table 1.3.1-3). These three examples are also the basis for the thermal analyses discussed in Appendix B-2.

Design Example 1 assumes the waste package spacing and drift spacing do not change; however, to accommodate the decreased number of waste packages needing disposal, the number of drifts is decreased by about 9 from 91 to 82 (assuming the same number of waste packages per drift, i.e., ~100). Reducing the number of drifts would result in a slightly higher linear thermal power than that used in the license application (see Appendix B-2.1 for further discussion of the line-averaged thermal line load), which would in turn affect both the multiscale thermal-hydrologic model (Section 3.3.2.1) (SNL 2008b) and the seepage model (Section 3.3.2.2) (BSC 2004h). Both models use the thermal line load (i.e., linear thermal power), so use of a thermal line load slightly higher than what was used in those models represents a change in the technical bases for these models.

In Design Example 2, waste packages are spaced farther apart to accommodate the reduction in waste packages while the drift spacing, drift length, and number of drifts are kept the same. In this case, the thermal line load would be slightly lower than that used in the license application. Similar to Design Example 1, the multiscale thermal-hydrologic model (Section 3.3.2.1) (SNL 2008b) and the seepage model (Section 3.3.2.2) (BSC 2004h) would be affected because of the change in thermal line load. Design Example 2 would also affect the seismic model (Section 3.3.3) because the waste packages would be farther apart than the 10 cm assumed in the seismic consequence calculations (SNL 2007k).

In Design Example 3, the number of CDSP waste packages remains the same rather than being reduced as is assumed for Design Examples 1 and 2. This approach means the waste package spacing, drift spacing, and number of drifts do not change; however, many CDSP waste packages would have to be only partially filled. Taking this design approach means that the radionuclide inventories in the CDSP waste packages would be different from what was modeled in the license application (DOE 2008), the thermal line load would be slightly lower than what was used in the license application, and the masses of waste inside the CDSP waste packages would be different from what was used in the seismic consequence model (Section 3.3.3) (SNL 2007k). This situation could possibly affect the in-package chemistry model (Section 3.3.2.5) (SNL 2007i) and criticality evaluations (Section 3.3.1), which assume that CDSP waste packages contain both DSNF and HLW. The slightly lower thermal line load could also affect the multiscale thermal-hydrologic model (Section 3.3.2.1) (SNL 2008b) and the seepage model (Section 3.3.2.2) (BSC 2004h) in that the thermal line load would be slightly lower than what was used in those models.

Table 3-4. Summary of Examples of Possible Design Changes Resulting from Reduced DSNF

Design Example	Number of CDSP Waste Packages	Waste Package Spacing	Drift Spacing	Number of Drifts	Potential Effects on Models and Analyses
1	Decreased by ~900	Same	Same	Decrease	Higher thermal line load Seismic consequence unchanged Multiscale thermal-hydrologic model affected (Section 3.3.2.1) Seepage models affected (Section 3.3.2.2)
2	Decreased by ~900	Increase	Same	Same	Lower thermal line load Seismic consequence calculations would be different (Section 3.3.3) Multiscale thermal-hydrologic model affected (Section 3.3.2.1) Seepage models affected (Section 3.3.2.2)
3	Same	Same	Same	Same	Disposal of partially filled CDSP packages would be involved Calculation of average radionuclide inventory of a CDSP package would change Lower thermal line load Seismic consequence would be different (different mass in some waste packages) (Section 3.3.3) Multiscale thermal-hydrologic model affected (Section 3.3.2.1) In-package chemistry model affected (Section 3.3.2.5) Criticality evaluations affected (Section 3.3.1)

3.2 Preclosure Safety Evaluation Results

Applying the multistep approach described in Section 2.1.3, Table 3-5 identifies those areas that could be affected by HLW glass composition change and those that could not (Step 1); the shaded areas are those that could not be affected.

For the areas not screened out in Table 3-5, a listing of the technical disciplines supporting each specific area is provided in Table 3-6 (Step 2). For each specific discipline an evaluation is provided to determine which technical discipline could be affected by changes to HLW glass composition; the shaded disciplines are those that could not be affected by the implementing the ABD Program.

The following subsections provide a summary of the evaluation to determine the technical disciplines that could be affected by changes to HLW glass composition or canister throughput.

Table 3-5. Screening of Preclosure Safety Areas

SAR Section(s)	SER Section(s)	Screening of Preclosure Safety Area
1.1	2.1.1.1	Although site description is an input to preclosure safety and forms the basis for several key analyses including identification of initiating events, it is not a preclosure safety area and could not be affected by preclosure safety results even if those results are changed by ABD glass composition. Therefore, this area is screened out from further evaluation.
1.2, 1.3, 1.4, 1.5	2.1.1.2, 2.1.1.7	Design of surface facilities, subsurface facilities, waste package and waste form is an iterative process with preclosure safety, and therefore could be affected by changes to HLW glass composition.
1.6	2.1.1.1, 2.1.1.2, 2.1.1.3	Identification of initiating events and development of event trees could not be affected by changes to HLW glass composition. Changes to glass composition could not alter external events and natural phenomena events or their frequencies. Because changes in glass composition do not affect canister physical characteristics (assumption in Section 1.4, second bullet), these changes could not result in new initiating events or in unique changes to event tress.
1.7	2.1.1.4	Reliability and event sequence categorization analyses could be affected by changes to HLW glass composition.
1.8	2.1.1.5	Consequence analyses could be affected by changes to HLW glass composition.
1.9 1.13	2.1.1.6 2.1.1.7	Nuclear safety design bases, procedural safety controls, and Q-List could be affected by HLW glass composition if the PCSA analysis results in new or modified important-to-safety structures, systems, and components or procedural safety controls.
1.10	2.1.1.8	Radiological engineering could be affected by changes to glass composition.
1.11	2.1.2	Due to the diversity of waste forms, retrieval plans could not be affected by changes to HLW glass composition. Therefore, this area is screened out from further evaluation.
1.12	2.1.3	Changes to HLW glass composition could not affect plans for permanent closure, decontamination, and dismantlement of surface facilities. Therefore, this area is screened out from further evaluation.
1.14	2.1.1.2, 2.1.1.3, 2.1.1.4, 2.1.1.6, 2.1.1.7	Criticality safety could be affected by changes to HLW glass composition.

NOTE: Shading indicates preclosure safety areas that could not be affected if HLW glass composition were to change because of implementing the ABD Program.

ABD = Accelerated Basin Deinventory

HLW = high-level radioactive waste

ITS = important to safety

PCSA = preclosure safety analysis

SAR = Safety Analysis Report

SER = Safety Evaluation Report

Table 3-6. Evaluation of Technical Disciplines that Could Be Affected by the Proposed ABD HLW Glass

Preclosure Safety Areas	Preclosure Safety Discipline
Design of surface facilities, subsurface facilities, waste package and waste form	Because of the iterative nature of the process, the specific disciplines that could be affected by changes to glass composition in the design of surface facilities, subsurface facilities, waste packages and waste form will be identified based on the conclusions of the PCSA and radiological engineering evaluations.
Reliability and event sequence categorization analyses	Reliability analyses determine the probability of failure of structures, systems, and components in response to initiating events and associated stressors (e.g., mechanical, thermal). Based on the assumption in the second bullet of Section 1.4, changes to HLW glass composition could not affect the response of HLW glass canisters to potential initiating events. Therefore, reliability analyses are screened from further evaluation.
	Hazard categorization analyses determine frequency of event sequences, taking into account initiating events and pivotal events probabilities as well as operational frequency. Changes to glass composition could not affect the probability of initiating events or pivotal events; however, the operational frequency could be affected if the number of canisters that will be received at the repository changes significantly. This potential effect will be evaluated further.
Consequence analysis	Onsite and offsite dispersion modeling could not be affected by changes to HLW glass composition since it is based on meteorological characteristics and site design; it is waste-form independent.
	Inhalation dose evaluations, which include source term development and release fractions, could be affected changes to HLW glass composition.
Identification of safety systems	The specific disciplines that could be affected by changes to nuclear safety design bases, procedural safety controls, and Q-List will be identified based on the conclusions the PCSA evaluation.
Radiological engineering	Subsurface and subsurface facilities shielding design support, which is based on direct exposure analyses, could be affected by changes to HLW glass composition.
	GROA worker dose calculations, which are based on direct exposure analyses and effluent estimates could be affected by changes to HLW glass composition.
Criticality safety	Criticality analyses of normal conditions and credible abnormal conditions, validation, and establishment of parameters important to criticality safety could all be affected by changes to HLW glass composition.

NOTE: Shading indicates preclosure safety disciplines that could not be affected if HLW glass composition were to change because of implementing the ABD Program.

GROA = geologic repository operations area

HLW = high-level radioactive waste

PCSA = preclosure safety analysis

3.2.1 Event Sequences Categorization Analyses

Six event sequence categorization analyses were performed for surface and subsurface facilities (BSC 2009a; BSC 2009b; BSC 2009c; BSC 2009d; BSC 2009f; BSC 2009g). Additionally, a single seismic event sequence categorization analysis was developed for all the facilities (BSC 2009e). Table 3-7 provides a listing of the facilities and whether HLW glass or DSNF canisters are handled in each facility.

Table 3-7. Summary of Facilities and Whether HLW Glass or DSNF Canisters Are Handled in Them

Facility	HLW Glass Canisters	DSNF Canisters
Canister Receipt and Closure Facility	Yes	Yes
Wet Handling Facility	No	No
Initial Handling Facility	Yes	No
Receipt Facility	No	No
Intra-site Operations	Yes	Yes
Subsurface Facility	Combined with all waste packages.	

NOTE: DSNF = DOE spent nuclear fuel
HLW = high-level radioactive waste

The Yucca Mountain PCSA concluded that there were no Category 1 event sequences (DOE 2008, Section 1.7.5). Several Category 2 event sequences associated with HLW canisters were identified. These event sequences involve releases of radionuclides (filtered or unfiltered) or direct exposure due to potential stresses onto HLW canisters in various configurations, including bare canisters, canisters in sealed transportation casks, and canisters in unsealed waste packages. The facilities where such event sequences could occur are the Canister Receipt and Closure Facility, the Initial Handling Facility, intra-site operations, and subsurface. These Category 2 event sequences are summarized in Table 3-8 (DOE 2008, Tables 1.7-7 to 1.7-17).

The assumed HLW and DSNF throughput, upon which event sequence categorization is based, is summarized in Table 3-9 (DOE 2008, Table 1.7-5). CDSP waste packages contain both DSNF and HLW (Section 3.1.1).

Event sequence categorization is based on conservative throughput estimates assuming multiple handling operations as shown in Table 3-9. Event sequence ISO09-TAD-SEQ2-DEL is closest to the Category 1 threshold at a mean frequency of 3E-01. The throughput would have to increase by a factor of 3 for this event sequence to require re-categorization. Reduction of the total number of canisters and waste packages would reduce event sequence frequencies. Based on the mean frequencies provided in Table 3-8, the extent of reduction is not anticipated to affect event sequence categorization (i.e., to change any of them from a Category 2 event sequence to a beyond-Category 2 event sequence). Therefore, it can be concluded that changes affecting canister throughput due to implementing the ABD Program would not affect event sequence categorization.

Table 3-8. Summary of Category 2 Event Sequences That Involve HLW Glass

Facility	Event Sequence Identifier	Event Sequence Description	Mean Frequency	End State
Initial Handling Facility	ESD12B-HWL-SEQ2-DEL	This event sequence represents a direct exposure during assembly and closure of a waste package containing 5 HLW canisters.	4E-02	Unfiltered radionuclide release
	ESD07-HLW-SEQ5-RRU	This event sequence represents a structural challenge to an HLW canister, during canister transfer by the CTM, resulting in an unfiltered radionuclide release. In this sequence the canister fails, the confinement boundary is not relied upon, and moderator is excluded from entering the canister.	7E-03	Unfiltered radionuclide release
	ESD12C-HWL-SEQ3-DEL	This event sequence represents a direct exposure during export of a waste package containing HLW canisters. In this sequence there are no pivotal events.	6E-3	Direct exposure, loss of shielding
	ESD12A-HWL-SEQ2-DEL	This event sequence represents a temporary loss of shielding during CTM operations, while an HLW canister is being transferred. In this sequence there are no pivotal events.	2E-3	Direct exposure, loss of shielding
	ESD13-HLW-SEQ2-DEL	This event sequence represents a thermal challenge to an HLW canister inside a transportation cask, due to a fire, resulting in a direct exposure from loss of shielding. In this sequence the canister remains intact, and the shielding fails.	7E-4	Direct exposure, loss of shielding

Table 3-8. Summary of Category 2 Event Sequences That Involve HLW Glass (continued)

Facility	Event Sequence Identifier	Event Sequence Description	Mean Frequency	End State
Canister Receipt and Closure Facility	ESD19-WP-H&D-SEQ3	This event sequence represents a direct exposure during export of a waste package containing a combination of a standardized canister containing DSNF and HLW canisters. In this event sequence there are no pivotal events.	9E-02	Direct exposure, loss of shielding
	ESD18-HLW-SEQ2	This event sequence represents a temporary loss of shielding during CTM operations, while an HLW canister is being transferred. In this sequence there are no pivotal events.	8E-2	Direct exposure, loss of shielding
	ESD19-WP-H&D-SEQ2	This event sequence represents a direct exposure during assembly and closure of a waste package containing a combination of a standardized canister containing DSNF and HLW canisters. In this event sequence there are no pivotal events.	2E-2	Direct exposure, loss of shielding
	ESD09-HLW-SEQ3-RRF	This event sequence represents a structural challenge to an HLW canister, during canister transfer by a CTM, resulting in a filtered radionuclide release. In this sequence the canister fails, the confinement boundary remains intact, and moderator is excluded from entering the canister.	1E-2	Filtered radionuclide release
	ESD09-HLW-SEQ5-RRU	This event sequence represents a structural challenge to an HLW canister, during canister transfer by a CTM, resulting in an unfiltered radionuclide release. In this sequence the canister fails, the confinement boundary fails, and moderator is excluded from entering the canister.	5E-4	Unfiltered radionuclide release

Table 3-8. Summary of Category 2 Event Sequences That Involve HLW Glass (continued)

Facility	Event Sequence Identifier	Event Sequence Description	Mean Frequency	End State
Canister Receipt and Closure Facility (continued)	CRCF-S-IE-HLW11-05	Seismic failure of CTM breaching HLW canister during processing to waste package	1E-4	Unfiltered radionuclide release
Intra-site Operations	ISO09-TAD-SEQ2-DEL	This event sequence represents a thermal challenge to an HLW canister inside a transportation cask, due to a fire, resulting in a direct exposure from loss of shielding. In this sequence, the canister remains intact, and the shielding fails.	3E-1	Direct exposure, loss of shielding
Subsurface Facility	SSO05-WP-SEQ3-DEL	This event sequence represents a thermal challenge to a canister inside a waste package, due to a fire, resulting in a direct exposure from loss of shielding. In this sequence, the waste package fails, and the canister remains intact.	1E-2	Direct exposure, loss of shielding
	SSO04-WP-SEQ2- DEL	This event sequence represents a direct exposure due to inadvertent TEV door opening or prolonged immobilization of the TEV in the heat causing a loss of shielding. In this sequence there are no pivotal events.	1E-3	Direct exposure, loss of shielding

NOTE: CTM = canister transfer machine
DSNF = DOE spent nuclear fuel
HLW = high-level radioactive waste
TEV = transport and emplacement vehicle

Source: DOE 2008, Tables 1.7-7 to 1.7-17.

Table 3-9. Assumed Throughput (Number of Transfers) in the PCSA

Facility	HLW Canisters	DSNF Canisters	CDSP Waste Packages	Transportation Casks
Canister Receipt and Closure Facility	11,760 for operational initiating events 19,602 for seismic analysis	6,215	6,215	1,860 rail-based (5 HLW canisters each) 385 Transportation casks containing DSNF canisters (up to 9 canisters each)
Initial Handling Facility	1000	—	—	100 rail-based (5 HLW canisters each) 500 truck (1 HLW canister each)
Intra-site Operations	—	—	—	1,960 rail-based (5 HLW canisters each) 500 truck (1 HLW canister each) 385 Transportation casks containing DSNF canisters (up to 9 canisters each)
Subsurface Facility	12,068. All waste package treated as one group			

NOTE: CDSP = codisposal
DSNF = DOE spent nuclear fuel
HLW = high-level radioactive waste

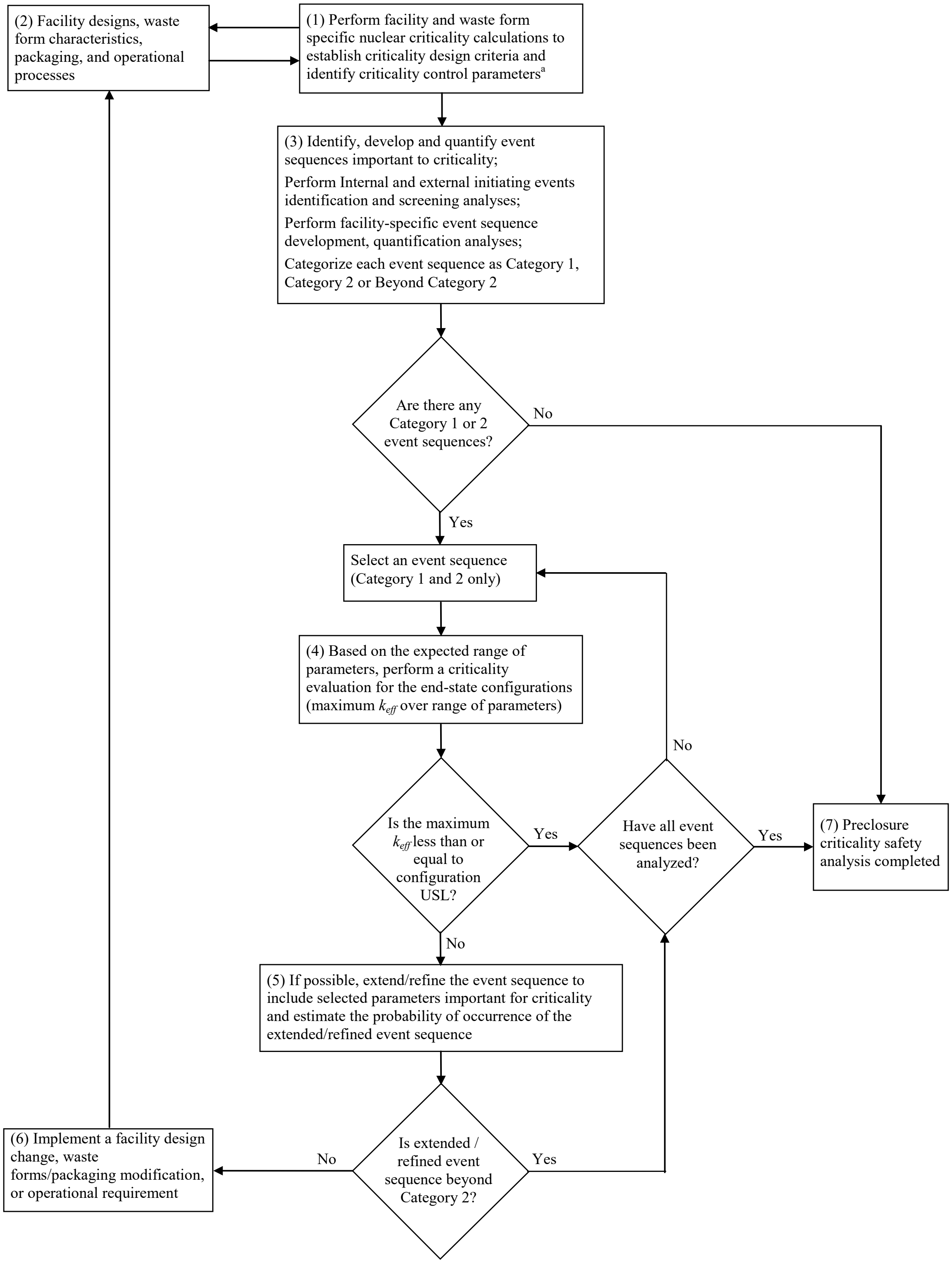
Source: DOE 2008, Table 1.7-5.

3.2.2 Criticality Safety

The preclosure criticality analysis process is depicted in Figure 3-2 (BSC 2008g, Figure 3-1). The surface and subsurface facilities and associated operations were designed to ensure that all event sequences important to criticality were beyond Category 2, thereby meeting the "prevention and control" regulatory requirement. To meet this objective, the criticality safety analysis process started with performing parametric criticality calculation for normal conditions and potential upset conditions that could affect any of the criticality control parameters. Based on the waste forms and preclosure operations, six parameters important to criticality were evaluated: waste form characteristics (covering physiochemical properties, mass, density / concentration, fissile material type, and enrichment), moderation, neutron absorbers, geometry, interaction, and reflection. For each waste form, the criticality analysis covered each waste form type separately or in combination with other waste forms if there is potential for neutronic coupling.

The potential for criticality during preclosure operations is determined based on calculating the effective neutron multiplication factor (k_{eff}) for a range of potential configurations. The configurations are modeled using Monte Carlo N-Particle (MCNP) and include the uncertainty associated with the Monte Carlo analysis at 95% confidence. A configuration is considered subcritical if k_{eff} is less than the upper subcritical limit (USL) for the configuration. The USL is a statistically derived limit that takes into account the bias, bias uncertainty, penalties associated with benchmarking (e.g., range of applicability of benchmark experiments and range of parameters of specific configuration), and an administrative margin to ensure subcriticality. Taking all these parameters into account, a single bounding USL of 0.89 was applied for all DSNF and CDSP configurations for preclosure operations (BSC 2008g, Section 2.3.2). Therefore, a configuration with a k_{eff} value less than the USL is considered subcritical. Note that the real margin of subcriticality is not just associated with the k_{eff}-to-USL comparison, but rather the underlying conservative and often bounding assumptions in the model, including composition, moderation, geometry, reflection, and interaction assumptions. A k_{eff} value close to the USL does not necessarily mean a small margin given the associated context of the configuration assumptions.

The basis for subcriticality of HLW glass as a separate waste form was the limited fissile material concentration as described in the license application (DOE 2008, Section 1.14.2.3.2.4), which was based on the analysis in the *Preclosure Criticality Safety Analysis* (BSC 2008f). The fissile material concentrations in the various HLW glass types are summarized in Table 3-10 (BSC 2008f, Table 3). The minimum subcritical limit for fissile solute from Table 1 of ANSI/ANS-8.1-1998 is 7.3 g/L for $^{239}Pu(NO_3)_4$. Because concentration limits for aqueous solutions are lower than those for other physical/chemical forms, the fact that the concentrations of fissile isotopes in Table 3-10 (DOE 2008, Table 1.14-1) are approximately one order of magnitude less than 7.3 g/L demonstrates that the margin of subcriticality for HLW glass is significant. Therefore, no additional criticality control parameters were identified. The higher fissile concentration of the proposed ABD HLW glass reduces the margin of subcriticality. The maximum fissile concentration of the proposed ABD glass is 2.5 g/L, which is 34% of the ANSI/ANS-8.1-1998 single parameter limit of 7.3 g/L. Taking into consideration the amount of boron in the borosilicate glass matrix and the additional gadolinium, which are both significant neutron absorbers, the basis for the ABD glass subcriticality for preclosure operations involving only HLW glass would remain valid.



NOTE: [a] May include evaluation against single- and multiparameter limits.
USL = upper subcritical limit

Figure 3-2. Preclosure Criticality Safety Analysis Process

Table 3-10. Fissile Isotopes in HLW Glass Canisters

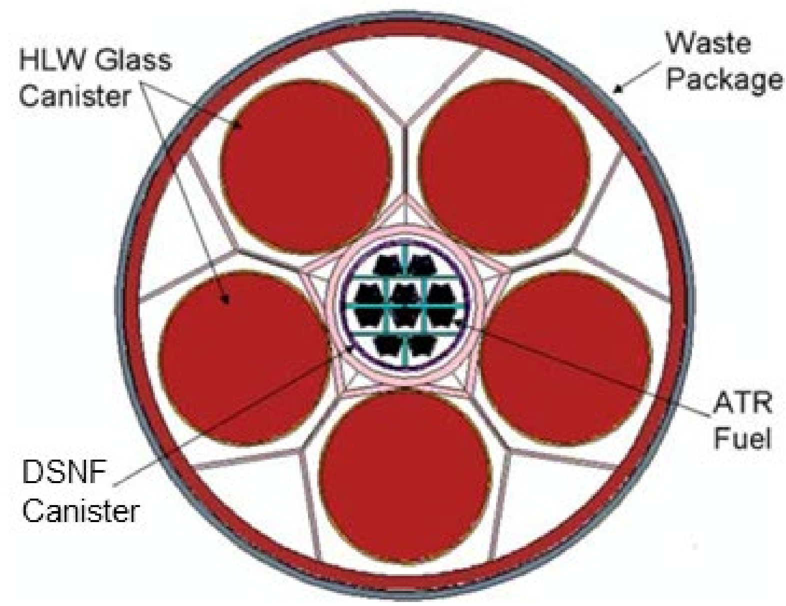
Fissile Isotope	Hanford Canister	Idaho National Laboratory Canister	SRS Canister	West Valley Demonstration Project Canister
^{233}U Mass (g per canister)	0.217	6.29×10^{-4}	5.80	9.37
^{235}U Mass (g per canister)	257	304	307	172
^{239}Pu Mass (g per canister)	343	32.4	280	141
^{241}Pu Mass (g per canister)	1.18	0.208	8.16	3.01
Total Fissile Isotope Mass (g per canister)	601	337	601	325
Nominal Glass Volume (L)	1,080	625	670	665
Fissile Isotope Concentration (g/L)	0.557	0.539	0.897	0.489

NOTE: SRS = Savannah River Site

Source: BSC 2008c, Table 3.

HLW glass operations in the Canister Receipt and Closure Facility and placement in CDSP waste packages result in neutronic coupling between DSNF canisters and HLW glass canisters. The interaction could occur during handling operations by the canister transfer machine (CTM), staging in the staging racks, and placement into CDSP waste packages. Note that HLW glass handled in the Initial Handling Facility is placed in waste packages without DSNF canisters. Additionally, the operations in the Initial Handling Facility do not allow for neutronic coupling between HLW canisters and naval SNF canisters. The parametric calculations pertinent to HLW glass canisters neutronic coupling with other waste forms are documented in *Nuclear Criticality Calculations for Canister-Based Facilities – DOE SNF* (BSC 2008c) and *Nuclear Criticality Calculations for Canister-Based Facilities – HLW Glass* (BSC 2008d). These documents include over 40,000 parametric calculations for various conditions covering perturbations to the various criticality control parameters for the representative DSNF types listed in Table 2-2. Three Mile Island (TMI) SNF, which is of relatively low enrichment, is evaluated in a separate calculation (BSC 2005a), and N Reactor SNF was not evaluated for preclosure safety because additional analyses to demonstrate reliability of the MCOs are required prior to receipt.

HLW glass canisters were modeled explicitly in some parametric calculations as illustrated in Figure 3-3 for the ATR SNF canister; other parametric calculations evaluated its potential reflection properties relative to other hypothetical materials and thicknesses as shown in Table 3-11 (BSC 2008c). Table 3-11 shows that the analyzed HLW glass was not the limiting reflector compared to other materials with similar thicknesses; however, it provided better reflection properties than water. The increase in fissile material concentration from 0.897 g/L up to 2.5 g/L is expected to result in stronger neutronic coupling between HLW canisters and DSNF canisters. The effects of stronger neutronic coupling between HLW glass and DSNF canisters was explored in *Nuclear Criticality Calculations for Canister-Based Facilities – HLW Glass* (BSC 2008d) for a range of standard and hypothetical glass compositions with increased fissile content. Table 3-12 provides a summary of the results that demonstrated that the analyzed representative DSNF types would remain subcritical with a significant margin even with orders-of-magnitude increase of fissile material in the glass. The high concentration of boron in the glass provides for the significant margin of subcriticality for higher fissile concentrations. Therefore, ABD glass will not require any additional controls to meet preclosure criticality safety requirements.



NOTE: ATR = Advanced Test Reactor
DSNF = DOE spent nuclear fuel
HLW = high-level radioactive waste

Figure 3-3. ATR SNF Canister with HLW Glass Canisters in a CDSP Waste Package

Table 3-11. k_{eff} Values for Individual Undamaged and Dry DSNF Canisters under a Variety of Close-Fitting Reflection Conditions

Close-Fitting 30 cm Thick Reflector	DSNF Canister Type						
	ATR	Enrico Fermi	FFTF	Fort St. Vrain	Shippingport LWBR	Shippingport PWR	TRIGA
Alloy 22	0.132	0.550	0.693	0.258	0.346	0.169	0.58
Concrete	0.184	0.474	0.612	0.289	0.318	0.199	0.540
HLW Glass	0.103	0.490	0.634	0.182	0.292	0.137	0.558
Lead	0.134	0.560	0.706	0.335	0.375	0.170	0.597
Natural U Metal	0.424	0.570	0.692	0.481	0.453	0.428	0.623
Stainless Steel	0.149	0.544	0.694	0.296	0.367	0.185	0.593
Titanium	0.097	0.473	0.628	0.220	0.316	0.128	0.555
Tuff	0.185	0.480	0.620	0.291	0.324	0.201	0.545
Water	0.151	0.396	0.532	0.256	0.261	0.163	0.503

NOTE: ATR = Advanced Test Reactor
DSNF = DOE spent nuclear fuel
FFTF = Fast Flux Test Facility
HLW = high-level radioactive waste
LWBR = Light Water Breeder Reactor
PWR = pressurized water reactor
TRIGA = Training, Research, Isotope, General Atomics

Source: BSC 2008c, Table 7-1.

Table 3-12. k_{eff} Values for an Individual Undamaged and Dry DSNF Canister with Close Fitting Reflection by Dry HLW Glass with a Range of Fissile Concentrations

Fissile Concentration (g/L)	DSNF Canister Type						
	ATR	Enrico Fermi	FFTF	Fort St. Vrain	Shippingport LWBR	Shippingport PWR	TRIGA
0.5	0.104	0.501	0.632	0.181	0.293	0.138	0.557
5.0	0.109	0.503	0.635	0.184	0.294	0.141	0.557
50.0	0.176	0.519	0.648	0.235	0.316	0.195	0.566

NOTE: ATR = Advanced Test Reactor
DSNF = DOE spent nuclear fuel
FFTF = Fast Flux Test Facility
HLW = high-level radioactive waste
LWBR = Light Water Breeder Reactor
PWR = pressurized water reactor
TRIGA = Training, Research, Isotope, General Atomics

Source: BSC 2008d, Table 7-2.

3.2.3 GROA Worker Dose and Shielding Design

The GROA worker dose estimates and shielding calculations are documented in facility-specific worker dose assessment calculations, the results of which are summarized in *GROA Worker Dose Calculation* (BSC 2008a). These calculations are supported by waste form specific dose rate calculations based on conservative source term estimates developed for bounding waste form compositions. The SRS HLW canisters proved to be the bounding HLW canister type for radial dose rates, whereas the Hanford HLW canister proved to be the bounding type for axial dose rates (BSC 2008i). The worker dose estimates are also supported by time and motion studies.

The ABD glass composition change relative to the currently assumed composition (provided in Appendix B-1) is not anticipated to result in a change to the gamma or neutron source term. The composition change does not reflect significant addition of gamma emitters. Because of the significant concentration of boron in the glass, the neutron source term would remain minimal. Additionally, the gamma dose rate for HLW glass accounts for essentially >99% of the total dose rate (BSC 2008b, Section 6.3.4; BSC 2003a).

HLW glass is handled in the same facilities that handle CSNF. The shielding for these facilities is designed based on the limiting CSNF canisters with the extremely conservative source term estimates based on conservative assembly characteristics. For example, the maximum pressurized water reactor (PWR) assembly source term is based on a Babcock and Wilcox 15×15 assembly with uranium loading of 475 kg at a burnup of 80 GWd/MTU and 5-year cooling time.

The PCSA concluded that there are no Category 1 event sequences that must be included in the worker dose estimates; ABD glass would not change this conclusion (as discussed in Section 3.2.2). Therefore, the proposed ABD glass will not affect the preclosure shielding design or worker dose estimates.

3.2.4 Preclosure Design and Operations

A new or modified waste form could affect the design of the surface and subsurface operational facilities if it were to adversely affect any aspect related to physical characteristics, preclosure nuclear safety, criticality safety, or radiological protection. The proposed ABD glass canisters are identical physically to the HLW glass canisters; therefore, no changes to handling or staging equipment design envelopes would be needed. As discussed in Section 3.2.1 and Section 3.2.2, no new design requirements or safety structures, systems, and/or components (SSCs) are expected to be required for ABD glass. In addition, the GROA worker dose estimates and shielding requirements are not expected to be affected by ABD glass (Section 3.2.3). Therefore, acceptance of ABD glass is not expected to affect preclosure design or operations.

3.3 Postclosure Performance Evaluation Results

Applying the approach described in Section 2.2 resulted in identifying the FEPs, models, and modeling cases presented in Section 2 of the SAR (DOE 2008) that could be affected by implementing the ABD Program. The subsections below discuss the evaluation results with respect to criticality FEPs (Section 3.3.1), noncriticality FEPs and models (Section 3.3.2), and modeling cases (Section 3.3.3). In addition, a consolidated list of the FEPs determined to be potentially affected by implementing the ABD Program—8 criticality FEPs and 27 noncriticality FEPs—is provided in Appendix A.

3.3.1 Criticality FEPs

The primary bases for postclosure criticality FEPs screening for CDSP waste packages are documented in *DOE SNF Phase I and II Summary Report* (Radulescu et al. 2004), which is supported by the following CDSP waste package viability reports for each DSNF type:

- *Evaluation of Codisposal Viability for Th/U Carbide (Fort Saint Vrain HTGR) DOE-Owned Fuel* (BSC 2001e)
- *Evaluation of Codisposal Viability for MOX (FFTF) DOE-Owned Fuel* (CRWMS M&O 1999f)
- *Evaluation of Codisposal Viability for UZrH (TRIGA) DOE-Owned Fuel* (CRWMS M&O 2000e)
- *Evaluation of Codisposal Viability for HEU Oxide (Shippingport PWR) DOE-Owned Fuel* (CRWMS M&O 2000c)
- *Evaluation of Codisposal Viability for U-Zr/U-Mo Alloy (Enrico Fermi) DOE-Owned Fuel* (CRWMS M&O 2000f)
- *Evaluation of Codisposal Viability for Th/U Oxide (Shippingport LWBR) DOE-Owned Fuel* (CRWMS M&O 2000d)
- *Evaluation of Codisposal Viability for U-Metal (N Reactor) DOE-Owned Fuel* (CRWMS M&O 2001)
- *Evaluation of Codisposal Viability for Melt and Dilute DOE-Owned Fuel* (BSC 2001d)

Note that when the *DOE SNF Phase I and II Summary Report* was issued, the plan was to treat aluminum-based fuels using a process named “melt and dilute,” which would result in an aluminum ingot that would be placed in a DSNF canister. Subsequent documentation was developed for disposal of ATR SNF as the representative fuel type for aluminum-based fuels (BSC 2004f). The criticality analyses for TMI were completed after the issuance of this report (Radulescu et al. 2004) and are summarized in *Intact* *and Degraded Mode Criticality Calculations for the Codisposal of TMI-2 Spent Nuclear Fuel in a Waste Package* (BSC 2004g).

The CDSP viability reports are supported by a suite of calculations that cover design calculations (e.g., thermal, structural), intact and degraded criticality calculations, geochemistry degradation and release analyses, and geochemistry accumulation analyses.

The criticality calculations for the various DSNF types evolved based on fuel loading considerations, basket and shot material, and treatments plans (CRWMS M&O 1999a; CRWMS M&O 1999g; CRWMS M&O 2000g; CRWMS M&O 2000h; CRWMS M&O 2000a; BSC 2001f; BSC 2002a; BSC 2004f; BSC 2004g; BSC 2004b; BSC 2006a; SNL 2007b). The level of conservatism in these analyses also evolved based on input from material degradation and release analyses and more credible considerations for the 10,000-year regulatory period.

The geochemistry models and analyses supporting the degraded and external criticality calculations for the various DSNF types also evolved based on improvements in codes, models, and thermodynamic databases from the late 1990s to 2007 (CRWMS M&O 1998a; CRWMS M&O 1999d; CRWMS M&O 1999c; CRWMS M&O 1999b; CRWMS M&O 1999e; CRWMS M&O 2000b; BSC 2001c; BSC 2001b; BSC 2001a; BSC 2002c; BSC 2002b; BSC 2003b; BSC 2004e; SNL 2010b). The latest geochemistry models are documented in *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007f) and *Geochemistry Model Validation Report: External Accumulation Model* (SNL 2007e).

The following subsections provide a summary of the results of the analysis steps described in Section 2.2.

3.3.1.1 Postclosure In-Package Criticality FEPs

The disposal criticality analysis was unique for each of the DSNF types, unlike CSNF for which disposal analysis basis configurations were developed for the entire population based on the neutronic characteristics of lumped low-enriched fuels that can be bounded by a manageable set of configurations. The configuration classes considered in the disposal criticality analysis for each DSNF type can be grouped into two classes for the purposes of the evaluation in this subsection, as follows:

- Configurations for which the degraded internal components of the DSNF canisters remain separated from the external degraded components
- Configurations for which the degraded components of the entire waste package including SNF, glass, and structure are intermingled

The potential for criticality during disposal is determined based on calculating k_{eff} for a range of configuration classes with probability of occurrence above a screening threshold. The configurations are modeled using MCNP and include the uncertainty associated with the Monte Carlo analysis at 95% confidence. A configuration is considered subcritical if k_{eff} is less than the critical limit for the configuration. The critical limit is a statistically derived limit that takes into account the bias, bias uncertainty, and penalties associated with benchmarking (e.g., range of applicability of benchmark experiments and range of parameters of specific configuration). An administrative margin is not considered for the postclosure critical limit because the objective of the disposal analysis is not to ensure subcriticality, but rather to calculate the probability of postclosure criticality. Taking all these parameters into account, critical limits and critical limit functions were derived for each DSNF type (BSC 2004i). The critical limit functions are based on relevant physical parameters (e.g., hydrogen-to-fissile-material ratio) or spectral parameters (e.g., energy of average lethargy causing fission). Most of the DSNF codisposal viability analyses referenced in this report were performed prior to final benchmarking and development of the critical limit functions; a conservative interim critical limit of 0.93 was assumed. Therefore, a configuration with a k_{eff} value less than the critical limit is considered subcritical. Note that the real margin of subcriticality is not just associated with the k_{eff}-to-critical limit comparison, but rather the underlying conservative and often bounding assumptions in the model, including composition, moderation, geometry, reflection, and interaction assumptions. A k_{eff} value close to the critical limit does not necessarily mean a small margin with the associated context of the configuration assumptions.

The evaluation documented in this subsection summarizes the effect of ABD glass (increased fissile material and addition of gadolinium) on the bounding cases for each of these two configuration classes.

Table 3-13 provides a summary of the relevant DSNF representative types, basket designs, and criticality control considerations. The SRS HLW glass canisters were planned to be placed in the 5-DHLW/DOE Short waste package. Therefore, the candidate DSNF types that could be codisposed with ABD glass are as follows:

- Uranium-zirconium hydride (UZrH) fuels represented by Training, Research, Isotope, General Atomics (TRIGA) SNF
- Mo and U-Zr Alloys represented by Enrico Fermi SNF
- Highly enriched uranium-Al fuels represented by ATR SNF. Note that this fuel could be placed in short DSNF standardized canisters (two stacked baskets of 10 elements each) or long DSNF standardized canisters (three stacked baskets).

Codisposal of short HLW glass canisters with long DOE standardized SNF canisters was not analyzed for Yucca Mountain. Nonetheless, this subsection includes a summary evaluation of the codisposal of DSNF in long DOE standardized SNF canisters with glass waste produced by implementing the ABD Program to provide perspective.

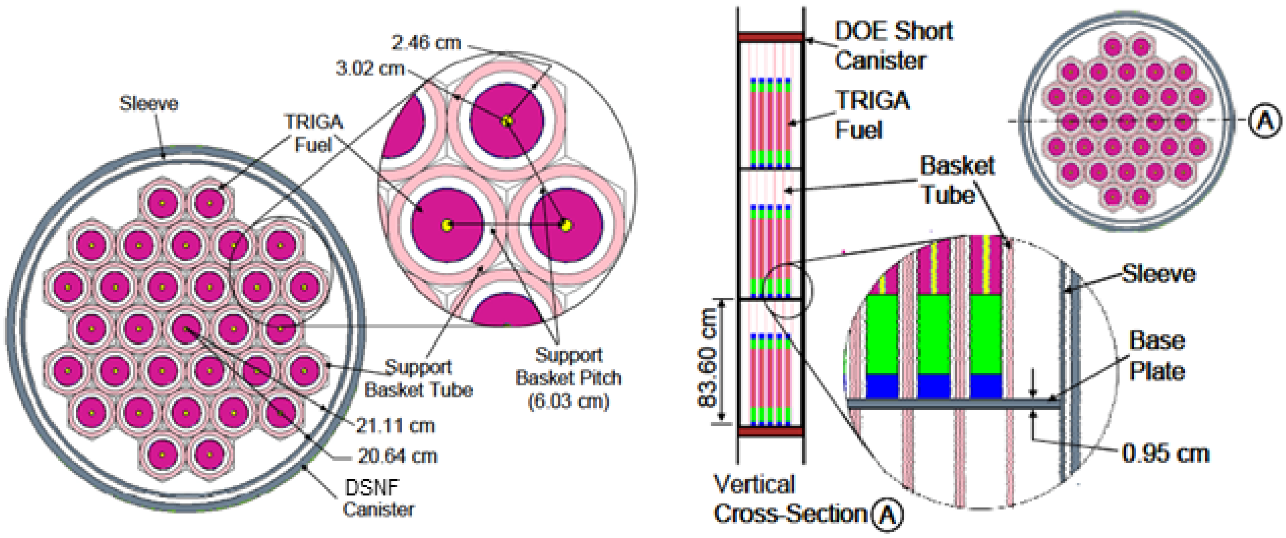
Table 3-13. Summary of Basket Designs and Criticality Control Considerations for the Representative DSNF Types

Representative Fuel Type	Canister Type	Basket Type	Basket Material	Shot	Fuel Loading
TRIGA	18 in.–Short	Pipe bundle	Ni-Gd alloy with 8 wt% Gd	None	93 rods in three stacks
Enrico Fermi	18 in.–Short	Pipe bundle	Stainless steel or Ni-Gd alloy	$Fe/GdPO_4$	22 cans in two stacks
ATR	18 in.–Short or Long	Rectangular grid	Ni-Gd alloy	$Al/GdPO_4$ shot is one of the evaluated options	20 or 30 elements in 2 or 3 stacks
TMI-2 debris	18 in.–Long	Single centering pipe	Stainless steel	None	A single canister
Fort St. Vrain	18 in.–Long	None	None	None	5 stacked blocks
Shippingport LWBR	18 in.–Long	Single centering rectangle	Stainless steel	$Al/GdPO_4$	A single seed assembly
Shippingport PWR	18 in.–Long	Single centering square	Stainless steel	None	A single assembly
FFTF	18 in.–Long	Spoke and wheel	NiG-Gd alloy	$Al/GdPO_4$	5 DFAs or 4 DFAs and an Ident

NOTE: ATR = Advanced Test Reactor
DFA = driver fuel assembly
FFTF = Fast Flux Test Facility
LWBR = Light Water Breeder Reactor
PWR = pressurized water reactor
TMI-2 = Three Mile Island Unit 2
TRIGA = Training, Research, Isotope, General Atomics

3.3.1.1.1 Codisposal of ABD Glass with TRIGA SNF

TRIGA SNF is the representative fuel type for the UZrH fuel group. The highly enriched uranium fuel, life improvement program variant is used in the analysis. A stainless-steel fuel element contains 70 wt% enriched ^{235}U in a self-moderating zirconium-hydride matrix ($UZrH_{1.6}$). A short DOE standardized SNF canister contains three baskets, each holding 31 fuel elements, with basket tubes made of nickel-gadolinium alloy with a nominal gadolinium loading of 2.0 wt%. Note that the disposal criticality analyses were based on baskets with 37 fuel elements each (a total of 111 fuel elements per canister) and only 12 rods with nickel-gadolinium alloy inserts in various arrangements; the gadolinium loading for these inserts was assumed to be 8 wt%. The total fissile mass in the canister is 15.2 kg ^{235}U. Figure 3-4 provides illustrations of the most recent basket design.



NOTE: DSNF = DOE spent nuclear fuel
TRIGA = Training, Research, Isotope, General Atomics

Figure 3-4. TRIGA SNF and Fuel Basket

Table 3-14 provides a summary of the most reactive configurations for TRIGA SNF and bases for subcriticality. The results for the degraded analysis indicate that the highest k_{eff} is achieved for intact fuel rods in a degraded waste package. This configuration may be characterized by intact TRIGA SNF settled in the bottom of the waste package, surrounded by clay. This configuration is highly unlikely because the fuel could not stay intact while all other components degrade based supporting geochemistry analyses. For this configuration, 4.1 kg Gd homogenized in a layer of clay that covers only the fuel is needed to maintain subcriticality. Based on a maximum gadolinium loss of approximately 55% from the supporting geochemistry calculations, at least 8.9 kg Gd were required to be included in the DSNF canister. More recent geochemistry modeling documented in *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007f) demonstrated that gadolinium loss from CDSP waste packages is negligible (i.e., <1%) during the 10,000-year regulatory period (SNL 2007b, Section 6.5.3.1).

Table 3-14. Most Reactive Configurations for TRIGA SNF and Bases for Subcriticality

Configuration Class Description	Maximum k_{eff}	Minimum Required Gadolinium
Intact flooded configurations: SNF and internal structures notdegraded	0.789	None
SNF partially degraded in place(various stages)	0.816	None
SNF partially or totally degraded inside the waste package with intact internal structures	0.834	At least 1.5 kg (17% of initial Gd amount) needed to remain between rods
Both SNF and internal structures ofthe waste package degraded (various stages)	0.636	No Gd needed
SNF intact (as assembly or pins) and degraded internal structures (variousstages)	0.915	4.1 kg of Gd homogenized in layer of clay that covers fuel.

NOTE: SNF = spent nuclear fuel

Source: Radulescu et al. 2004, Table 10-6; CRWMS M&O 2000c, Table 6-1.

Increasing the fissile material in the HLW glass would not be expected to significantly enhance its reflection properties due to the high concentration of absorbers in the glass (boron and gadolinium). Therefore, an ABD glass canister would not appreciably increase the maximum k_{eff} for the configuration classes in which the degraded internal components of the DSNF canisters remain separated from the external degraded components of the waste package. As presented in Table 3-14, the maximum k_{eff} for this configuration class is 0.816.

The fissile mass in a TRIGA SNF canister based on the limiting TRIGA SNF type is 15.2 kg ^{235}U (Radulescu 2004, Section 10.2). An ABD glass canister with 2,500 g ^{235}U would increase the fissile material mass in the waste package by approximately 16%. The enrichment of the uranium in the ABD glass of 5 wt% ^{235}U is much less than the enrichment of TRIGA SNF at 70 wt% ^{235}U. The Gd loading in ABD glass at a Gd/fissile ratio of one is greater than the required Gd loading for TRIGA SNF at a Gd/fissile ratio of ~0.6 (calculated from Radulescu 2004, Section 10.2). Therefore, an ABD glass canister would likely not increase the maximum k_{eff} for the configuration classes in which the degraded components of the entire waste package including SNF, glass, and structure are intermingled. For the fully degraded case, the presence of ABD glass could in fact reduce the uranium enrichment such that less gadolinium may be required to maintain subcriticality for these configurations. As presented in Table 3-14, the maximum k_{eff} for this configuration class is 0.915.

It is anticipated that codisposal of TRIGA canisters with ABD glass would have relatively minor effects on the reactivity of the most limiting degraded configurations. This conclusion would have to be supported by updating the existing models and analyses with the ABD glass composition. The outcome of these analyses would likely support one or more of the following conclusions:

- UZrH represented by TRIGA, with the current basket design and neutron absorber loading, can be codisposed with up to "*x*" ABD glass canisters. The actual number of ABD glass canisters may be five if the degradation analyses conclude that there is no credible degradation scenario that results in significant preferential release of the gadolinium in the ABD glass. As discussed above, this situation is likely to be the case based on the conclusions of the *Geochemistry Model*

Validation Report: Material Degradation and Release Model (SNL 2007b), which demonstrated that gadolinium loss from CDSP waste packages is negligible (i.e., <1%) during the 10,000-year regulatory period (SNL 2007a, Section 6.5.3.1).

- UZrH fuel represented by TRIGA, with the current basket design but with increased poison loading, can be codisposed with up to "*x*" ABD glass canisters. The use of shot was explored and shown effective for TRIGA SNF (CRWMS M&O 1999g).
- Only select types of UZrH fuels with lower enrichment (e.g., standard TRIGA fuels with 20 wt% ^{235}U, which represents a large fraction of the fuel group) can be codisposed with up to "*x*" ABD glass canisters.

3.3.1.1.2 Codisposal of ABD Glass with Fermi SNF

Enrico Fermi fast reactor fuel is the representative type for the U-Mo and U-Zr Alloys fuel group. The Enrico Fermi fuel pin matrix is made of uranium-molybdenum alloy (approximately 10 wt% molybdenum alloyed with uranium of 25.69 wt% ^{235}U enrichment). A fuel section of 140 pins has a total mass of approximately 22.3 kg, of which 4.8 kg is ^{235}U. The fuel is metallurgically bonded to a zirconium tube that serves as cladding, resulting in no gap between cladding and fuel. Zirconium end pieces are fitted to the fuel rods, and 140 fuel rods plus 4 stainless-steel connecting rods were installed in each fuel assembly.

A short DSNF standardized canister contains two basket assemblies, with each basket assembly consisting of 12 nickel-gadolinium alloy tubes. Each tube contains an −01 aluminum canister, which itself contains an −04 aluminum canister holding 140 loose fuel pins from a single assembly. The total fissile mass in the canister is 115.6 kg ^{235}U. The space between the tubes is filled with $Fe/GdPO_4$ shot. Cross-sectional views of fuel tubes and a short DOE standardized SNF canister containing Enrico Fermi SNF are shown in Figure 3-5.

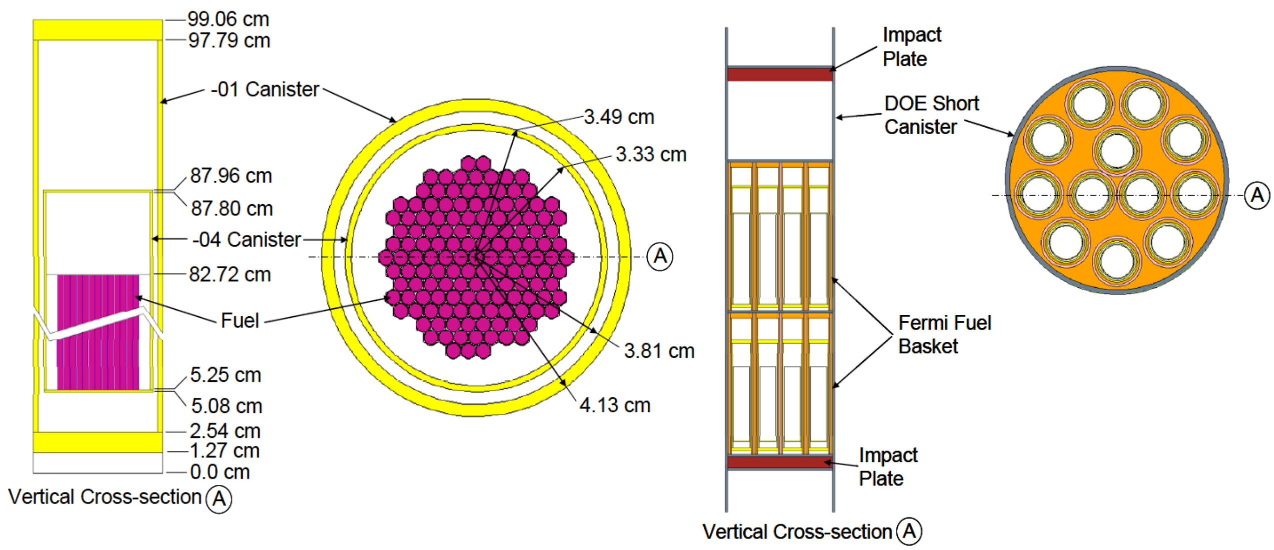


Figure 3-5. Enrico Fermi SNF and Fuel Basket

Table 3-15 provides a summary of the most reactive configurations and bases for subcriticality. The results for the degraded analysis indicate that the highest k_{eff} is achieved for settled fuel pins at the bottom of the waste package immersed in a uniform mixture of the degradation products of the DSNF canister (FeOOH, AlOOH, and $GdPO_4$) covered with a layer of clay (from degraded glass). For the configuration with optimum moderation to remain subcritical, a minimum quantity of 9 kg Gd must be uniformly distributed in the mixture. The supporting geochemistry calculations showed that the maximum gadolinium loss is less than 2.3%. More recent geochemistry modeling documented in *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007f) demonstrated that gadolinium loss from CDSP waste packages is negligible (<1%) during the 10,000-year regulatory period (SNL 2007b, Section 6.5.3.1).

Table 3-15. Most Reactive Configurations for Enrico Fermi SNF and Bases for Subcriticality

Configuration Class Description	Maximum k_{eff}	Minimum Required Gadolinium
Intact flooded configurations: SNF and internal structures notdegraded	0.835	None
SNF partially degraded in place(various stages)	0.884	6 kg of Gd in degraded canister.
Both SNF and internal structures ofthe waste package degraded (various stages)	0.905	9 kg of Gd homogenized in waste mixture.
SNF intact (as assembly or pins) and degraded internal structures (variousstages)	0.922	9 kg of Gd homogenized in layer of clay that covers fuel.

NOTE: SNF = spent nuclear fuel

Source: Radulescu et al. 2004, Table 10-3; CRWMS M&O 2000f, Section 7.5.1.

Increasing the fissile material in the HLW glass would not be expected to significantly enhance its reflection properties due to the high concentration of absorbers in the glass (boron and gadolinium). Therefore, an ABD glass canister would not appreciably increase the maximum k_{eff} for the configuration classes in which the degraded internal components of the DSNF canisters remain separated from the external degraded components of the waste package. As presented in Table 3-15, the maximum k_{eff} for this configuration class is 0.884.

The fissile mass in a Fermi SNF canister is 115.6 kg ^{235}U (Radulescu 2004, Section 10.4). An ABD glass canister with 2,500 g ^{235}U would increase the fissile material mass in the waste package by approximately 2%. The enrichment of the uranium in the ABD glass of 5 wt% ^{235}U is much less than the enrichment of the Enrico Fermi SNF at 25.69 wt% ^{235}U. The Gd loading in ABD glass at a Gd/fissile ratio of one is greater than the required loading for Fermi SNF at a Gd/fissile ratio of ~0.08 (calculated from Radulescu 2004, Section 10.4). Therefore, an ABD glass canister would not be anticipated to increase the maximum k_{eff} for the configuration classes in which the degraded components of the entire waste package including SNF, glass, and structure are intermingled. For the fully degraded case, the presence of ABD glass could in fact reduce the reactivity of this configuration due to the added gadolinium. As presented in Table 3-15, the maximum k_{eff} for this configuration class is 0.922.

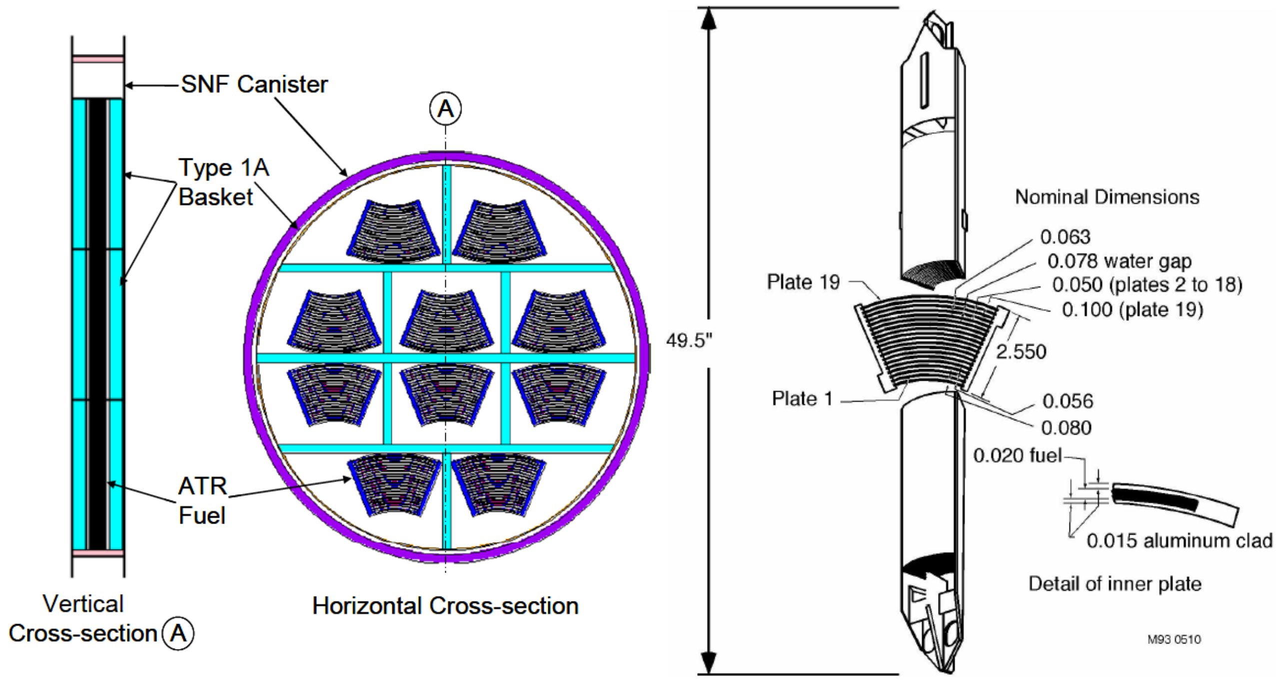
It is anticipated that codisposal of Fermi canisters with ABD glass would have relatively minor effects on the reactivity of the most limiting degraded configurations. This conclusion would have to be supported by updating the existing models and analyses with the ABD glass composition. The outcome of these analyses would likely support one or more of the following conclusions:

- Mo and U-Zr Alloys represented by Fermi, with the current basket design and neutron absorber loading, can be codisposed with up to "*x*" ABD glass canisters. The actual number of ABD glass canisters may be five if the degradation analyses conclude that there is no credible degradation scenario that results in significant preferential release of the Gd in the ABD glass. As discussed above, this is likely to be the case based on the conclusions of the *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007b), which demonstrated that gadolinium loss from CDSP waste packages is negligible (i.e., <1%) during the 10,000-year regulatory period (SNL 2007a, Section 6.5.3.1).
- Mo and U-Zr Alloys represented by Fermi, with the current basket design but with increased absorber loading in the form of $GdPO_4$ shot, can be codisposed with up to "*x*" ABD glass canisters.

3.3.1.1.3 Codisposal of ABD Glass with ATR SNF

ATR fuel is the representative fuel type for the Al-based fuel group, the largest fuel group based on canister count. The ATR fuel element consists of 19 curved aluminum-clad uranium aluminide (UAlx) plates containing highly enriched (93±1 wt% ^{235}U) uranium. The highest nominal fuel loading for a fresh fuel element is 1,075 g of ^{235}U. A short or long DSNF standardized canister contains two or three baskets each holding 10 fuel elements. The basket plates are made of nickel-gadolinium alloy with a nominal gadolinium loading of 2 wt%. The use of aluminum shot with $GdPO_4$ was also investigated. The total fissile material mass in short ATR canister is 21.5 kg ^{235}U. Cross-sectional views of a short DOE standardized SNF canister containing ATR SNF are shown in Figure 3-6.

Table 3-16 provides a summary of the most reactive ATR configurations and bases for subcriticality. The results for the degraded analysis indicate that the highest k_{eff} is achieved for degraded ATR SNF within the compartments of the basket. For the configuration with optimum moderation to remain subcritical, the addition of aluminum shot is required. For the fully degraded case, the criticality analyses concluded that retention of 50% of the gadolinium from the degraded basket would be sufficient to maintain subcriticality. The supporting geochemistry analysis (SNL 2010b, Section 4.2) concluded that 99.55% of the gadolinium would be retained for bounding seepage flux assumptions.



NOTE: ATR = Advanced Test Reactor
SNF = spent nuclear fuel

Figure 3-6. ATR SNF and Fuel Basket

Table 3-16. Most Reactive Configurations for ATR SNF and Bases for Subcriticality

Configuration Class Description	Maximum k_{eff}	Minimum Required Gadolinium
Intact flooded configurations: SNF and internal structures notdegraded	0.715	None
SNF partially degraded in place (various stages)	0.696	Ni-Gd basket, aluminum shot modeled with minimal 0.05 wt% Gd. Note that the k_{eff} for the case without the aluminum is unacceptably high (0.998).
Degraded SNF inside canister and degraded internal structures of the waste package (various stages)	0.645	Ni-Gd basket.
Fully degraded SNF and waste package components)	0.829	Retention of 50% of Gd from degraded basket.

NOTE: SNF = spent nuclear fuel

Source: BSC 2004f, Table 18.

Increasing the fissile material in the HLW glass would not be expected to significantly enhance its reflection properties due to the high concentration of absorbers in the glass (boron and gadolinium). Therefore, an ABD glass canister would not appreciably increase the maximum k_{eff} for the configuration classes in which the degraded internal components of the DSNF canisters remain separated from the external degraded components of the waste package. As presented in Table 3-16, the maximum k_{eff} for this configuration class is 0.696 (BSC 2004f, Table 18).

The fissile mass in a short (10 ft) ATR canister is 21.5 kg ^{235}U (BSC 2004f, Section 5.1.1). An ABD glass canister with 2,500 g ^{235}U would increase the fissile material mass in the ATR waste package by approximately 12%. The enrichment of the uranium in the ABD glass of 5 wt% ^{235}U is much less than the enrichment of the ATR SNF at 93 wt% ^{235}U. For the partially degraded-in-place cases, ABD glass would not affect the reactivity of the system. An ABD glass canister is not anticipated to increase the maximum k_{eff} for the configuration classes in which the degraded components of the entire waste package including SNF, glass, and structure are intermingled. For the fully degraded case, the presence of ABD glass could in fact reduce the enrichment of the uranium and increase the gadolinium concentration further increasing the margin of subcriticality for these configurations. As presented in Table 3-16, the maximum k_{eff} for this configuration class is 0.829 crediting only 50% of the gadolinium from the degraded basket.

It is anticipated that codisposal of ATR canisters with ABD glass would have relatively minor effects on the reactivity of the most limiting degraded configurations. This conclusion would have to be supported by updating the existing models and analyses with the ABD glass composition. The outcome of these analyses would likely support one or more of the following conclusions:

- Aluminum-based fuels represented by ATR, with the current basket design and neutron absorber loading, could be codisposed with up to "*x*" ABD glass canisters. The actual number of ABD glass canisters may be five if the degradation analyses conclude that there is no credible degradation scenario that results in significant preferential release of the Gd in the ABD glass. As discussed above, this is likely to be the case based on the supporting geochemistry analysis (SNL 2010, Section 4.2), which concluded that 99.55% of the gadolinium would be retained for bounding seepage flux assumptions.
- Aluminum-based fuels represented by ATR, with the current basket design but with increased absorber loading in the form of $GdPO_4$ shot, can be codisposed with up to "*x*" ABD glass canisters.

3.3.1.1.4 Codisposal of ABD Glass with Long DOE Standardized SNF Canisters

The five DSNF types in the long DSNF standardized SNF canister are Fast Flux Test Facility (FFTF), Fort St. Vrain, Three Mile Island Unit 2 (TMI-2) debris, Shippingport PWR, and Shippingport Light Water Breeding Reactor (LWBR). Codisposal of these fuels with short HLW glass canisters was not analyzed for Yucca Mountain; nonetheless, they are discussed in this subsection to provide perspective.

TMI-2 debris is the representative fuel type for the low-enriched uranium oxide fuel group. The typical fuel assembly constituting the debris is a Babcock & Wilcox 15×15 with maximum enrichment of 2.96 wt% ^{235}U and a maximum beginning-of-life ^{235}U content of 13.72 kg. A long DSNF standardized canister contains one of three TMI-2 canister types: a defueling canister holding debris large enough to grapple, a knockout canister holding vacuumed debris, or a filter canister holding debris caught in filters. The uranium loading for a single TMI-2 canister ranges from zero to 441.9 ±99.9 kg

Fort St. Vrain fuel is the representative fuel type for the U/Th Carbide fuel group. It consists of a mixture of small spheres of the order of 0.0450 to 0.0750 cm diameter of uranium (enriched to 93.5 wt% ^{235}U) and thorium carbide. The individual spheres are coated with multiple, thin layers of pyrolytic carbon and silicon carbide, which serve as tiny pressure vessels to contain fission products and the Th/U carbide matrix. In the fuel elements, the coated spheres are bound in a carbonized matrix to form fuel compacts that are loaded into drilled holes in a large hexagonal graphite prism comprising one fuel element. Fuel holes containing the fuel compacts and coolant channels are distributed in a triangular array within the fuel element. A long DSNF standardized canister contains five hexagonal graphite fuel elements with no basket.

Shippingport LWBR fuel is the representative fuel type for the ^{233}U/Th Oxide fuel group. A seed assembly contains 8 types of seed rods in four fuel regions, with a total of 619 cylindrical fuel rods in a triangular pitch array, supported by a hexagonal Zircaloy-4 outer shell. The fuel rods contain either thoria (ThO_2) or a mixture of thoria and UO_2. The uranium is 98.23 wt% ^{233}U. Two different enrichments of the UO_2-ThO_2 matrix were used, a lower enrichment of 4.337 wt% and a higher enrichment of 5.202 wt%. A long DSNF standardized canister contains a rectangular stainless-steel basket holding a single LWBR seed assembly. The space not occupied by the fuel assembly and basket is filled with aluminum shot containing $GdPO_4$.

Shippingport PWR fuel is the representative fuel type for the highly enriched uranium oxide fuel group. The Core 2 Seed 2 fuel cluster is used because it has a higher ^{235}U loading per cluster than other types of fuel clusters. It is composed of four fuel subclusters arranged in a square array with spacing between them that accommodated a cruciform-shaped control rod during operation. Each subcluster contains 19 fuel and two neutron absorber plates. A fuel plate is formed by sandwiching UO_2-ZrO_2-CaO alloy wafers between two Zircaloy-4 cover plates and four side strips. The initial ^{235}U enrichment is 93.2 wt%. A long DSNF standardized canister contains a single square basket of stainless-steel guide plates holding a single PWR fuel cluster.

FFTF fuel is the representative type for the mixed-oxide fuel group. There are four basic types of fuel pins and one experimental variant. Each pin contains mixed oxide fuel ($UO_{1.96}$ and $PuO_{1.96}$) surrounded by stainless-steel cladding. The FFTF standard driver fuel assembly (DFA) contains 217 Type 4.1 fuel pins (which have the highest fissile material content) within a stainless-steel Type 316 hexagonal duct. Some assemblies have been disassembled, and up to 217 fuel pins have been placed in a 5 in. stainless-steel pipe known as an Ident-69 container. A long DSNF standardized canister contains a spoked-wheel basket constructed of nickel-gadolinium alloy holding five DFAs surrounding a single Ident-69 container. Only five of the six basket compartments will be used for any fully loaded canister. The space not occupied by the fuel assemblies, the Ident-69 container, or the basket is filled with aluminum shot containing $GdPO_4$.

The canisters for Fort St. Vrain, Shippingport PWR, and TMI-2 do not include added neutron absorbers in the baskets or in the form of shot. Owing to its corrosion resistant properties, the limiting credible configuration for Fort St. Vrain is for intact fuel assemblies inside a failed canister, which is subcritical without requiring additional neutron absorbers. The most limiting credible configuration for Shippingport PWR is also an intact assembly in a failed canister. Based on the near-optimal assembly design, this configuration is subcritical without requiring additional neutron absorbers. The most reactive credible configuration for TMI-2 fuel debris is subcritical based on assumptions of debris size and ranges of physically possible void fractions (i.e., no pellets floating in water). Increasing the fissile material in the HLW glass does not significantly enhance its reflection properties due to the presence of neutron absorbers in the glass. Therefore, an ABD glass canister is not anticipated to increase the maximum k_{eff}

for these configuration classes in which the degraded internal components of the DSNF canisters remain separated from the degraded components of the waste package. For fully degraded configurations (i.e., waste package components mixed with degraded fuel), codisposal with ABD glass would lower the enrichment of the degraded material composition and/or (in the case of TMI-2) add gadolinium. Therefore, codisposal with ABD glass could in fact lower the reactivity of these configurations, albeit they are not limiting for these fuels. This conclusion would have to be supported by updating the existing models and analyses with the ABD glass composition.

FFTF canisters include neutron absorbers both in the basket structure and Al/$GdPO_4$ shot for canisters with an Ident-69 canister. Although it is not necessarily the most reactive fuel type within the group, FFTF fuels contain a large majority of total fissile material. Due to the fissile material type (^{239}Pu), plutonium enrichment (up to 29.28 wt%), and mass (~9 kg ^{239}Pu per DFA), there are a variety of configuration classes ranging from partially to fully degraded that require crediting neutron absorbers either in the baskets or in the degraded mixture. Codisposal of FFTF with ABD glass is not anticipated to increase the maximum reactivity of configuration classes in which the degraded internal components of the DSNF remain separated from the ABD glass canisters. The current analysis for fully degraded configurations cannot be readily extrapolated to ABD glass due to the differing fissile material type and the fact that the added gadolinium in the ABD glass is a small fraction of what is already incorporated in the FFTF canister. Nonetheless, if FFTF is considered for codisposal with ABD glass, the criticality and supporting geochemistry analysis would have to be updated; the updated analysis may require higher concentration of gadolinium in the shot or limit the types of FFTF canisters that can be codisposed with ABD glass to those without an Ident-69.

Shippingport LWBR canisters include absorbers in the form of Al/$GdPO_4$ shot. Several partially and fully degraded configurations require crediting gadolinium to remain subcritical. Similar to the other highly enriched uranium fuel types, codisposal of Shippingport LWBR with ABD glass is not anticipated to increase the reactivity of configurations with degraded fuel and intact waste package components and may in fact lower the reactivity of the fully degraded configurations due to lowering the enrichment of the composition and increasing the gadolinium concentration.

3.3.1.2 Postclosure External Criticality FEPs

Postclosure criticality FEPs address criticality potential in the near field (invert) and far field (fractured tuff and lithophysae) under normal conditions, under conditions associated with rock fall, and under conditions associated with seismic and igneous events. External criticality potential is addressed in *Geochemistry Model Validation Report: External Accumulation Model* (SNL 2007e). The SNF types addressed in *Geochemistry Model Validation Report: External Accumulation* are CSNF, N Reactor, TMI and FFTF SNF. Table 3-17 (SNL 2007e, Table 6.9-1[a]) presents the ranges of minimum critical mass required to accumulate in the invert and far field as well as the calculated accumulation from the waste package for the SNF types evaluated for external criticality. The results indicate that under bounding seepage fluxes, an insufficient amount of fissile material would be release and/or accumulate to pose a criticality concern. Other DSNF types (e.g., Shippingport LWBR, Fort St. Vrain) are not expected to increase the potential for external criticality due to corrosion resistance of the fuel, cladding, or coating materials. TRIGA, Shippingport PWR, ATR, and Fermi SNF have not been analyzed in detail for external fissile mass transport and accumulation. However, considering the processes that must occur to release and accumulate fissile material in the near field and far field, these waste forms are not expected to result in an increase in the total probability of criticality.

The NRC evaluation of external criticality FEPs documented in its SER (NRC 2015b) states:

> "The NRC staff considered that two of the most significant DOE assumptions unique to near- and far-field criticality were (i) fissile material would accumulate in the optimum geometry for criticality and (ii) neutron absorbers and fission products would not be located nearby. Despite these assumptions, SAR Table 2.2-8 indicates that near- and far-field criticality is a negligible contributor to the overall probability of criticality [the cutoff for including probabilities in this table is two orders of magnitude lower than the probability limit, or 10^{-8} per year, in 10 CFR 63.342(a)]. Therefore, the NRC staff did not perform a detailed review of the near- and far-field criticality FEPs."

The assumed enrichment for the CSNF models is similar to that for ABD glass (i.e., 5 wt% ^{235}U). The total uranium mass in a CSNF waste package is significantly (over two orders of magnitude) higher than the total uranium mass in an ABD glass canister. Therefore, codisposal of any DSNF type with ABD glass is not expected to affect the argument used to exclude the eight external criticality FEPs on the basis of low probability.

Table 3-17. Summary of External Criticality

Initiating Event	Waste Package Type	Calculated Accumulation or MassReleased from Waste Package	Mass of U or Pu (for FFTF) in kg Required To Achieve Critical Limit of k_{eff} = 0.96			
		Uranium Mass, unlessotherwise noted (kg)	Invert	Fractured Tuff	Lithophysae Array	Large Lithophysae
Seismic	DOE3 (N Reactor)	Not calc[a]	266,000	Inf[b]	Not calc	Not calc
	DOE9 (TMI-2 Fuel)	Not calc	350	Inf	Not calc	Not calc
	CSNF	90.3	126	Inf	Similar to igneous	Not calc
	DOE1 (FFTF) (Plutonium mass)	0	1.66	4.3	4.0	2.2
Igneous	DOE3 (N Reactor)	0.109	Inf	Inf	Not calc	Inf
	DOE9 (TMI-2 Fuel)	9.24	538	Inf	Not calc	Inf
	CSNF	74.8	159	Inf	1,390	Inf
	DOE1 (FFTF) (Plutonium mass)	2.49×10^{-2}	1.66	4.3	4.0	2.2

NOTE: [a]"Not calc" means that this waste form was bounded by another waste from and or configuration. In most cases, this simply meant that, if CSNF waste was very subcritical, then TMI and N Reactor had to be also.

[b]"Inf" means that a criticality event for the analyzed waste form in this geometry is not possible even if an infinite mass were released.

CSNF = commercial spent nuclear fuel

FFTF = Fast Flux Test Facility

TMI-2 = Three Mile Island Unit 2

3.3.2 Noncriticality FEPs and Nominal Scenario Models and Analyses

Using the approach described in Section 2.2.2, the study team identified 27 noncriticality FEPs that would likely be affected by implementing the ABD Program; a list of these FEPs is given in Appendix A. The models and analyses that are associated with these FEPs and might be affected by implementing the ABD Program are discussed below. The models and analyses discussed in this subsection reflect the nominal scenario class. Modeling cases under the other scenario classes (i.e., early failure, igneous, seismic, and human intrusion) are discussed in Section 3.3.3.

Early in the evaluation process, FEPs related to the glass degradation model (BSC 2004c) and the multiscale thermal-hydrologic model (SNL 2008b) were identified as having the potential to be affected by implementing the ABD Program. However, subsequent investigation indicated that the glass expected to be produced in the ABD Program is similar to the glass analyzed in the Yucca Mountain technical baseline with respect to degradation characteristics and will not change the technical basis for the glass degradation model. This investigation also indicated that the change in the thermal load due to implementing the ABD Program is within the range of uncertainty and variability already considered in the multiscale thermal-hydrologic model and related thermal management analyses. The results of evaluating the glass degradation model and the thermal-only aspects of the multiscale thermal-hydrologic model are briefly described in the next few paragraphs and explained in more detail in Appendix B.

The DHLW glass degradation model developed for use in repository performance simulations used to support the Yucca Mountain license application is based on an understanding of the key processes leading to glass dissolution and the release of radionuclides (BSC 2004c). These processes include water diffusion, ion exchange, and hydrolysis reactions, with the affinity-controlled hydrolysis of silica bonds limiting the dissolution rate. The rate equation includes explicit terms for the effects of the temperature and solution pH on the dissolution rate, whereas the effects of the glass and solution compositions are bounded by using limiting coefficient values. The effects of other processes such as radiolysis and interactions with microbes on glass degradation occur primarily through changes to the pH of the seepage water and are taken into account by the ranges of those parameter values. Separate coefficient values were determined to represent minimum and maximum rates that could be attained under acidic and alkaline conditions from the results of laboratory tests conducted with representative glasses having a range of compositions. The maximum rates predicted to occur in alkaline solutions are based on responses measured in American Society for Testing and Materials (ASTM) C1285 product consistency Method A tests (PCT-A). These represent the most likely conditions to occur during the regulated isolation period because alkaline seepage waters are expected to be generated in breached waste packages due to the preferential release of alkali metals from borosilicate waste glasses when initially contacted by water. The results of PCT-A conducted with surrogate ABD glass (Crawford et al. 2021) are within the range of values used to parameterize the current HLW glass degradation model, particularly the pH values and boron concentrations. Furthermore, preparation of the surrogate ABD glass for use in PCT-A by crushing indicates the fracture behavior is the same as that of glasses used to parameterize the HLW glass degradation model, so the accessible surface area of ABD glass waste forms will be well represented by the surface area term used in the current HLW glass degradation model. The correspondence is discussed in detail in Appendix B-1. Therefore, the degradation rates of ABD glass are expected to be within the uncertainty range of the model used to support the license application.

The multiscale thermal-hydrologic model plays an important role in providing thermal-hydrologic parameters to other process models and to the TSPA. The evaluation of potential effects to the thermal-only aspects of the model assumes that ABD glass processing would result in about 900 fewer CDSP short waste packages (Section 3.1.1). Reducing the quantity of CDSP waste packages would affect thermal conditions and repository footprint. This situation could in turn have consequences on downstream processes. An analysis of design factors affecting thermal hydrology in a repository is described in SNL (2008b, Section 6.1.5), which gives the parameters that constrain the distance between drifts and the size of the required repository footprint. For the ABD Program evaluation, simplified thermal calculations were performed to quantify the effect of reduced heat and/or reduced repository footprint using the parameters of importance. Details of the calculations are given in Appendix B-2. The results of the calculations are summarized here.

The thermal effects due to changes to the line-averaged thermal load are within the design thermal constraints (Section 2.2.1). The thermal effects due to changes to the areal power density and the areal mass loading are also within the design thermal constraints. Reducing the number of CDSP waste packages would affect the makeup of the waste packages represented in the limiting waste string (i.e., the seven-waste package sequence). Table B-13 shows a possible alternative waste package sequence reflecting the change in relative amounts of the CDSP waste packages while maintaining the same line-averaged thermal load.

The simplified calculations described above were developed assuming thermal conduction was the only mechanism for heat removal, including thermal-hydrologic effects leading to lower temperatures. A study of the range of acceptable design thermal loadings examined the effect of thermal hydrology on the ability to achieve design thermal limits (SNL 2008c, Section 6.2). The study evaluated the following cases: (1) heat dissipation effects at the edge of the repository layout, and (2) the hydrological effects of percolation flux. These results show that there is substantial margin to meet thermal limits when these cases are considered. This outcome provides additional confidence that the minor differences in the thermal aspects of the multiscale thermal-hydrologic model due to implementing the ABD Program would not affect the technical basis for licensing a Yucca Mountain-similar repository.

While the design thermal loadings study points to the effects of considering both thermal and hydrologic heat removal processes in thermal modeling, it does not provide information about how the thermal-hydrologic aspects of the multiscale thermal-hydrologic model or other models with a thermal-hydrologic component may be affected by implementing the ABD Program. That subject is addressed in the subsections below.

The remainder of Section 3.3.2 discusses nominal scenario models and analyses in the Yucca Mountain technical baseline with aspects that would likely be affected by the ABD Program in ways that are difficult to characterize fully without additional study beyond the scope of this evaluation. In particular, the subsections below address the multiscale thermal-hydrologic model, seepage models, in-drift convection and condensation models, inventory models, and in-package chemistry and radionuclide models. Note that the postclosure evaluation results presented below generally assume there will be about 900 fewer CDSP waste packages, which is consistent with Design Examples 1 and 2 presented in Section 3.1.2. The exception is when discussing Design Example 3, which assumes the number of CDSP waste packages remains the same.

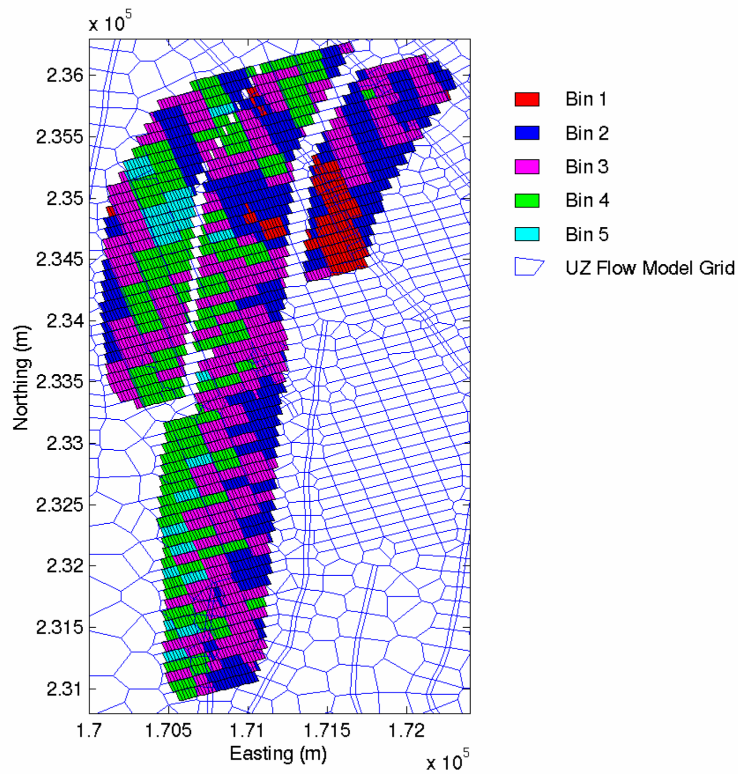
3.3.2.1 Multiscale Thermal-Hydrologic Model

The multiscale thermal-hydrologic model (SNL 2008b) provides in-drift thermal-hydrologic conditions, particularly temperature and relative humidity, to other process models and the TSPA. The ABD glass-related issue with the most significant effect on the multiscale thermal-hydrologic model is the removal of about 900 CDSP waste packages, which reduces the ratio of the number of cooler CDSP waste packages to the number of hotter CSNF and naval SNF waste packages from ~0.4 to ~0.3 (Section 3.1.1). This change could cause differences in the subsurface design of a Yucca Mountain-similar repository such as differences in waste package spacing, drift spacing, and/or number of drifts (Section 3.1.2). Since the multiscale thermal-hydrologic model uses grid patterns that reflect the subsurface design, differences in the subsurface design could affect model gridding. The removal of about 900 CDSP waste packages also affects the abstraction process used to prepare results from the multiscale thermal-hydrologic model for use by other process models and the TSPA. This subsection presents a description of the multiscale thermal-hydrologic model followed by a discussion of the potential effects of implementing the ABD Program on the model gridding and the abstraction process.

Note that multiple coupled process models with a thermal-hydrologic component exist in the Yucca Mountain technical baseline besides the multiscale thermal-hydrologic model. The suite of mountain-scale coupled process models (BSC 2005b) is not discussed further because the effects are limited primarily to the same model gridding issue that is examined below for the multiscale thermal-hydrologic model. Other potentially affected drift-scale models with a thermal-hydrologic component are addressed in other subsections as appropriate. For example, Section 3.3.2.2 discusses seepage models; potentially affected seepage models with a thermal-hydrologic component are addressed in that subsection.

The multiscale thermal-hydrologic model uses four submodels solved at different spatial scales and dimensionality. Final model results reflect a three-dimensional (3D) system in which the repository footprint is subdivided into 3,264 subdomains of equal area (SNL 2008b, Section 6.2.12.1[a]). The multiscale thermal-hydrologic model accounts for waste-package specific heat output with a seven-waste package sequence containing six full packages and two half packages at either end of a repeating unit cell used to fill the drifts with waste packages. Two of the full packages are CDSP waste packages; the other packages—both full and half—are CSNF waste packages. Parametric uncertainty is incorporated through seven host-rock thermal conductivity–percolation flux uncertainty cases. For each of the uncertainty cases, the multiscale thermal-hydrologic model calculates time-dependent thermal-hydrologic variables: temperature and relative humidity for the full and half packages in the unit cell along with the respective drip shields at each of the subdomain locations. In addition, the multiscale thermal-hydrologic model calculates the time-dependent values for the average drift wall temperature, duration of boiling at the drift wall, invert temperature, invert saturation, and invert liquid flux at each of the subdomains.

The abstraction process for the multiscale thermal-hydrologic model final results has two parts. The first part involves assigning the 3,264 subdomains to one of five percolation subregions or bins that are defined based on the percolation-flux distribution at the base of a particular geologic unit for a given infiltration scenario and climate regime as determined from the unsaturated zone flow model (SNL 2007m). The percolation flux values for each subdomain location are sorted and grouped into bins according to the following cumulative probability intervals: 0.0−0.05, 0.05−0.30, 0.30−0.70, 0.70−0.95, and 0.95−1.00. Figure 3-7 shows the correlation of percolation bins with the 3,264 subdomain locations. As can be seen, subdomains belonging to any particular bin are generally scattered throughout the repository. Even within a single drift, multiple bins are represented.



NOTE: The grid for unsaturated zone flow model (SNL 2007m) is shown by the blue mesh. The grid for the multiscale thermal-hydrologic model submodel processing the percolation fluxes was designed to match the unsaturated zone flow model grid.

UZ = unsaturated zone

Source: SNL 2008b, Figure VIII-1[a].

Figure 3-7. Correlation of the Five Percolation Bins with the Multiscale Thermal-Hydrologic Model Subdomain Locations in the Repository Footprint

The second part of the abstraction process involves determining a representative CDSP waste package and a representative CSNF waste package for each of the five percolation bins (SNL 2008b, Appendix VIII[a]). For example, consider the population of CDSP waste packages for any given bin. The multiscale thermal-hydrologic model final results include the peak waste package temperature and boiling duration at the waste package for each of those CDSP waste packages. The representative CDSP waste package for the bin is the CDSP waste package having values for those two parameters with smallest combined deviation from the median values for the bin.

This abstraction approach is used for the multiscale thermal-hydrologic model results from each of the seven cases representing the combined thermal conductivity–percolation flux uncertainty (DOE 2008, Section 2.3.5.4.1.4). Within the TSPA, the abstracted multiscale thermal-hydrologic model results are implemented by the engineered barrier system thermal-hydrologic environment submodel, which then provides the abstracted results to six other TSPA components and submodels.

The potential differences in the multiscale thermal-hydrologic model abstracted results due to a reduction of about 900 CDSP waste packages are difficult to characterize fully with the available information. For a Yucca Mountain-similar repository, there would be multiple design options possible to deal with the CDSP waste package reduction (Section 3.1.2) including changing the waste package spacing, drift length, drift spacing, number of drifts, CDSP waste package loading configuration, etc. The net effect on abstracted model results would change depending on the combination of design options selected. For illustration purposes, consider the three design examples with variations from the Yucca Mountain design presented in Table 3-4: Design Example 1 (same waste package spacing, decreased number of drifts), Design Example 2 (increased waste package spacing, same number of drifts), and Design Example 3 (same waste package spacing and number of drifts implying different CDSP loading configuration).

In Design Example 1, the decrease in the number of drifts could lead to a new model gridding for the multiscale thermal-hydrologic model as well as the unsaturated zone flow model, which uses a matching grid to facilitate transfer of the percolation-flux distribution. The change in repository layout and gridding could lead to some differences in the percolation fluxes assigned to multiscale thermal-hydrologic model subdomain locations. The dimensions of the subdomains could change. Even if the subdomain dimensions remain the same, fewer drifts mean fewer subdomain locations, leading to some locations not previously on the repository boundary becoming subject to edge effects in the thermal-hydrologic calculations. The abstraction process is affected from the start with the assignment of bins. The subdomain population assigned to each percolation bin would change depending on which subdomains remain as well as any changes to the percolation fluxes for the remaining subdomains given the new layout and gridding. The determination of representative CDSP and CSNF waste packages for each bin could change because of differences not only in the population of waste packages for each type (CDSP versus CSNF) but also in the associated multiscale thermal-hydrologic model results upon which the determination of representative waste packages is based.

Design Example 2 considers increasing the waste package spacing to accommodate the reduction in the number of CDSP waste packages rather than decreasing the number of drifts. As such, the model gridding would not be affected, which means the percolation binning of the subdomains would not be affected. The magnitude of the change in waste package spacing and distribution of the remaining waste packages would depend on the specifics of the implementation of Design Example 2. Regardless of the implementation, increased waste package spacing could lead to minor differences in the multiscale thermal-hydrologic model results for each waste package. Appendix B-2.4.2 explores the effect of increased waste package spacing on thermal calculations. In addition, increased waste package spacing could change the population of waste packages assigned to a particular subdomain. As a result, the determination of the representative CDSP and CSNF waste packages could change.

Design Example 3 is simpler in some ways because it assumes the number of waste packages, the waste package spacing, and the repository layout stay the same, which means the percolation binning would not be changed. The key change introduced with Design Example 3 is the loading configuration of the CDSP waste packages. In the TSPA, all CDSP waste packages have the same inventory. While it might be possible to design a new loading plan resulting in a new inventory that is the same for all CDSP waste packages, that new inventory would be different than the CDSP waste package inventory in the TSPA. There are also other possibilities for the implementation of Design Example 3. For instance, the new loading plan could involve having a portion of the CDSP waste packages fully loaded—the inventory would be the same as the TSPA CDSP waste package inventory—and having a reduced inventory for the remainder of the CDSP waste packages. The new loading plan could have thermal effects as well stemming from a change to the established thermal line load, the seven-waste package sequence, or both

(Appendix B-2). Any loading plan different from that analyzed in the TSPA could change the multiscale thermal-hydrologic model results, which could change the determination of the representative CDSP and CSNF waste packages.

As seen with these design examples, the effects of implementing the ABD Program on the multiscale thermal-hydrologic model abstracted results vary according to the specific design changes proposed to accommodate the reduction of about 900 CDSP waste packages. In addition, the complexity of the multiscale thermal-hydrologic model makes it difficult to predict the net model response to a new subsurface design. Interactions between multiple affected aspects of the model can produce unexpected results that can be found only by running the model with the reduced inventory, something that is outside the scope of this evaluation. Nonetheless, changing the multiscale thermal-hydrologic model as described above to incorporate the effects of implementing the ABD Program would not be expected to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements. A discussion of this overall conclusion along with the semiquantitative basis used for its support is presented in Appendix C.

3.3.2.2 Seepage Models

There are multiple ambient and coupled processes seepage models designed to investigate various aspects of seepage. The discussion below centers on the ambient seepage model for PA, the drift seepage abstraction, and the TSPA drift seepage submodel. The seepage model for PA generates the ambient seepage lookup tables that are part of the drift seepage abstraction used by the TSPA drift seepage submodel.

Other seepage models are not discussed because they are not likely to be significantly affected by implementing the ABD Program. The seepage calibration model and the flow focusing model provide seepage-related parameters that depend on the host rock rather than drift conditions. The thermal-hydrologic seepage model provides a qualitative indication of thermal seepage, but the results are not included in the TSPA. The thermal-hydrologic-mechanical and thermal-hydrologic-chemical seepage models (BSC 2004d; SNL 2007d) are used to conduct studies of possible changes to rock properties due to thermal-hydrologic-mechanical and thermal-hydrologic-chemical processes. The studies concluded that these coupled processes and property changes have no significant impact on seepage. Given the similarity between the thermal and chemical characteristics of the ABD glass and the existing SRS glass, the study conclusion supporting the FEP exclusion justifications is not likely to be changed by implementing the ABD Program. Another role of the thermal-hydrologic-chemical seepage model is to corroborate the near-field chemistry model, which provides seepage water compositions to TSPA. However, given that seepage is assumed to be zero during the thermal period, water compositions predicted for the thermal period by any model are not used in the TSPA. Implementing the ABD Program would not change this situation.

Some of the seepage properties calculated as part of seepage modeling could potentially be changed by implementing the ABD Program. Similar to the situation with the multiscale thermal-hydrologic model, the potential differences in the seepage calculations depend on the specific design changes proposed to accommodate the reduction of about 900 CDSP waste packages. To facilitate the discussion, the following definitions are provided (SNL 2007a, Section 6.1.3):

- **Drift Segment Length—**The length of one waste package (5 m) plus the space between waste packages (0.1 m) for a total of 5.1 m.
- **Average Seepage Rate—**The amount of water seeping into the drift opening per unit of time. Note that any calculated value below the critical seepage threshold is considered to have no seepage.
- **Seepage Percentage—**The seepage rate divided by the percolation rate across the reference area (drift width × drift segment length) multiplied by 100 to turn it into a percentage.
- **Seepage Fraction—**The fraction of waste packages, by type, in a percolation bin that experience seepage in a given realization.

The seepage calculations rely on two inputs—the percolation flux distribution and the duration of boiling at the drift wall—from the multiscale thermal-hydrologic model calculations (Section 3.3.2.1). The drift seepage abstraction directs seepage to be set at zero during the thermal period. When the drift wall temperature drops below 100°C, the ambient seepage lookup tables are used to determine seepage rates. The lookup tables provide seepage rates as a function of percolation flux, capillary strength, and fracture permeability. The ambient seepage rate at a particular location is determined based on the percolation flux provided by the multiscale thermal-hydrologic model. The seepage percentage reflects how much of the percolation flux enters the drift as seepage. The seepage fraction for any given percolation bin indicates the portion of waste packages in locations within the bin that experience seepage. There are two aspects to the calculation: (1) determining the set of waste package locations for the bin, which depends on the population of subdomain locations assigned to the bin as described in Section 3.3.2.1, and (2) determining the number of those waste package locations with a seepage rate above the critical seepage threshold.

Design Example 1 with its decrease in the number of drifts would affect the model gridding and possibly the values for percolation flux for each element of the new gridding, the boiling duration (i.e., start of ambient seepage), and the population of subdomain locations for the percolation bins from the multiscale thermal-hydrologic model (Section 3.3.2.1). As a result, the average seepage rate, the seepage percentage, and the seepage fraction could be affected.

Design Example 2 increases the waste package spacing, which could affect some of the multiscale thermal-hydrologic model results used in seepage calculations. As discussed in Section 3.3.2.1, a change in waste package spacing would not affect the percolation rates associated with the subdomain locations, meaning that assignment of subdomains into percolation bins would not be affected. However, the determination of the representative CDSP and CSNF waste packages could change, causing differences in the associated results per waste package, including the boiling duration at the waste package. This parameter controls when the thermal period is over and ambient seepage can be turned on.

Besides the downstream effects caused by differences in the multiscale thermal-hydrologic model results, increasing the waste package spacing also has a direct effect on ambient seepage rate calculations in that the drift segment length would increase. The average seepage rate is calculated for the drift opening over the waste package (i.e., drift width × drift segment length where the drift segment length = waste package length + waste package spacing). Therefore, increasing the waste package length has the same effect on seepage rate as increasing the waste package spacing. A study considered the effect of increasing the waste package length by 10% on ambient seepage rate calculations in the TSPA (SNL 2010a, Appendix P13[a]). The study found that a 10% increase in waste package length causes a 10% increase in average seepage rate. For potential effects on dose, this study pointed to another study involving increased seepage flux after drift collapse in lithophysal zones (SNL 2010a, Appendix P17). The results of this second study indicated that the change in predicted annual dose is roughly proportional to the change in the seepage rate for cases in which the major dose-controlling radionuclide is solubility controlled in the engineered barrier system. For radionuclides not subject to solubility limits, the increased seepage rate would have a negligible effect. On the basis of the second study, the first study concluded that the overall effect on dose of increasing the waste package length by 10% was minor. This conclusion can be extended to apply to the effects of increasing waste package spacing as well.

In addition, the seepage percentage could be affected because the calculation considers the seepage rate and percolation rate across the reference area defined by the drift width × drift segment length. Again, increasing the waste package spacing increases the drift segment length.

The seepage fraction could also be affected because the increased seepage rate due to increased waste package spacing could cause some locations with seepage previously below the threshold to change to having seepage above the threshold. To evaluate the potential effect on total dose associated with an increase in seepage fraction due to increasing the waste package length by 10%, a study of results from the 1,000,000-year nominal modeling case, the 10,000-year seismic ground motion modeling case, and the 1,000,000-year seismic ground motion modeling case was conducted to determine the degree of correlation between seepage fraction and expected annual dose (SNL 2010a, Appendix P17). The 10,000-year igneous intrusion modeling case and the 1,000,000-year igneous intrusion modeling case, which are also major contributors to dose, were not examined because after an igneous event occurs, all waste packages are evaluated in a seeping environment. The study found that the rank correlation between seepage fraction and dose for the three cases was very low supporting the conclusion that, even if the seepage fraction were to increase, the effect on dose would be negligible.

Design Example 3 assumes that the repository layout and waste package spacing are the same, but the CDSP loading plan is different. In this case, the modeling grid is not affected nor are the percolation flux or percolation binning affected. The different loading plan could cause differences in the duration of boiling leading to some locations starting ambient seepage sooner. The locations that could experience changes would depend on the details of the new loading plan.

Seepage rates, seepage percentage, and seepage fraction could be changed in different ways by implementing the ABD Program depending on the specific choices made to account for the decrease in the number of CDSP waste packages. As mentioned above, one study (SNL 2010a, Appendix P17) suggested that, even if there is an increase in seepage rates, the resulting increase in modeled dose would be minor. In addition, the potential dose effects of an increase in seepage rates would likely be offset to some unknown extent by the decrease in the number of waste packages disposed of. On balance, changes made to the seepage models to accommodate implementation of the ABD Program would not be expected to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all regulatory requirements (Appendix C).

3.3.2.3 In-Drift Convection and Condensation Models

As the name implies, the in-drift natural convection model was used to analyze the convection cells that develop in the drifts due to thermally driven natural convection (SNL 2007h). Similarly, the in-drift condensation model was used to simulate the condensation due to water vapor transport from hotter to cooler parts of the drifts. Although both models are process-level models, only the in-drift condensation model provides output that is abstracted for use in the TSPA. The in-drift natural convection model provides inputs for use by other process-level models.

The in-drift natural convection model accounts for the open spaces in the drift as well as components such as the invert, waste package, drip shield, and drift wall (SNL 2007h, Sections 6.1 and 6.1[a]). For computational efficiency, two-dimensional (2D) simulations were used to conduct sensitivity studies to guide selection of appropriate submodels for the 3D simulations. The 2D simulations were also used to generate correlations of equivalent thermal conductivity for use in porous media models (SNL 2007h, Section 6.4). The in-drift natural convection model and porous media models represent the in-drift configuration in very different ways. For the 2D simulations, the in-drift natural convection model considers a half-drift cross-sectional area extending from the drift center to the drift wall with explicit representation of whether the space is open or occupied by the invert, waste package, or drip shield. In contrast, porous media models such as the multiscale thermal-hydrologic model cannot explicitly represent the combination of open and occupied spaces within the drift. Instead, the drift and everything in it is modeled as a porous medium. The correlations of equivalent thermal conductivity provided by the 2D simulations are applied to the porous medium to approximate the effects of turbulent natural convection heat transfer within the drift.

As mentioned above, the domain of the 2D convection simulations is based on half-drift cross-sectional area extending from the center of a waste package to the drift wall, so potential changes driven by reducing the number of CDSP waste packages such as a change to the number of drifts (Design Example 1) or the waste package spacing (Design Example 2) have no effect on the results. In addition, since the heat source is for the single waste package, waste package-to-waste package variability in heat generation is not a factor for the 2D simulations. The heat output used is consistent with the 1.45 kW line load, so there would be a difference in results if, as is possible for Design Example 3, the line load is reduced prompting a reduction in the heat output of the single waste package. Of course, such a change would be in a favorable direction in terms of meeting the various thermal design limits.

The 3D simulations (SNL 2007h, Sections 6.2 and 6.1[a]) were used to provide the in-drift condensation model with axial dispersion coefficients. The 3D simulations are independent of the number of drifts since the domain is a single partial drift (71 m long) filled with two of the seven-waste package sequences. A change in the line load and/or the waste package sequence would affect the heat source input and potentially the length of the drift selected for analysis, which would cause differences in the model output. A change in waste package spacing would also cause differences.

The in-drift condensation model (SNL 2007h, Sections 6.3, 6.1[a], and 6.2[a]) uses single-node representations of each waste package along seven selected drifts, as well as separate nodes for the drip shield, invert, and drift wall at each waste package location. The model simulations considered a number of discrete calculation times and found that only the 1,000-year calculation time for the low-invert transport cases resulted in condensation (DOE 2008, Section 2.3.5.4.2.4). The abstracted model results are a series of linear regressions (against percolation flux) reflecting four dispersion coefficient–drip shield ventilation cases, each considered equally likely to occur. These regressions provide the probability of occurrence of condensation for a waste package location and the condensation rate if condensation occurs. The TSPA drift wall condensation submodel selected one case per realization and applied it to the time period between 1,000 and 2,000 years.

The dispersion coefficients from the 3D simulations of the in-drift natural convection model are used by the in-drift condensation model. As a result, any differences in the dispersion coefficients due to implementing the ABD Program would cause downstream differences in the condensation results. Similarly, any differences in the percolation rates caused by potential gridding changes in the unsaturated-zone flow model calculations (Section 3.3.2.1) would cause downstream differences in the condensation results.

The direct ABD Program-related effects on the in-drift condensation model depend on specific design modifications chosen to accommodate the reduction in the number of CDSP waste packages. The in-drift condensation model considers seven specific drifts selected across the repository (SNL 2007h, Figure 6-2[a]). A change in the number of drifts (Design Example 1) could cause different drifts to be selected. In Design Example 2, a change in the waste package spacing, which is lumped into the waste package length for convenience, could cause a difference in model results. The drift wall temperatures are approximated through simplified calculations using the line-load heat source, so they would be affected only if the line load is changed as is possible for Design Example 3. For other parts of the condensation analysis, the seven-waste package sequence is used to incorporate waste package-to-waste package variability in heat generation. A change in the sequence could cause differences in the model results.

The TSPA drift wall condensation submodel uses linear regressions abstracted from the in-drift condensation model to evaluate the occurrence and magnitude of drift wall condensation for the typical waste packages (i.e., the typical CDSP and CSNF waste packages) within each percolation bin. The typical CDSP and CSNF waste packages are treated equally in terms of how the in-drift condensation abstraction is applied. The potential ABD Program-related effects on the percolation binning and selection of typical CDSP and CSNF waste packages are discussed in Section 3.3.2.1.

The effects of implementing the ABD Program on condensation results are likely to be minor given that the water volume contributed by condensation is small compared to the water volume contributed by seepage. First, in the TSPA, condensation can occur only for the time period between 1,000 to 2,000 years while seepage can occur from the time the thermal period ends to the end of 1,000,000-year simulation period. Second, even during the 1,000 years when condensation can occur, the condensation volume is limited. The cases with higher condensation rates (10 to 45 kg/[m·yr]) tend to have probabilities of occurrence of only about 1% to 2% (SNL 2007h, Appendices A.1[a] and B.1[a]). When the probability of occurrence for condensation is at its highest, it is still only about 20%, and then the condensation rate is less than 1.5 to 2 kg/(m·yr) depending on the case selected. A comparison can be made to example seepage calculations (SNL 2007a, Section 6.4[a]) reflecting different infiltration scenarios. During the monsoon climate state, which corresponds to the time frame condensation can occur, the calculated seepage ranged from 4.6 to 470.8 kg/yr per waste package. In this instance, "per waste package" refers to the drift segment length of 5.1 m. The seepage range expressed in the same units as condensation is 0.9 to 92.3 kg/(m·yr).

Although differences in the in-drift condensation model results can occur because of implementing the ABD Program, these differences are not expected to alter the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C).

3.3.2.4 Inventory Models

The radionuclide inventory used in the TSPA was developed through a process based on the best information available for a wide variety of wastes. An initial radionuclide screening analysis was conducted to identify radionuclides that could be important to dose, for both the 10,000-year and 1,000,000-year time frames. The analysis continued by developing a representative radionuclide inventory for CDSP and CSNF waste packages. The process focused on developing a conservative estimate and resulted in total numbers of CDSP and CSNF waste packages, as well as radionuclide inventories for CDSP and CSNF waste packages given in grams of radionuclide per waste package for each waste package type (SNL 2010a, Table 6.3.7-3).

To account for uncertainty and variability, a radionuclide inventory uncertainty multiplier was developed for each waste type, i.e., CSNF, DSNF, and HLW glass. The uncertainty multiplier for the DSNF radionuclide inventory (except for ^{238}U) has a triangular distribution with a minimum of 0.45, a most likely value of 0.62, and a maximum of 2.9. The uncertainty multiplier for HLW glass has a triangular distribution with a minimum of 0.7, a most likely value of 1, and a maximum value of 1.5 (SNL 2010a, Table 6.3.7-7). The nominal inventory of ^{235}U in DSNF is 25,100 g/waste package and in HLW is 1,410 g/waste package (SNL 2010a, Table 6.3.7-3). The total number of CDSP waste packages is 3,416 (SNL 2010a, Table 6.3.7-3).

Implementing the ABD Program will affect the inventory model in several ways. First, it will remove about 30 MTHM of certain types of SNF from the inventory of DSNF to be disposed of (Table 3-3), which is likely to be replaced by 30 MTHM of a different kind of DSNF. However, this 30 MTHM represents only 1.3% of the 2,268 MTHM of DSNF designated for disposal in the repository and thus will have a limited effect on the model of the average DSNF radionuclide inventory, given the uncertainties in that model outlined above.

Second, implementing the ABD Program will decrease the number of CDSP waste packages to be disposed of by about 900 (Section 3.1.1), from 3,416 to 2,516, a reduction of ~26%. This reduction clearly affects the inventory model in terms of total number of waste packages and total radionuclides disposed of.

Third, implementing the ABD Program could change estimates of the nominal inventory of radionuclides per CDSP waste package. Changes associated with the ABD program include bypassing the uranium extraction processes during dissolution of the fuel (SRNL 2020). As a result, the ABD glass will have a larger abundance of uranium isotopes, including ^{235}U and ^{238}U, the latter of which will be added to ensure the fissile content of the glass remains below enrichment limits. Because each CDSP can hold five canisters of HLW glass, decreasing the number of CDSP waste packages to be disposed by about 900 decreases the number of HLW glass canisters to be disposed of by about 4,500. In creating estimates of the grams of each radionuclide per waste package, a certain mix of HLW glass canisters from different sources was considered. Removing 4,500 HLW glass canisters from that mix would change those estimates and how they are created. In addition, increasing the fissile material limit, to accommodate the increased uranium abundance, from 0.897 kg/L to 2.5 kg/L will also change the estimate of grams of fissile material in each waste package. However, given the significant uncertainty bounds already included in the inventory model, revised estimates of grams of each radionuclide per waste package are likely to be either within those bounds or not far from them. Finally, the radionuclide inventory model would be affected by changes to the repository layout or the in-drift configuration only if Design Example 3 were implemented. In this design example, the number of waste packages remains the same, but the amount of HLW glass and/or DSNF in each CDSP waste package is decreased. In this case, the "representative" CDSP waste package inventory (SNL 2010a, Table 6.3.7-3) would no longer be applicable. The exact effects of implementing Design Example 3 on the radionuclide inventory model would depend on how any waste package loading change was implemented.

Implementing the ABD Program will decrease the number of CDSP waste packages to be disposed of by about 900, which is the probably the most significant change to the inventory model. Implementing the ABD Program could lead to changes in the estimates of grams of each radionuclide per CDSP waste package, but these changes are not likely to be as significant, given the uncertainties already included in those estimates. In any case, the anticipated changes are not expected to alter the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C).

3.3.2.5 In-Package Chemistry and Radionuclide Solubility Models

The in-package chemistry process model was used in the Yucca Mountain TSPA model to determine the ranges of pH, ionic strength, and fluoride concentration in water inside a TSPA model HLW glass waste form cell as a function of local conditions (SNL 2007i, Section 8.2[a]). These parameter range constraints were used in the TSPA model to limit the values of pH, ionic strength, and fluoride sampled by the TSPA model. The values sampled from these ranges, in turn, were used by the TSPA model to calculate radionuclide solubilities. This subsection addresses the extent to which implementing the ABD Program would affect these calculations.

One of the multiple processes the in-package chemistry process model simulates is the degradation of HLW glass over time. Under certain conditions glass degradation can affect the calculated pH, ionic strength, and fluoride concentration in the water in the HLW glass waste form model cell. However, because the in-package chemistry model abstractions simply establish the range constraints (and distributions) of pH, ionic strength, and fluoride concentration and not the expected values, the specific glass composition is generally expected to have limited effects on the in-package chemistry model abstractions.

The ranges of pH were determined by running the in-package chemistry process model, titrating excess acid and base into the cells, and identifying the pH values for which excess acid and base neutralizing capacity occur (SNL 2007i, Section 8.2.1[a]). The lower end of the pH range was found to be limited by the substantial buffer capacity of the Ni/Fe oxides that accumulate as steel degrades. This lower-end pH buffer capacity is common to all waste packages containing steel. The constraint on the upper end of the pH range depends on the waste form. For HLW glass, the upper pH limit is constrained by the dissolution of carbon dioxide from the gas phase. Because the sources of acid and base neutralizing capacity in the HLW glass cell are not altered by implementing the ABD Program, the in-package chemistry abstraction model pH ranges used by the TSPA model are not affected by potential ABD activities.

The ranges of ionic strength in the in-package chemistry abstraction model were determined using a sensitivity analysis. The sensitivity analysis showed that liquid influx rate, time, and relative humidity have important effects on ionic strength. Less important are the degradation rates of the waste form and waste package. The sensitivity analysis did not examine the effects on ionic strength of potential changes to the glass composition.

Two in-package chemistry models were used to determine ionic strength ranges for the TSPA model: a liquid influx model and a vapor influx model. The liquid influx model used a set of in-package chemistry process model simulations to determine the range of ionic strength over time under various conditions. That model was applied to relative humidity above 98%. Below a relative humidity of 98%, the vapor influx model is used. The vapor influx model used literature data and Pitzer calculations of simple salt systems to develop a relationship between ionic strength and relative humidity for relative humidity below 98%. The prediction of ionic strength under vapor influx conditions is not affected by glass composition because the vapor influx model does not use the glass as input for ionic strength prediction.

For the liquid influx model, implementing the ABD Program could potentially affect the ionic strength range used in the TSPA model. Ionic strength is important in the TSPA model for constraining colloid stability and the ranges of pH and fluoride concentration. The ranges of pH and fluoride concentration, in turn, affect the ranges of radionuclide solubility. The glass composition cannot affect the lower end of the ionic strength range because the lower limit is controlled by the ionic strength of the groundwater. However, it could potentially affect the upper limit when the liquid influx rate is low (SNL 2007i, Figure 6-45[a]). The composition of Ca and Mg in the ABD glass samples, as indicated in Table 5-3 of Crawford et al. (2021), differs markedly from the glass modeled in the in-package chemistry process model simulations. While the Na_2O content is approximately the same as that used in the process model simulations (11%–13%), the MgO content of the ABD glass samples is about ten times lower (~0.2% compared to ~2.2%) and the CaO content is about half (~0.65% compared to ~1.3%) (SNL 2007i, Table 4-6[a]). Because chemical divides can potentially affect the salting out of divalent ions and because the valences of the dominant ions in groundwater have an exponential effect on ionic strength, these changes in the glass composition can potentially affect the upper limit of ionic strength in the in-package chemistry model abstractions. Uncertainty in ionic strength is included in the TSPA (SNL 2007i, Section 8.2.2[a]). The higher ionic strengths that might be associated with the HLW glass produced by the ABD Program may already be accounted for in the TSPA, but a chemical analysis would be needed to ensure that ionic strengths associated with ABD glass do not exceed the uncertainty bounds already considered.

There is also a potential for the fluoride concentration ranges of the in-package chemistry abstraction model to be affected by the HLW glass composition that results from implementing the ABD Program. Fluoride concentration is highly sensitive to concentrations of Ca and Mg. The maximum fluoride concentration calculated by the in-package chemistry process model for the HLW glass cell for the Yucca Mountain radionuclide solubility calculations was limited by the precipitation of fluorite (CaF_2) and sellaite (MgF_2) (SNL 2007i, Section 6.10.3[a]). In comparison, waste form cells containing SNF had low Ca and Mg concentrations, allowing fluoride to concentrate as ionic strength increased. High fluoride concentrations increase radionuclide solubility. With lower concentrations of Ca and Mg in the ABD glass, maximum fluoride concentrations calculated by the in-package chemistry process model for the HLW glass cell could be much higher, leading to higher radionuclide solubilities. Uncertainty in radionuclide solubilities resulting from higher fluoride concentrations is accounted for in the TSPA (SNL 2010a, Section 6.3.7.5.2). The higher radionuclide solubilities that might be associated with the HLW glass produced by the ABD Program may already be accounted for in the TSPA, but a chemical analysis would be needed to ensure that anticipated fluoride concentrations did not exceed the uncertainty bounds already considered in the TSPA (SNL 2007c, Table 6.3-3).

As for temperature effects, implementing the ABD Program could potentially change the temperature history of the waste form and waste package because of the change in the ratio between hot and cold waste packages (Section 3.1.1). This temperature difference, however, is expected to be small in comparison to the large ranges of uncertainty in the kinetic rates propagated in the in-package chemistry model simulations. Therefore, it is not expected that ABD activities would produce temperature changes that would markedly affect in-package chemistry model abstractions and radionuclide solubilities.

3.3.3 Modeling Cases for Early Failure, Igneous, Seismic, and Human Intrusion Scenario Classes

Four scenario classes based on initiating events are considered in the TSPA for Yucca Mountain: (1) the nominal scenario class, which includes all FEPs that are screened for inclusion according to the FEPs screening process, except those FEPs related to early waste package and drip shield failure, and igneous or seismic activity; (2) the early failure scenario class, which includes FEPs related to early waste package and drip shield failure due to manufacturing or material defects or to pre-emplacement operations including improper heat treatment; (3) the igneous scenario class, which is comprised of the igneous intrusion and volcanic eruption modeling cases, both of which are based on unlikely low-probability events; and (4) the seismic scenario class, which is comprised of the seismic ground motion modeling case and the seismic fault displacement modeling case. Based on regulatory specification, the modeling case for the human intrusion scenario class considers the consequences of inadvertent human intrusion on repository performance.

The effects of implementing the ABD Program on the nominal scenario class were discussed in Section 3.3.2. To the extent that FEPs in the nominal scenario are included in these other scenario classes, these effects would extend to these other scenario classes. Implementing the ABD Program will affect all scenarios and modeling cases, primarily because of the decrease in the number of CDSP waste packages (900 CDSP waste packages, which is ~8% reduction overall and ~26% for CDSP waste packages, as discussed in Section 3.1.1) and the associated reduction in the ratio of the number of CDSP waste packages to the number of CSNF and naval SNF waste packages from ~0.4 to ~0.3. How this change in inventory affects the various modeling cases is discussed in the following subsections.

Note that, as implied above, the postclosure evaluation results presented below generally assume there will be about 900 fewer CDSP waste packages, which is consistent with Design Examples 1 and 2 presented in Section 3.1.2. Discussions of Design Example 3 are an exception since this example assumes the number of CDSP waste packages remains the same.

3.3.3.1 Modeling Cases for Early Failure Scenario Class

The early failure scenario class includes two modeling cases: the drip shield early failure modeling case and the waste package early failure modeling case (SNL 2010a, Section 6.4). These modeling cases evaluate the consequences of waste package and drip shield failures that could result from manufacturing or handling-induced defects. The time at which a waste package or drip shield failure, resulting from a manufacturing or handling-induced defect, might occur is difficult to predict. Consequently, the failure is conservatively assumed to occur at the beginning of the realization. A further conservatism is that, upon failure, the waste package is assumed to completely fail such that it provides no barrier capability. The treatment of the drip shield is different in the two modeling cases. In the waste package early failure modeling case, the drip shield is assumed to function normally above the completely failed waste package. In the drip shield early failure modeling case, the drip shield and the waste package are assumed to completely fail, i.e., provide no barrier capability, at the beginning of the realization.

Reducing the number of CDSP waste packages (and, hence, the number of drip shields) included in the TSPA analyses affects both modeling cases in two ways. First, changing the number of waste packages would affect the probability of early failure of both waste packages and drip shields. Calculating the probability of waste package early failure and drip shield early failure is a function of the number of waste packages and drip shields, respectively, in the realization (SNL 2010a, Section 6.4). Changing the number of waste packages and drip shields will therefore change the probability of occurrence of waste package early failure and drip shield early failure.

Second, reducing the number of waste packages affects the doses calculated for the early failure scenario. The calculation included a single CSNF waste package and a single CDSP waste package in each of the five percolation subregions. The calculation also included cases with and without seepage in each of the percolation subregions. This approach resulted in the calculation of 20 dose histories for each realization. The equation used to calculate dose in each of the early failure modeling cases is a function of the probability of failure of the component (waste package or drip shield, discussed in the previous paragraph), the fraction of each waste package type in the repository, the fraction of each waste package type in the percolation subregion, the fraction of waste packages that are in the percolation subregion, and the number of waste packages in the repository (SNL 2010a, Section 6.1.2.4.2).

Reducing the number of CDSP waste packages in the repository such as would occur with Design Examples 1 or 2 could therefore lead to different expected dose results for these modeling cases. Dose calculations for Design Example 3, in which the same number of CDSP waste packages are emplaced, but the CDSP waste packages are partially filled, could be more complicated. The probability of failure for both waste packages and drip shields would be the same as in the TSPA case, but having some, or all, of the CDSP waste packages only partially filled would affect the dose histories calculated for each realization.

Changing the number of CDSP waste packages to be disposed of will affect probability and dose calculations in the early failure modeling case, but the effects are small and are not likely to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C). The waste package early failure modeling case and the drip shield early failure modeling case are both minor contributors to total dose (SNL 2010a, Section 8.2). In addition, the potential changes to the results would be small because the change in the number of waste packages involved in these modeling cases is small.

3.3.3.2 Igneous Intrusion Modeling Case

The igneous intrusion modeling case (SNL 2010a, Section 6.5.1) evaluates the consequences of an igneous intrusion event at the repository. The event is conceptualized as an igneous dike intersecting the underground workings of the repository. The interconnected nature of the underground workings, along with the flow characteristics of the intruding magma, cause the magma to fill all the emplacement drifts of the repository. As the magma flows into the emplacement drifts, it is assumed to engulf all waste packages and drip shields. The magma destroys all the waste packages, drip shields, and fuel cladding. Waste forms, encapsulated in solidified basalt, are exposed to percolating groundwater. As the waste forms degrade, radionuclides are mobilized and transported to the saturated zone below the repository.

To characterize the potential differences that could be created by the introduction of the ABD glass waste, the probability and consequences of the event must be considered. The probability of this event was evaluated by the probabilistic volcanic hazard analysis (PVHA) (CRWMS M&O 1996). The consequence of the event is to expose all the waste forms to dissolution and transport after the event.

The size, or footprint, of the repository is an input to the PVHA. Consequently, potential changes to the repository design could lead to differences in the PVHA results. The repository design has evolved over time. The repository footprint used for the TSPA was different from the repository footprint used for the PVHA. The difference in the calculated mean of the frequency of intersection that could result from the change in repository layout was evaluated (BSC 2004a, Section 6.5.2.1). The changes were shown to be within the uncertainty incorporated in the PVHA. Any changes resulting from the inclusion of ABD glass waste, e.g., Design Example 1, are likely to be well within the uncertainty incorporated in the PVHA.

Implementing the ABD Program will reduce the inventory of waste disposed of in the repository (Section 3.3.2.4). This reduction in inventory will result in a different calculated expected dose for this modeling case. However, the differences in inventory are small and are not likely to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C). Since all of the waste packages are destroyed in this modeling case, the expected dose would likely be reduced because of the reduction in the number of waste packages being emplaced. The magnitude of the reduction would probably be small because radionuclide content of the CDSP waste packages is much smaller than the radionuclide content of the CSNF waste packages, both on a per waste package basis and in terms of the total number of waste packages of each type in the inventory (SNL 2010a Table 6.3.7-3).

3.3.3.3 Volcanic Eruption Modeling Case

The volcanic eruption modeling case (SNL 2010a, Section 6.5.2.1) evaluates events involving a volcanic conduit that intersects the repository footprint and provides a pathway for the atmospheric release of volcanic tephra and radioactive waste from the repository. The events are characterized in terms of both probability and consequences. The event is conceptualized as a basaltic magma rising through the crust and intersecting the repository as an igneous dike. The probability of this event was evaluated by the PVHA (CRWMS M&O 1996).

The size, or footprint, of the repository is an input to the PVHA. Consequently, potential changes to the repository design could lead to differences in the PVHA results. The repository design has evolved over time. The repository footprint used for the TSPA was different from the repository footprint used for the PVHA. The difference in the calculated mean of the frequency of intersection that could result from the change in repository layout was evaluated (BSC 2004a, Section 6.5.2.1). The changes were shown to be within the uncertainty incorporated in the PVHA. Any changes resulting from the inclusion of ABD glass waste, e.g., Design Example 1, are likely to be well within the uncertainty incorporated in the PVHA.

Eruptive conduits may develop along an igneous dike as it intrudes. Erupting magma flowing in the conduits can entrain radioactive waste and transport it to the surface. At the surface, a portion of the erupting magma may form a tephra column that can transport the waste away from the repository to the accessible environment. If one of the conduits intersects a repository drift, then the waste packages within the conduit are assumed to be destroyed and the enclosed waste is entrained in the erupting magma. The mass of waste and radionuclide inventory to be included in the eruptive event are calculated based on (1) the proportion of CSNF and naval SNF waste packages to CDSP waste packages in the repository, (2) a sampled parameter that selects the number of waste packages affected by the eruptive event, and (3) a factor that accounts for the proportion of waste-contaminated magma erupted into the tephra column (SNL 2010a, Section 6.5.2.2). Reducing the number of CDSP waste packages in the repository as assumed in Design Examples 1 or 2 could lead to a different calculated radionuclide inventory in the erupted waste. Design Example 3 could also lead to a different calculated dose because a portion of the erupted waste could be from partially filled CDSP waste packages. However, the differences in inventory are small and are not likely to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C), primarily because the volcanic eruption modeling case is a minor contributor to total dose (DOE 2008, Figure 2.4-18).

3.3.3.4 Seismic Ground Motion Modeling Case

This modeling case evaluates the damage to waste packages resulting from seismic ground motion (SNL 2010a, Section 6.6.1.2.2). The analyses focus on the interactions between the major components of the engineered barrier system, namely the waste packages, drip shields, and the pallets on which the waste packages rest. Three-dimensional models evaluate the interactions between multiple waste packages, drip shields, and pallets that result from seismic ground motions. These kinematic calculations are appropriate for early times after repository closure, when drip shields are intact and the waste packages can move freely in response to the seismic ground motions. The analyses are used to define the history of impact parameters for interactions—collisions of the waste packages, drip shields, and pallets—as a function of the applied ground motions. The results of the calculations provide input parameters for detailed finite element analyses of individual waste package-to-waste package and waste package-to-pallet impacts. The detailed finite element models are used to characterize damage areas on waste packages and to estimate the size of the damage area for a given event.

Later in repository history drip shields are collapsed and the drifts are filled with rubble. At these times the kinematic calculations, which assume that waste packages are free to move in response to the ground motions, are no longer appropriate. Seismic ground motions occurring at these later times are evaluated using detailed finite element models, but these models have different initial and boundary conditions. Calculations using these models evaluate the deformation and damage of a drip shield under static and dynamic loading conditions and provide estimates of damage to waste packages surrounded by rubble after the drip shield plates have failed.

Damage abstractions consist of damage lookup tables developed for waste package-to-waste package and waste package-to-pallet impacts. The damage induced by these impacts is calculated from the kinematic impact parameters for end-to-end impacts and for waste package-pallet impacts by using lookup tables. A direct correlation is made between damaged surface area and impact velocity, angle of impact, force of impact, and/or impact location, allowing the kinematic calculations to represent the damage to multiple waste packages (SNL 2007j, Section 6.3).

Overall, there are significantly fewer waste package-to-waste package impacts than waste package-to-pallet impacts in the kinematic analyses. Furthermore, the damaged area computed from waste package-to-waste package impacts is generally far less than the damaged area computed from waste package-to-pallet impacts (SNL 2007j, Section 6.3.2.2.3).

In addition to damaging the components of the EBS, seismic ground motion can damage the integrity of the drift itself. Drift degradation, from seismically induced rock fall, in the nonlithophysal rock units is excluded from the TSPA analyses based on low consequence (excluded FEP 1.2.03.02.0B). Seismically induced drift collapse in the lithophysal rock units is included in the TSPA analyses. Drifts in the lithophysal zones are predicted to collapse into small rock fragments as a result of seismic ground motion. The volume of material that collapses is a function of the peak ground velocity (PGV) of the event. Rockfall abstractions were developed for three different values of PGV. These abstractions are used to predict the rockfall volume from a seismic event (SNL 2007k, Section 6.7). The rockfall can impact the drip shield fragility and restrict the movement of the waste packages during seismic events, especially in cases in which the drift is fully collapsed.

Seismically induced drift collapse can also impact seepage calculations in the TSPA model. Two aspects of the seepage implementation in the model need to be considered to understand the impacts of drift collapse. Calculations for drift seepage differ for collapsed and noncollapsed drifts. The seepage model implementation is different for the lithophysal and nonlithophysal rock units. The calculations are summarized in 3D tables of mean seepage rate and seepage rate standard deviation as functions of percolation flux, capillary strength, and fracture permeability. Two sets of lookup tables were developed: one for intact drifts and another for collapsed drifts (SNL 2007a, Sections 6.4.2.4.2 and 6.2.1 [a]). The use of these two sets of tables is different for drifts in lithophysal and nonlithophysal units (SNL 2010a, Section 6.3.3.1.2).

The effect of implementing the ABD Program on the seismic ground motion modeling case depends on how the reduction in the number of CDSP waste packages is managed. Implementing Design Example 1 (reducing the repository footprint and maintaining waste package spacing) should not impact any of the seismic damage abstractions developed for TSPA. However, reducing the number of drifts could lead to different results for the implementation of the seepage model. The seepage lookup tables should not be affected and the methodology for implementing the tables would not need to change, but the number of CDSP waste packages in the lithophysal and/or nonlithophysal rock units could change and the values for seepage parameters discussed in Section 3.3.2.2 could change.

Implementing Design Example 2 (maintaining repository footprint by increasing the spacing between waste packages) could lead to different damage abstractions because of changes to waste package spacing. The TSPA calculations are based on a 10 cm spacing between waste packages; implementing Design Example 2 will increase this spacing. Increasing the waste package spacing will not affect the analyses of waste package-to-pallet impacts, but it would affect the kinematic analyses of waste package-to-waste package impacts, thereby affecting the calculated damage abstractions for the waste package-to-waste package impacts.

Implementing Design Example 3 (maintaining repository footprint and waste package spacing by reducing CDSP waste package contents) could also affect the damage abstractions because the partially filled CDSP waste packages will have less mass than the waste packages used in the TSPA calculations. Reducing the mass in CDSP waste packages would affect the kinematic calculations used to create the damage abstractions.

The expected dose calculated for this modeling case could be different if ABD glass waste is disposed of. The expected dose for the 10,000-year seismic ground motion calculation does not include the CSNF waste packages because the probability of damaging them from seismic ground motion is very low during the first 10,000 years after repository closure (SNL 2010a, Section 6.6.1.3.1). Consequently, reducing the number of CDSP waste packages or the quantity of waste in the CDSP waste packages in the TSPA analyses could affect the calculated expected dose in the first 10,000 years.

Reducing the number of CDSP waste packages could also affect the expected dose for the 1,000,000-year seismic ground motion modeling case, which includes damage to CSNF waste packages, because this calculation includes uncertainty in the occurrence and extent of damage to each type of waste package (SNL 2010a, Section 6.1.2.4.4). Reducing the number of CDSP waste packages decreases the ratio of CDSP waste packages to CSNF and naval SNF waste packages from ~0.4 to ~0.3, which could be reflected in the results of the dose calculation.

In summary, for the seismic ground motion modeling case, implementing the ABD Program could affect seismic damage abstractions, implementation of the seepage model, the number of waste packages in lithophysal and/or nonlithophysal rock units, and the number or ratio of waste packages considered in the dose calculations. However, these effects are expected to be small and are not likely to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C).

3.3.3.5 Fault Displacement Modeling Case

This modeling case evaluates the damage to waste packages caused by fault displacement (SNL 2010a, Section 6.6.1.3.1). If displacement occurs on a fault with a waste package emplaced above it, then the relative displacement of the fault blocks could cause shearing of the waste package. The probabilistic seismic hazard analysis (PSHA) (CRWMS M&O 1998b) provided critical information for the fault displacement hazard analysis. The PSHA was supplemented by analyses completed by the TSPA team and documented in *Seismic Consequence Abstraction* (SNL 2007k). The amount of displacement required to damage a waste package depends on the design and the type of waste package; an analysis was completed to determine the minimum displacement required to damage each type of waste package (SNL 2007k, Table 6-59). Known faults that could potentially damage waste packages through this mechanism were identified by the PSHA. The PSHA analysis also considered two “generic” faults included to represent small displacement faults that would not be recognized during surface geologic mapping but would be encountered as drifts were excavated.

The maximum displacement on the identified faults, as a function of maximum exceedance frequency, was inferred by the PSHA experts (SNL 2007k, Table 6-61). The information on the minimum displacement required to damage a waste package can be combined with the inferred fault displacements to determine the mean annual exceedance frequencies that could cause waste package damage on each of the faults (SNL 2007k, Table 6-65). Note that this value is different for the CSNF and the CDSP waste packages because of differences in the sizes of these types of waste packages.

For the TSPA modeling case it was necessary to estimate the number of potential intersections between emplacement drifts in the repository and the identified faults. The potential number of intersections between the “generic” faults and the emplacement drifts was estimated based on the frequency of faults observed in the underground excavations developed during site characterization (SNL 2007k, Section 6.11.2.2). The expected number of waste packages to be emplaced on each fault was also determined (SNL 2007k, Table 6-66).

Calculating the dose for this modeling case includes the number of each type of waste package and the seepage conditions at the location of each waste package damaged by the fault displacement event (SNL 2010a, Section 6.1.2.4.4). Reducing the number of CDSP waste packages by 900, e.g., Design Examples 1 and 2, would change the ratio of CSNF and naval SNF waste packages to CDSP waste packages, thereby affecting dose calculations. Changing the repository footprint (e.g., Design Example 1) could also affect the seepage conditions, and thus the dose calculations, by having fewer repository drifts. Emplacing partially filled CDSP waste packages (e.g., Design Example 3) could also affect dose calculations because each CDSP waste package would have less waste than estimated in the TSPA. However, these effects are expected to be small and are not likely to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C) primarily because the fault displacement modeling case is a minor contributor to total dose (DOE 2008, Figure 2.4-18).

3.3.3.6 Human Intrusion Modeling Case

The human intrusion modeling case (SNL 2010a, Section 6.7) is a stylized analysis based on regulatory requirements imposed by the NRC (10 CFR 63.322). Several aspects of the analysis are specified by the NRC rule. The human intrusion event involves a drill bit penetrating a single waste package. For any given event, both the waste package type (i.e., CSNF or CDSP) that is intersected and the waste package location (i.e., percolation subregion) are uncertain. The probability of an event intersecting a CSNF or a CDSP waste package is set equal to the fraction of each waste package type in the repository inventory. Consequently, in calculating the dose consequences for the human intrusion modeling case in the TSPA, the probability of the human intrusion event intersecting a CSNF waste package is about 0.7 and the probability of the event intersecting a CDSP waste package is about 0.3.

Reducing the number of CDSP waste packages in the inventory while the number of CSNF waste packages remains constant, i.e., Design Examples 1 or 2, would change these probabilities to about 0.77 and 0.23, respectively and could lead to different expected dose results. If Design Example 3 were implemented, the probabilities of the events intersecting each type of waste package would be the same but the partially filled CDSP waste packages would affect the dose calculation. However, these effects are expected to be small and are not likely to change the overall conclusion that a generic repository similar to Yucca Mountain would meet all applicable regulatory requirements (Appendix C). It is also worth noting that the calculated dose for the human intrusion modeling case is well below the regulatory standard (DOE 2008, Section 2.4.3.3.1).

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4. CONCLUSIONS

Potential effects of implementing the ABD Program at SRS on future repository licensing were evaluated using the license application submitted for the proposed repository at Yucca Mountain as a framework. This evaluation included the effects on preclosure safety for both a generic repository and a generic repository that is similar to the one proposed for Yucca Mountain. It also included the effects on the postclosure PA for a repository that is similar to the one proposed for Yucca Mountain.

Potential effects of implementing the ABD Program on preclosure safety for both repositories include the following:

- Event sequence categorization will not be affected. That is, no event sequences that are categorized as Category 2 would need to be re-categorized as Category 1. Similarly, no event sequences that are categorized as beyond Category 2 would need to be re-categorized as Category 2.
- The criticality safety technical bases for preclosure operations will remain valid.
- Preclosure shielding design and worker dose estimates will remain valid.
- Preclosure design and operations will not be affected.

For a generic repository, the effects of implementing the ABD Program on postclosure performance are more challenging to evaluate at this time because much of the information needed to complete a meaningful evaluation is not available. The postclosure performance of a future repository will be considered as a part of a postclosure PA developed for that specific disposal concept. The future PA will include the design of the repository, which will necessarily consider the characteristics of the host rock and the waste projected to be emplaced in that repository. As appropriate, these characteristics will be provided by site-specific data deemed representative of the proposed repository site and design. The PA will evaluate the fissile content of all the HLW glass as well as how the HLW affects and is affected by the host rock and the design of the waste packages and repository. Thus, FEPs concerning criticality, waste form degradation, and thermal loading will be evaluated as part of the postclosure PA, along with any other FEPs that affect or are affected by implementing the ABD Program at SRS. Current disposal concepts of generic repositories do not contain enough detail to do a PA similar to the PA described in this report for a repository similar to Yucca Mountain. However, based on the expected characteristics of the ABD glass, no concerns regarding disposal of that waste form in a generic repository were identified.

For a repository that is similar to the one proposed for Yucca Mountain, the effects on the design of the repository and the postclosure PA from implementing the ABD Program include the following:

- The eight postclosure in-package criticality FEPs considered in the Yucca Mountain repository license application would remain screened out of repository PA calculations. The three DSNF types in the short DSNF standardized canister (i.e., TRIGA, Enrico Fermi, ATR) could be codisposed with ABD glass without anticipated significant effects on the reactivity of degraded configurations. The bases for this conclusion include minimal changes to reflection for partially degraded configurations as well as reduced enrichment and added gadolinium for fully degraded configurations. Because the DSNF basket designs have not been finalized, additional neutron absorbers can be added in the form of $GdPO_4$ shot, if deemed necessary.

- The eight postclosure external criticality FEPs that were considered in the Yucca Mountain repository license application would remain screened out of repository PA calculations. The conclusion is supported by the fact that the probability of external criticality is an insignificant contributor to the total probability of criticality. Additionally, the amount of fissile material in ABD glass is much lower than similar waste forms for which external criticality was deemed incredible.
- Approximately 900 fewer CDSP waste packages would need to be disposed. This change represents a ~26% decrease in the number of CDSP waste packages to be disposed, a ~8% decrease in the total number of waste packages to be disposed, and a decrease in the ratio of the number of CDSP waste packages to the number of CSNF and naval SNF waste packages from ~0.4 to ~0.3.
- Reducing the number of CDSP waste packages (and DSNF canisters) to be disposed and the change in the ratio of CDSP to SNF waste packages may affect several different components of the postclosure PA:
    - Multiscale thermal-hydrologic model
    - Seepage models
    - In-drift convection and condensation models
    - Inventory models
    - Early failure modeling cases
    - Igneous intrusion modeling case
    - Volcanic eruption modeling case
    - Seismic ground motion modeling case
    - Fault displacement modeling case
    - Human intrusion modeling case
- The change in the glass composition may affect the technical bases for several different components of the postclosure PA:
    - In-package chemistry models
    - Radionuclide solubility models

As noted throughout this report, however, all of the above changes to these components of the postclosure PA are expected to be small and are not expected to change the conclusion that a repository similar to the one proposed for Yucca Mountain would meet all applicable requirements.

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APPENDIX A—POSTCLOSURE FEP EVALUATION RESULTS

Using the approach described in Section 2.2.2, the study team evaluated each of the 374 postclosure FEPs presented in *Features, Events, and Processes for the Total System Performance Assessment: Analyses* (SNL 2008a) for design, modeling, and inclusion/exclusion justifications that could be affected by implementing the ABD Program. The evaluation results identified 8 criticality FEPs and 27 noncriticality FEPs as being potentially affected. A consolidated list of these FEPs is provided in Table A-1, which includes the FEP number, name, description, and SAR screening decision as well as evaluation comments indicating the manner in which the FEP could be affected.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
8 Potentially Affected Criticality FEP				
2.1.14.15.0A	In-Package Criticality (Intact Configuration)	The waste package internal structures and the waste form remain intact. If there is a breach (or are breaches) in the waste package that allows water to either accumulate or flow-through the waste package, then criticality could occur in situ.	Excluded low probability	In-package intact criticality MCNP models in the criticality calculations for the eight representative DSNF types will have to be revised to reflect the ABD glass composition, including increased fissile material and neutron absorber (i.e., Gd) concentrations. These revisions are expected to demonstrate that intact configurations will remain subcritical for all representative DSNF types codisposed with ABD glass. The only differences between the four “intact configuration” FEPs are related to probability of waste package failure, which is directly correlated to probability of moderator presence. These considerations are not affected by ABD glass. Other considerations important to criticality (e.g., neutron absorber misload, fuel misload) are the same for all eight in-package criticality FEPs and are not affected by ABD glass. The current analyses do not consider out-of-specification HLW composition as credible.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.16.0A	In-Package Criticality (Degraded Configurations)	The waste package internal structures and the waste form may degrade. If a potentially critical configuration (sufficient fissile material and neutron moderator, lack of neutron absorbers) develops, a criticality event could occur in situ. Potential in situ critical configurations are defined in Figures 3.2a and 3.2b of *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003).	Excluded low probability	Degraded criticality calculations and supporting geochemistry degradation and release models and analyses will have to be revised based on the new ABD glass composition. These analyses will support one of the following conclusions: • All degraded configurations remain subcritical for the analyzed DSNF types (with current basket material design and neutron absorber loading) codisposed with "x" number of ABD glass canisters per waste package. This outcome is the likely conclusion for all DSNF types. • Additional neutron absorbers in the DSNF basket or in the form of fillers (shot) are needed to demonstrate subcriticality for some of the degraded configurations. This conclusion could be reached if the geochemistry analyses demonstrate that there is a credible mechanism for releasing a significant fraction of the added Gd in the ABD glass from the waste package (e.g., a highly soluble gadolinium compound is formed instead of $GdPO_4$ or Gd_2O_3). This situation is not anticipated, but if credible, the poison loading for FFTF, Fermi, Shippingport PWR, Shippingport LWBR, ATR, or TRIGA could be increased. • De-rate the SNF loading in some DSNF canisters. This option is not viable for Fort St. Vrain, Shippingport PWR, Shippingport LWBR, and TMI. • Preclude the codisposal of a specific DSNF type with ABD glass. • Revisit the credibility of the analyzed degraded configurations. The current postulated degraded configurations, some of which were analyzed in the late 1990s, were developed with considerations well beyond 10,000 years (unlike the degradation assumptions for CSNF waste packages) and were extremely conservative (some nonmechanistic configurations). The consideration of these noncredible configurations may be obviated for the ABD glass analyses.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.18.0A	In-Package Criticality Resulting from a Seismic Event (Intact Configuration)	The waste package internal structures and the waste form remain intact either during or after a seismic disruptive event. If there is a breach (or are breaches) in the waste package that allow(s) water to either accumulate or flow through the waste package, then criticality could occur in situ.	Excluded low probability	In-package intact criticality MCNP models in the criticality calculations for the eight representative DSNF types will have to be revised to reflect the ABD glass composition, including increased fissile material and neutron absorber (i.e., Gd) concentrations. These revisions are expected to demonstrate that intact configurations will remain subcritical for all representative DSNF types codisposed with ABD glass. The only differences between the four “intact configuration” FEPs are related to probability of waste package failure, which is directly correlated to probability of moderator presence. These considerations are not affected by ABD glass. Other considerations important to criticality (e.g., neutron absorber misload, fuel misload) are the same for all eight in-package criticality FEPs and are not affected by ABD glass. The current analyses do not consider out-of-specification HLW composition as credible.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.19.0A	In-Package Criticality Resulting from a Seismic Event (Degraded Configurations)	Either during or as a result of a seismic disruptive event, the waste package internal structures and the waste form may degrade. If a critical configuration develops, criticality could occur in situ. Potential in situ critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003, Figures 3.2a and 3.2b).	Excluded low probability	Degraded criticality calculations and supporting geochemistry degradation and release models and analyses will have to be revised based on the new ABD glass composition. These analyses will support one of the following conclusions: • All degraded configurations remain subcritical for the analyzed DSNF types (with current basket material design and neutron absorber loading) codisposed with "x" number of ABD glass canisters per waste package. This outcome is the likely conclusion for all DSNF types. • Additional neutron absorbers in the DSNF basket or in the form of fillers (shot) is needed to demonstrate subcriticality for some of the degraded configurations. This conclusion could be reached if the geochemistry analyses demonstrate that there is a credible mechanism for releasing a significant fraction of the added Gd in the ABD glass from the waste package (e.g., a highly soluble gadolinium compound is formed instead of $GdPO_4$ or Gd_2O_3). This situation is not anticipated, but if credible, the poison loading for FFTF, Fermi, Shippingport PWR, Shippingport LWBR, ATR or TRIGA could be increased. • De-rate the SNF loading in some DSNF canisters. This option is not viable for Fort St. Vrain, Shippingport PWR, Shippingport LWBR, and TMI. • Preclude the codisposal of a specific DSNF type with ABD glass. • Revisit the credibility of the analyzed degraded configurations. The current postulated degraded configurations, some of which were analyzed in the late 1990s, were developed with considerations well beyond 10,000 years (unlike the degradation assumptions for CSNF waste packages) and were extremely conservative (some nonmechanistic configurations). The consideration of these noncredible configurations may be obviated for the ABD glass analyses.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.21.0A	In-Package Criticality Resulting from Rockfall (Intact Configuration)	The waste package internal structures and the waste form remain intact either during or after a rockfall event. If there is a breach (or are breaches) in the waste package that allow(s) water to either accumulate or flow through the waste package, then criticality could occur in situ.	Excluded low probability	In-package intact criticality MCNP models in the criticality calculations for the eight representative DSNF types will have to be revised to reflect the ABD glass composition, including increased fissile material and neutron absorber (i.e., Gd) concentrations. These revisions are expected to demonstrate that intact configurations will remain subcritical for all representative DSNF types codisposed with ABD glass. The only differences between the four "intact configuration" FEPs are related to probability of waste package failure, which is directly correlated to probability of moderator presence. These considerations are not affected by ABD glass. Other considerations important to criticality (e.g., neutron absorber misload, fuel misload) are the same for all eight in-package criticality FEPs and are not affected by ABD glass. The current analyses do not consider out-of-specification HLW composition as credible.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.22.0A	In-Package Criticality Resulting from Rockfall (Degraded Configurations)	Either during or as a result of a rockfall event, the waste package internal structures and the waste form may degrade. If a critical configuration develops, criticality could occur in situ. Potential in situ critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003, Figures 3.2a and 3.2b).	Excluded low probability	Degraded criticality calculations and supporting geochemistry degradation and release models and analyses will have to be revised based on the new ABD glass composition. These analyses will support one of the following conclusions: • All degraded configurations remain subcritical for the analyzed DSNF types (with current basket material design and neutron absorber loading) codisposed with "x" number of ABD glass canisters per waste package. This outcome is the likely conclusion for all DSNF types. • Additional neutron absorbers in the DSNF basket or in the form of fillers (shot) is needed to demonstrate subcriticality for some of the degraded configurations. This conclusion could be reached if the geochemistry analyses demonstrate that there is a credible mechanism for releasing a significant fraction of the added Gd in the ABD glass from the waste package (e.g., a highly soluble gadolinium compound is formed instead of $GdPO_4$ or Gd_2O_3). This situation is not anticipated, but if credible, the poison loading for FFTF, Fermi, Shippingport PWR, Shippingport LWBR, ATR or TRIGA could be increased. • De-rate the SNF loading in some DSNF canisters. This option is not viable for Fort St. Vrain, Shippingport PWR, Shippingport LWBR, and TMI. • Preclude the codisposal of a specific DSNF type with ABD glass. • Revisit the credibility of the analyzed degraded configurations. The current postulated degraded configurations, some of which were analyzed in the late 1990s, were developed with considerations well beyond 10,000 years (unlike the degradation assumptions for CSNF waste packages) and were extremely conservative (some nonmechanistic configurations). The consideration of these noncredible configurations may be obviated for the ABD glass analyses.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.24.0A	In-Package Criticality Resulting from an Igneous Event (Intact Configuration)	The waste package internal structures and the waste form remain intact either during or after an igneous disruptive event. If there is a breach (or are breaches) in the waste package that allow(s) water to either accumulate or flow through the waste package, then criticality could occur in situ.	Excluded low probability	In-package intact criticality MCNP models in the criticality calculations for the eight representative DSNF types will have to be revised to reflect the ABD glass composition, including increased fissile material and neutron absorber (i.e., Gd) concentrations. These revisions are expected to demonstrate that intact configurations will remain subcritical for all representative DSNF types codisposed with ABD glass. The only differences between the four “intact configuration” FEPs are related to probability of waste package failure, which is directly correlated to probability of moderator presence. These considerations are not affected by ABD glass. Other considerations important to criticality (e.g., neutron absorber misload, fuel misload) are the same for all eight in-package criticality FEPs and are not affected by ABD glass. The current analyses do not consider out-of-specification HLW composition as credible.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.14.25.0A	In-Package Criticality Resulting from an Igneous Event (Degraded Configurations)	Either during or as a result of an igneous disruptive event, the waste package internal structures and the waste form may degrade. If a critical configuration develops, criticality could occur in situ. Potential in situ critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003, Figures 3.2a and 3.2b).	Excluded low probability	Degraded criticality calculations and supporting geochemistry degradation and release models and analyses will have to be revised based on the new ABD glass composition. These analyses will support one of the following conclusions: • All degraded configurations remain subcritical for the analyzed DSNF types (with current basket material design and neutron absorber loading) codisposed with "x" number of ABD glass canisters per waste package. This outcome is the likely conclusion for all DSNF types. • Additional neutron absorbers in the DSNF basket or in the form of fillers (shot) is needed to demonstrate subcriticality for some of the degraded configurations. This conclusion could be reached if the geochemistry analyses demonstrate that there is a credible mechanism for releasing a significant fraction of the added Gd in the ABD glass from the waste package (e.g., a highly soluble gadolinium compound is formed instead of $GdPO_4$ or Gd_2O_3). This situation is not anticipated, but if credible, the poison loading for FFTF, Fermi, Shippingport PWR, Shippingport LWBR, ATR or TRIGA could be increased. • De-rate the SNF loading in some DSNF canisters. This option is not viable for Fort St. Vrain, Shippingport PWR, Shippingport LWBR, and TMI. • Preclude the codisposal of a specific DSNF type with ABD glass. • Revisit the credibility of the analyzed degraded configurations. The current postulated degraded configurations, some of which were analyzed in the late 1990s, were developed with considerations well beyond 10,000 years (unlike the degradation assumptions for CSNF waste packages) and were extremely conservative (some nonmechanistic configurations). The consideration of these noncredible configurations may be obviated for the ABD glass analyses.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
27 Potentially Affected Noncriticality FEPs				
1.1.07.00.0A	Repository Design	This FEP addresses the consideration of the design of the repository and the ways in which the design contributes to long-term performance. The performance assessment must account for design features, material characteristics, and the ways in which the design influences the evolution of the in-drift environment.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	Several factors could lead to differences in the design of a repository layout for a Yucca Mountain-similar repository. Reducing the number of waste packages to be disposed is the factor that is most likely to lead to differences. The reduction in the number of CDSP waste packages could be addressed in different ways. The total length of the drifts could be reduced, either by eliminating some drifts or by reducing the lengths of some or all of the drifts. The spacing between waste packages could be increased. Alternatively, some combination of these changes could be implemented.
1.2.02.03.0A	Fault Displacement Damages EBS Components	Movement of a fault that intersects drifts within the repository may cause the EBS components to experience related movement or displacement. Repository performance may be degraded by such occurrences as tilting of components, component-to-component contact, or drip shield separation. Fault displacement could cause a failure as significant as shearing of drip shields and waste packages by virtue of the relative offset across the fault, or as extreme as exhumation of the waste to the surface.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	Table 6.6-1 of SNL (2010a) gives the "Expected Waste Package Failure Due to Fault Displacement" and values are given for both CSNF and CDSP waste packages. The values given in this table will change if the number of CDSP waste packages is decreased by 900.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
1.2.03.02.0A	Seismic Ground Motion Damages EBS Components	Seismic activity that causes repeated vibration of the EBS components (drip shield, waste package, pallet, and invert) could result in severe disruption of the drip shields and waste packages, through vibration damage or through contact between EBS components. Such damage mechanisms could lead to degraded performance.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	"…expected dose for the Seismic GM Modeling Case for 10,000 years relies on a number of simplifying approximations…Only CDSP WPs are considered because the probability of damaging the CSNF WP is very low during the first 10,000 years" (SNL 2010a, Section 6.6.1.3.1). The change in the number of CDSP waste packages (Section 3.1.1) and possible changes in radionuclide solubilities (Section 3.3.2.5) will affect the technical basis for modeling the effects of seismic ground motion for a repository similar to that proposed for Yucca Mountain. In addition, design modifications to address the reduced number of waste packages could include changing the spacing between waste packages, which is important to calculations estimating the damage from waste package-to-waste package interactions caused by seismic ground motion (Section 3.3.3.4).
1.2.03.02.0C	Seismic-Induced Drift Collapse Damages EBS Components	Seismic activity could produce jointed-rock motion and/or changes in rock stress leading to enhanced drift collapse that could impact drip shields, waste packages, or other EBS components. Possible effects include both dynamic and static loading.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	Lithophysal vs. nonlithophysal fractions The analysis of drift collapse is significantly different for the lithophysal and nonlithophysal zones. Reducing the number of CDSP waste packages by 900 will change the number of waste packages in each zone and, possibly, the ratio of waste package types in each zone. The details depend on how the design is changed.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
1.2.03.02.0D	Seismic-Induced Drift Collapse Alters In-Drift Thermal-hydrology	Seismic activity could produce jointed-rock motion and/or changes in rock stress leading to enhanced drift collapse and/or rubble infill throughout part or all of the drifts. Drift collapse could impact flow pathways and condensation within the EBS, mechanisms for water contact with EBS components, and thermal properties within the EBS.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	The analysis of drift collapse is significantly different for the lithophysal and nonlithophysal zones. Reducing the number of CDSP waste packages by 900 will change the number of waste packages in each zone and, possibly, the ratio of waste package types in each zone. The details depend on how the design is changed.
1.2.04.06.0A	Eruptive Conduit to Surface Intersects Repository	As a result of an igneous intrusion, one or more volcanic vents may form at land surface. The conduit(s) supplying the vent(s) could pass through the repository, interacting with and entraining waste.	Included 10 CFR 63.311 (proposed)	“The fraction of each waste package type, commercial SNF or codisposal, and hence the mass and inventory content of waste erupted is proportional to the fraction of commercial SNF versus codisposal waste packages emplaced in the repository.” (DOE 2008, Section 2.4.2.3.2.1.12.2). Reducing the number of CDSP waste packages will change the inventory content of waste erupted, because it will change the CSNF/CDSP waste package ratio. Reducing the number of CDSP waste packages by ~900 may reduce size of the repository footprint and, consequently, reduce the probability of an intersection by an eruptive conduit.
1.2.04.07.0A	Ashfall	Finely divided waste particles may be carried up a volcanic vent and deposited on the land surface from an ash cloud.	Included 10 CFR 63.311 (proposed)	“The fraction of each waste package type, commercial SNF or codisposal, and hence the mass and inventory content of waste erupted is proportional to the fraction of commercial SNF versus codisposal waste packages emplaced in the repository.” (DOE 2008, Section 2.4.2.3.2.1.12.2). Reducing the number of CDSP waste packages will change the inventory content of waste erupted because the CSNF/CDSP waste package ratio will change. Reducing the number of CDSP waste packages by ~900 may reduce size of the repository footprint and, consequently, reduce the probability of an intersection by an eruptive conduit.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
1.4.02.02.0A	Inadvertent Human Intrusion	Humans could accidentally intrude into the repository. Without appropriate precautions, intruders could experience high radiation exposures. Moreover, containment may be left damaged, which could increase radionuclide release rates to the biosphere. Inadvertent human intrusion might occur during scientific, mineral or geothermal exploration.	Included 10 CFR 63.321 (proposed)	"Based on the proportion of emplaced waste package types, the probability of sampling a commercial SNF waste package is about 0.7, whereas the probability of selecting a codisposal waste package is about 0.3" (DOE 2008, Section 2.4.3.3.1). Reducing the number of CDSP waste packages will change these probabilities of sampling because the CSNF/CDSP waste package ratio will change.
2.1.01.01.0A	Waste Inventory	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, and the nonradionuclide inventory (typically in units of density or concentration), including chemical waste constituents. The radionuclide composition of the waste will vary due to initial enrichment, burn-up, the number of fuel assemblies per waste package, and the decay time subsequent to discharge of the fuel from the reactor.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	Implementing the ABD Program will decrease the number of CDSP waste packages to be disposed of and could change the estimates of grams of each radionuclide per CDSP waste package.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.03.08.0A	Early Failure of Waste Packages	Waste packages may fail prematurely because of manufacturing defects, improper sealing, or other factors related to quality control during manufacture and emplacement.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	The calculation of early failure probability values is a function of the number of waste packages or drip shields in the realization (SNL 2010a, Equation 6.4-1). The results of evaluating Equation 6.4-1 for waste package early failures and drip shield early failures in the TSPA are given in SNL (2010a, Table 6.4-1). If the number of waste packages and/or drip shields included in the calculation is changed, then the results reported in SNL (2010a, Table 6.4-1) will be different. "The TSPA model computes the dose resulting from early failure of a single waste package of each type occurring in each of the five percolation subregions, with and without seepage in each percolation subregion, for a total of 20 dose histories for each epistemic realization. The results are then combined with the sampled rate of waste package early failure, the numbers of waste packages of each type, and the seepage fraction for each percolation bin to calculate the expected dose for each epistemic realization" (DOE 2008, Section 2.4.2.3.2.1.12.1). The numbers of waste packages of each type are determined from the CDSP/CSNF ratio. If this ratio changes because of a change in the number of CDSP waste packages, then the calculated expected dose for this modeling case will be different.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.03.08.0B	Early Failure of Drip Shields	Drip shields may fail prematurely because of manufacturing defects, improper sealing, or other factors related to quality control during manufacture and emplacement.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	The calculation of early failure probability values is a function of the number of waste packages or drip shields in the realization (SNL 2010a, Equation 6.4-1). The results of evaluating Equation 6.4-1 for waste package early failures and drip shield early failures in the TSPA are given in SNL (2010a, Table 6.4-1). If the number of waste packages and/or drip shields included in the calculation is changed then the results reported in SNL (2010a, Table 6.4-1) will be different. "The TSPA model computes the dose resulting from early failure of a single waste package of each type occurring in each of the five percolation subregions, with and without seepage in each percolation subregion, for a total of 20 dose histories for each epistemic realization. The results are then combined with the sampled rate of waste package early failure, the numbers of waste packages of each type, and the seepage fraction for each percolation bin to calculate the expected dose for each epistemic realization" (DOE 2008, Section 2.4.2.3.2.1.12.1). The numbers of waste packages of each type are determined from the CDSP/CSNF ratio. If this ratio changes because of a change in the number of CDSP waste packages, then the calculated expected dose for this modeling case will be different.
2.1.08.01.0A	Water Influx at the Repository	An increase in the unsaturated water flux at the repository may affect thermal, hydrologic, chemical, and mechanical behavior of the system. Increases in flux could result from climate change, but the cause of the increase is not an essential part of the FEP.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	Percolation subregions are used to evaluate water influx at the repository. Properties of the multiscale thermal-hydrologic model subdomains are used to define these subregions. If the reduction in waste packages is addressed by reducing the number, and/or distribution, of repository drifts then the number of multiscale thermal-hydrologic model subdomains will decrease and the percolation subregions could be redefined, giving a different result.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.08.02.0A	Enhanced Influx at the Repository	An opening in unsaturated rock may alter the hydraulic potential, affecting local saturation around the opening and redirecting flow. Some of the flow may be directed to the opening where it is available to seep into the opening.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	Calculations of seepage into the drift openings could be affected in different ways by implementing the ABD Program based on how the reduction in CDSP waste packages is addressed. The seepage modeling inputs from the multiscale thermal-hydrologic model (i.e., percolation rates and boiling duration at the waste packages) could be affected as could the calculation of three seepage properties: seepage rate, seepage percentage, and seepage fraction. For example, a change in the number of drifts could affect the gridding used to reflect the repository layout as well as percolation rates, the boiling duration, and the seepage properties. Alternatively, a change in waste package spacing changes the drift segment length, which could affect boiling duration and seepage properties.
2.1.08.04.0A	Condensation Forms on Roofs of Drifts (Drift-Scale Cold Traps)	Emplacement of waste in drifts creates thermal gradients within the repository. Such thermal gradients can lead to drift-scale cold traps characterized by latent heat transfer from warmer to cooler locations. This mechanism can result in condensation forming on the roof or other parts of the drifts, leading to enhanced dripping on the drip shields, waste packages, or exposed waste material.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	The in-drift condensation model could be affected in different ways by implementing the ABD Program based on what design changes might be made to accommodate the reduction in CDSP waste packages. For example, the condensation calculations are completed for seven drifts within the current repository layout. The selection of drifts could change with a change in layout. A change in waste package spacing could also cause differences. A change in the line load heat source and/or the seven-waste package sequence could affect different parts of the condensation calculations. In addition, changes in repository layout could cause upstream differences in how percolation is handled and percolation binning that affect condensation since condensation is evaluated as a function of percolation.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.08.04.0B	Condensation Forms at Repository Edges (Repository-Scale Cold Traps)	Emplacement of waste in drifts creates thermal gradients within the repository. Such thermal gradients can lead to repository-scale cold traps characterized by latent heat transfer from warmer to cooler locations. This mechanism can result in condensation forming at repository edges or elsewhere in the EBS, leading to enhanced dripping on the drip shields, waste packages, or exposed waste material.	Included 10 CFR 63.311 (proposed) 10 CFR 63.331	The in-drift condensation model could be affected in different ways by implementing the ABD Program based on what design changes might be made to accommodate the reduction in CDSP waste packages. For example, the condensation calculations are completed for seven drifts within the current repository layout. The selection of drifts could change with a change in layout. A change in waste package spacing could also cause differences. A change in the line load heat source and/or the seven-waste package sequence could affect different parts of the condensation calculations. In addition, changes in repository layout could cause upstream differences in how percolation is handled and percolation binning that affect condensation since condensation is evaluated as a function of percolation.
2.1.08.11.0A	Repository Resaturation Due to Waste Cooling	Following the peak thermal period, water in the condensation cap may flow downward, resaturating the geosphere dryout zone and flowing into the drifts. This may lead to an increase in water content and/or resaturation in the repository.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	Changes in the number of waste packages and the repository layout would change the technical basis for how this FEP was included. Temperature at the drift wall, which is used to determine when seepage occurs, is calculated by multiscale thermal-hydrologic model. Changes to multiscale thermal-hydrologic model could lead to differences in the timing of seepage initiation. However, the thermal analysis in Appendix B-2 suggests that these changes would be very small.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.08.14.0A	Condensation on Underside of Drip Shield	Condensation of water on the underside of the drip shield may affect the waste package hydrologic and chemical environment.	Excluded low consequence	The exclusion justification could be affected by changing the number of waste packages, reducing the number of cooler waste packages, and changes to the repository layout. The in-drift condensation model, which was used to evaluate the FEP, could be affected in different ways by implementing the ABD Program based on what design changes might be made to accommodate the reduction in CDSP waste packages. For example, the condensation calculations are completed for seven drifts within the current repository layout. The selection of drifts could change with a change in layout. A change in waste package spacing could also cause differences. A change in the line load heat source and/or the seven-waste package sequence could affect different parts of the condensation calculations. In addition, changes in repository layout could cause upstream differences in how percolation is handled and percolation binning that affect condensation since condensation is evaluated as a function of percolation.
2.1.09.01.0B	Chemical Characteristics of Water in Waste Package	Chemical characteristics of the water in the waste packages (pH and dissolved species) may be affected by interactions with steel and other materials used in the waste packages or waste forms, as well as by the inflowing water from the drifts and near-field host rock. The in-package chemistry, in turn may influence dissolution and transport as contaminants move through the waste, EBS, and down into the unsaturated zone.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	The pH is not likely to be significantly different because of the ABD glass. The high ionic strength maximum for the liquid influx condition could be affected because of the change in magnesium and calcium content of the ABD glass.
2.1.09.04.0A	Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and EBS	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.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	The lower calcium and magnesium concentrations in the ABD glass may lead to higher fluoride concentrations in the water.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.11.01.0A	Heat Generation in EBS	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 exothermic reactions. Considerations of the heat generated by radioactive decay should take different properties of different waste types, including DOE SNF, into account.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	The seven-waste package sequence, used in multiscale thermal-hydrologic model, was developed to implement a thermal management strategy and to provide a reasonable representation of the relative number of different waste package types in the repository. One result of the removal of ~900 CDSP waste packages is that the sequence will no longer provide a reasonable representation of the relative number of different waste package types in the repository. This difference could be addressed in several different ways, depending on how the design is modified. A reduction in the number of emplacement drifts would lead to a reduction in the number of multiscale thermal-hydrologic model subdomains and an associated redefinition of the percolation subregions. The representative CSNF and CDSP waste packages for the new subregions would have to be determined.
2.1.11.02.0A	Non-Uniform Heat Distribution in EBS	Uneven heating and cooling at edges of the repository may lead to nonuniform thermal effects during both the thermal peak and the cool-down period.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	The seven-waste package sequence, used in multiscale thermal-hydrologic model, was developed to implement a thermal management strategy and to provide a reasonable representation of the relative number of different waste package types in the repository. One result of the removal of ~900 CDSP waste packages is that the sequence will no longer provide a reasonable representation of the relative number of different waste package types in the repository. This difference could be addressed in several different ways, depending on how the design is modified. A reduction in the number of emplacement drifts would lead to a reduction in the number of multiscale thermal-hydrologic model subdomains and an associated redefinition of the percolation subregions. The representative CSNF and CDSP waste packages for the new subregions would have to be determined.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.11.09.0A	Thermal Effects on Flow in the EBS	High temperatures in the EBS may influence seepage into, and flow within, the waste and EBS. Thermally-induced changes to fluid saturation and/or relative humidity could influence in-package chemistry. Thermal gradients in the repository could lead to localized accumulation of moisture. Wet zones could form below the areas of moisture accumulation.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	Potential effects of implementing the ABD Program occur primarily through the potential differences in the multiscale thermal-hydrologic model and the in-drift convection and condensation models. The nature of the effects depends on how the reduction in the number of CDSP waste packages is handled. The multiscale thermal-hydrologic model relies on the seven-waste package sequence, which could be affected. Also, a change in repository layout such as a reduced number of drifts would lead to a reduction in the number of multiscale thermal-hydrologic model subdomains and an associated redefinition of the percolation subregions causing differences in percolation binning. The representative CSNF and CDSP waste packages for the new subregions would have to be determined. The in-drift convection and condensation models could be affected in different ways by implementing the ABD Program based on what design changes might be made to accommodate the reduction in CDSP waste packages. For example, the condensation calculations are completed for seven drifts within the current repository layout. The selection of drifts could change with a change in layout. A change in waste package spacing could also cause differences. A change in the line load heat source and/or the seven-waste package sequence could affect different parts of the convection and condensation calculations. In addition, changes in repository layout could cause upstream differences in how percolation is handled and percolation binning that affect condensation since condensation is evaluated as a function of percolation.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.1.11.09.0C	Thermally Driven Flow (Convection) in Drifts	Temperature differentials may result in convective flow in the EBS. Convective flow within the drifts could influence in-drift chemistry.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	Convective flow in the EBS is evaluated with the in-drift convection model, which generates (1) equivalent thermal conductivities approximating the in-drift configuration for the multiscale thermal-hydrologic model and (2) axial dispersion coefficients for the in-drift condensation model. Neither the multiscale thermal-hydrologic model nor the in-drift condensation model explicitly includes in-drift convection. The 2D convection simulations consider a single waste package with heat source consistent with the 1.45 kW line load. The 3D convection simulations consider a single partial drift (71 m long) filled with two of the seven-waste package sequences. A change in the line load and/or the waste package sequence would affect the heat source input and potentially the length of the drift selected for analysis, which would cause differences in the model output. A change in waste package spacing would also cause differences.
2.2.07.10.0A	Condensation Zone Forms Around Drifts	Condensation of the two-phase flow generated by repository heat may form in the rock where the temperature drops below the local vaporization temperature. Waste package emplacement geometry and thermal loading may affect the scale at which condensation caps form (over waste packages, over panels, or over the entire repository), and the extent to which "shedding" will occur as water flows from the region above one drift to the region above another drift or into the rock between drifts.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	The effects of this FEP are implemented in the TSPA primarily through the multiscale-thermal hydrologic model. It incorporates vapor condensation in the drifts in evaluating near-field and in-drift thermal hydrology. The multiscale thermal-hydrologic model relies on the seven-waste package sequence, which could be affected. Also, a change in repository layout such as a reduced number of drifts would lead to a reduction in the number of multiscale thermal-hydrologic model subdomains and an associated redefinition of the percolation subregions causing differences in percolation binning. The representative CSNF and CDSP waste packages for the new subregions would have to be determined.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.2.07.11.0A	Resaturation of Geosphere Dry-Out Zone	Following the peak thermal period, water in the condensation cap may flow downward into the drifts. Influx of cooler water from above, such as might occur from episodic flow, may accelerate return flow from the condensation cap by lowering temperatures below the condensation point. Percolating groundwater will also contribute to resaturation of the dryout zone. Vapor flow, as distinct from liquid flow by capillary processes, may also contribute.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	This FEP is incorporated into the TSPA primarily by the multiscale thermal-hydrologic model, which provides temperature and relative humidity information in the drift during and after the thermal period. This model could be affected by implementing the ABD Program in different ways depending on the design choices made to deal with the reduction in CDSP waste packages. The modeling grid used to reflect the repository layout could be affected as well as the modeling results and abstraction process. One of the multiscale modeling results that can be affected is the boiling duration at the waste package, which is used in seepage calculations to determine when seepage can begin after the thermal period. Ambient seepage is conservatively assumed even when the system is still cooling down. The ambient seepage calculations, including those for seepage rate, seepage percentage, and seepage fraction, could be also be affected in different ways by implementing the ABD Program depending on the design choices.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.2.10.10.0A	Two-Phase Buoyant Flow/Heat Pipes	Heat from waste can generate two-phase buoyant flow. The vapor phase (water vapor) could escape from the mountain. A heat pipe consists of a system for transferring energy between a hot and a cold region (source and sink respectively) using the heat of vaporization and movement of the vapor as the transfer mechanism. Two-phase circulation continues until the heat source is too weak to provide the thermal gradients required to drive it. Alteration of the rock adjacent to the drift may include dissolution that maintains the permeability necessary to support the circulation (as inferred for some geothermal systems).	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	This FEP is incorporated into the TSPA primarily by the multiscale thermal-hydrologic model, which provides temperature and relative humidity information in the drift during and after the thermal period. This model could be affected by implementing the ABD Program in different ways depending on the design choices made to deal with the reduction in CDSP waste packages. The modeling grid used to reflect the repository layout could be affected as well as the modeling results and abstraction process. The thermal-hydrologic seepage model incorporates heat-pipe behavior, but the results are not used directly in the TSPA, which assumes zero seepage during the thermal period and ambient seepage afterwards. The thermal-hydrologic-mechanical and thermal-hydrologic-chemical seepage models include heat-pipe behavior, but these models are used to conduct studies of possible changes to rock properties due to thermal-hydrologic-mechanical and thermal-hydrologic-chemical processes. The studies concluded that these coupled processes and property changes have no significant impact on seepage. Implementing the ABD Program would not change the study conclusions. Another role of the thermal-hydrologic-chemical seepage model is to corroborate the near-field chemistry model, which provides seepage water compositions to TSPA. However, given that seepage is assumed to be zero during the thermal period, water compositions predicted for the thermal period by any model are not used in the TSPA. Implementing the ABD Program would not change this situation.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

FEP Number[a]	FEP Name	FEP Description	SAR Screening Decision	Evaluation Comments
2.2.10.12.0A	Geosphere Dry-Out Due to Waste Heat	Repository heat evaporates water from the unsaturated zone rocks near the drifts as the temperature exceeds the vaporization temperature. This zone of reduced water content (reduced saturation) migrates outward during the heating phase (about the first 1,000 years) and then migrates back to the waste packages as heat diffuses throughout the mountain and the radioactive sources decay. This FEP addresses the effects of dryout within the rocks.	Included 10 CFR 63.311 (proposed) 10 CFR 63.321 (proposed) 10 CFR 63.331	This FEP is incorporated into the TSPA primarily by the multiscale thermal-hydrologic model, which provides temperature and relative humidity information in the drift during and after the thermal period. This model could be affected by implementing the ABD Program in different ways depending on the design choices made to deal with the reduction in CDSP waste packages. The modeling grid used to reflect the repository layout could be affected as well as the modeling results and abstraction process. The thermal-hydrologic seepage model incorporates dryout behavior, but any effects due to implementing the ABD Program are not significant because the model results only provide a qualitative indication of thermal seepage. The TSPA assumes zero seepage during the thermal period and ambient seepage afterwards. The thermal-hydrologic-mechanical and thermal-hydrologic-chemical seepage models include dryout behavior, but these models are used to conduct studies of possible changes to rock properties due to thermal-hydrologic-mechanical and thermal-hydrologic-chemical processes. The studies concluded that these coupled processes and property changes have no significant impact on seepage. Implementing the ABD Program would not change the study conclusions. Another role of the thermal-hydrologic-chemical seepage model is to corroborate the near-field chemistry model, which provides seepage water compositions to TSPA. However, given that seepage is assumed to be zero during the thermal period, water compositions predicted for the thermal period by any model are not used in the TSPA. Implementing the ABD Program would not change this situation.

Table A-1. List of FEPs Potentially Affected by ABD Program Implementation (continued)

NOTE: ABD = Accelerated Basin Deinventory
ATR = Advanced Test Reactor
CDSP = codisposal (waste package)
CFR = Code of Federal Regulations
CSNF = commercial spent nuclear fuel
DSNF = DOE spent nuclear fuel
EBS = engineered barrier system
FEP = feature, event, and/or process
FFTF = Fast Flux Test Reactor
GM = ground motion
HLW = high-level radioactive waste
LWBR = Light Water Breeder Reactor
MCNP = Monte Carlo N-Particle
PWR = pressurized water reactor
SAR = Safety Analysis Report
SNF = spent nuclear fuel
TMI = Three Mile Island
TRIGA = Training, Research, Isotope, General Atomics
TSPA = total system performance assessment

Source: [a] DOE 2008, Table 2.2-5 for FEP number, name, description, and postclosure PA screening decision.

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APPENDIX B—SUPPLEMENTARY ANALYSES

B-1. ASSESSMENT OF PCT WITH SURROGATE ABD GLASS

At Savannah River National Laboratory, product consistency tests were conducted on glasses made with total fissile loadings of 525 and 2,670 g/m^3 using actual radioactive waste melter feed to measure the effect of a high fissile loading on the glass durability (Crawford et al. 2021). The results of those tests were used to evaluate the suitability of the existing DHLW glass degradation model to represent the high fissile loaded glasses proposed in the ABD Program in repository assessments. The chemical and key isotopic compositions of the nominal and high fissile glasses reported by Crawford et al. (2021) are summarized in Table B-1 (values taken from Tables 5-3 and 5-5 in Crawford et al. [2021]). The key differences between the two glasses that could affect the chemical durability are the higher aluminum content and slightly higher ^{238}U uranium content of the high fissile glass. Other differences are not expected to affect the degradation behavior. The compositions of both glasses are compared with those of several Defense Waste Processing Facility (DWPF) reference glasses and the Environmental Assessment (EA) glass in Table B-2. Of the key glass-forming constituents, the aluminum contents of the nominal and high fissile glasses are higher than in most other DWPF glasses, and the boron and alkaline earth contents are lower. The silica and total alkali metal contents are similar in all glasses.

Table B-1. Chemical and Isotopic Compositions of Nominal and High Fissile Glasses (wt%)

Chemical Composition					
Oxide	**Nominal Glass**	**High Fissile Glass**	**Oxide**	**Nominal Glass**	**High Fissile Glass**
Al_2O_3	6.51	10.14	MnO	3.32	3.46
B_2O_3	4.98	4.64	MoO_3	0.02	0.02
BaO	0.04	0.04	Na_2O	12.70	12.82
CaO	0.67	0.65	NiO	0.71	0.75
CdO	0.01	0.01	P_2O_5	0.22	0.23
Ce_2O_3	0.12	0.12	SiO_2	48.78	48.24
Cr_2O_3	0.06	0.07	SO_4^{2-}	0.31	0.32
CuO	0.04	0.03	SrO	0.02	0.02
Fe_2O_3	10.86	11.40	ThO_2	0.46	0.48
Gd_2O_3	0.04	0.10	TiO_2	0.03	0.03
K_2O	0.05	< 0.07	U_3O_8	1.52	2.41
La_2O_3	0.02	0.02	ZnO	0.02	0.02
Li_2O	3.69	3.50	ZrO_2	0.02	0.06
MgO	0.18	0.19	**Total**	**95.40**	**99.77**
Isotopic Composition					
Isotope	**Nominal Glass**	**High Fissile Glass**	**Isotope**	**Nominal Glass**	**High Fissile Glass**
^{93}Zr	1.32E-02	1.49E-02	^{238}U	1.48E+00	2.13E+00
^{99}Tc	4.34E-04	3.57E-04	^{238}Pu	4.81E-04	4.42E-04
^{232}Th	3.88E-01	3.82E-01	^{239}Pu	4.44E-03	4.54E-03
^{233}U	3.16E-04	2.93E-04	^{240}Pu	4.30E-04	4.86E-04
^{234}U	3.66E-04	1.54E-03	^{241}Pu	1.37E-05	2.98E-05
^{235}U	1.47E-02	9.38E-02	^{242}Pu	7.94E-05	4.70E-05
^{236}U	8.76E-04	7.88E-03	**Total**	**1.90**	**2.64**
^{237}Np	1.36E-03	1.34E-03			

Table B-2. Glass Compositions, in Oxide wt %

Oxide	Nominal Glass	High Fissile Glass	Batch-1[a]	Blend[a]	HM[a]	PUREX[a]	EA[b]
Al_2O_3	6.51	10.14	4.88	4.16	7.15	2.99	3.7
B_2O_3	4.98	4.64	7.78	8.05	7.03	10.33	11.3
BaO	0.04	0.04	0.15	0.18	0.11	0.20	—
CaO	0.67	0.65	1.22	1.03	1.01	1.09	1.12
CdO	0.01	0.01	—	—	—	—	—
Ce_2O_3	0.12	0.12	—	—	—	—	—
Cr_2O_3	0.06	0.07	0.11	0.13	0.19	0.15	—
Cs_2O	—	—	0.06	0.08	0.06	0.06	—
CuO	0.04	0.03	—	—	—	—	—
Fe_2O_3	10.86	11.40	12.84	10.91	7.78	13.25	7.38
Gd_2O_3	0.04	0.10	—	—	—	—	—
K_2O	0.05	< 0.07	3.33	3.68	2.21	3.41	0.06
La_2O_3	0.02	0.02	—	—	—	—	—
Li_2O	3.69	3.50	4.43	4.44	4.62	3.22	4.26
MgO	0.18	0.19	1.42	1.49	1.41	1.41	1.72
MnO	3.32	3.46	2.11	2.08	2.15	2.07	1.34
MoO_3	0.02	0.02	0.08	0.15	0.22	0.08	—
Na_2O	12.70	12.82	9.00	9.13	8.56	12.62	16.8
Nd_2O_3	—	—	0.15	0.22	0.55	0.06	16.8
NiO	0.71	0.75	0.75	0.89	0.41	1.19	0.57
P_2O_5	0.22	0.23	—	—	—	—	—
RuO_2	—	—	0.02	0.03	0.04	0.01	—
SiO_2	48.78	48.24	50.2	51.9	55.8	46.5	48.7
SO_4^{2-}	0.31	0.32	—	—	—	—	—
SrO	0.02	0.02	—	—	—	—	—
ThO_2	0.46	0.48	—	—	—	—	—
TiO_2	0.03	0.03	0.68	0.89	0.56	0.68	0.70
U_3O_8	1.52	2.41	—	—	—	—	—
ZnO	0.02	0.02	—	—	—	—	—
ZrO_2	0.02	0.06	0.10	0.14	0.33	0.05	0.46

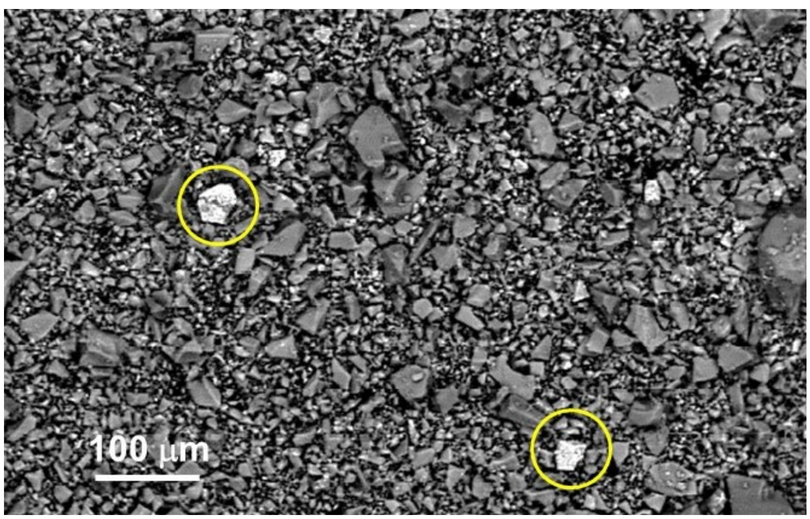
NOTE: EA = Environmental Assessment
HM = high aluminum
PUREX = plutonium and uranium extraction

Source: [a]Composition from Marra et al. 1991.
[b]Composition from Jantzen et al. 1992.

B-1.1 Product Consistency Tests with ABD Glasses

The PCT method was developed specifically to analyze the chemical durability of waste glass using a short-term method that could be conducted within a hot cell. It is a static test performed using a defined size fraction of crushed glass (−100 +200 mesh US standard) and defined glass-to-water mass ratio (10:1) in a sealed steel vessel at 90°C for 7 days. The concentrations of B, Li, and Na measured in the solution are normalized to the mass fractions in the glass to quantify and compare the test responses of different glasses (ASTM C1285-21 Step 25.3). The PCT response of a glass depends on the test conditions and exposed glass surface area. Particles of crushed glass have highly irregular shapes with sharp fracture edges.

A backscattered electron photomicrograph of crushed glass smaller than the size fraction used in the PCTs provided in Crawford et al. (2021) is reproduced as Figure B-1. The two largest bright-contrast particles are circled in yellow, and many smaller bright-contrast particles can be seen. The majority of large- and mid-size particles have darker contrast. The brightness of a particles indicates the average electron density (the average atomic number), and the brighter particles most likely have higher concentrations of fission products. This figure indicates the fission products were not uniformly distributed in the source high fissile glass. Images and composition analyses of the particles used in the PCTs were not provided in the report, but they are expected to contain a similar proportion of bright and dull particles with and without high concentrations of fission products. Similarly, the samples of crushed glass dissolved for composition analyses can only be assumed to have the same proportions.



Source: Crawford et al. 2021, Figure 5-4.

Figure B-1. Backscattered Electron Photomicrograph Showing Crushed Grains Smaller Than Size Fraction Used in PCTs

The shapes and size distribution of particles of crushed high fissile glass are similar to particles of crushed DWPF and other glasses used in tests conducted to parameterize the HLW degradation model. This observation supports the expectation that the fracture behavior of ABD glass logs in standard canisters will be similar to that of HLW glasses addressed by the DHLW glass degradation model. In that model, the combined effects of thermally induced fracturing during production and mechanical fracturing during handling increases the surface area beyond the geometric surface area of the cast glass log by a factor of four. The equation used to track the surface area of glass accessible to seepage water over time in the DHLW glass degradation model is appropriate for use with ABD glass.

The results of four replicate PCTs conducted with the nominal glass (NG-1 through NG-4) are summarized in Table B-3 and the results of four replicate PCTs conducted with the high fissile glass (HF-1 through HF-4) are summarized in Table B-4. The measured elemental concentrations, pH values measured at room temperature, and normalized concentrations *NC*(*i*) were reported in Crawford et al. (2021). The normalized concentrations are calculated by dividing the measured solution concentration of an element by the mass fraction of that element in the glass based on the batched or measured glass composition as follows:

$$NC(i) = \frac{c(i)}{f(i)} \qquad \textit{Equation B-1}$$

where *c*(*i*) is the measured concentration of element *i* in solution and *f*(*i*) is the mass fraction of element *i* in the glass. The mass fractions of B, Li, Na, Si, and U calculated from the reported compositions of the nominal and high fissile glasses are summarized in Table B-5. Note that *NC*(*i*) represents the mass of glass dissolved, not the mass of element *i*, such that values calculated using the concentrations of different elements can be compared directly. Values of *NC*(*i*) are usually expressed in units of grams glass per liter solution. The releases of soluble constituents provide a conservative bound for the releases of less soluble constituents, including radionuclides immobilized in the glass. The release of boron was used to represent the dissolution rate of glasses in tests run to parameterize the HLW glass degradation model. The boron release is used to provide an upper bound to the releases of radionuclides from the glasses. The releases of silicon and uranium in the PCTs are included for comparison.

Table B-3. Results of PCTs with Nominal Glass as Normalized Concentrations and Normalized Elemental Mass Loss

	B (mg/L)	NC(B) (g/L)	NL(B) (g/m^2)	Li (mg/L)	NC(Li) (g/L)	NL(Li) (g/m^2)	Na (mg/L)	NC(Na) (g/L)	NL(Na) (g/m^2)
NG-1	5.66	0.366	0.184	6.69	0.390	0.196	36.50	0.387	0.195
NG-2	6.06	0.392	0.197	7.19	0.419	0.211	39.60	0.420	0.211
NG-3	5.91	0.382	0.192	7.04	0.411	0.206	38.80	0.412	0.207
NG-4	5.86	0.379	0.190	7.04	0.411	0.206	38.10	0.404	0.203
average	—	0.380	0.191	—	0.408	0.205	—	0.406	0.204
std dev	—	0.011	0.005	—	0.012	0.006	—	0.014	0.007

	Si (mg/L)	NC(Si) (g/L)	NL(Si) (g/m^2)	U (mg/L)	NC(U) (g/L)	NL(U) (g/m^2)	pH(RT)
NG-1	61.40	0.269	0.135	2.46	0.191	0.096	10.24
NG-2	64.40	0.282	0.142	2.67	0.207	0.104	10.33
NG-3	63.40	0.278	0.140	2.61	0.202	0.102	10.30
NG-4	62.90	0.276	0.139	2.63	0.204	0.103	10.31
average	—	0.276	0.139	—	0.201	0.101	10.30
std dev	—	0.005	0.003	—	0.007	0.004	0.04

NOTE: NG = nominal glass
std dev = standard deviation

Table B-4. Results of PCTs with High Fissile Glass as Normalized Concentrations and Normalized Elemental Mass Loss

	B (mg/L)	NC(B) (g/L)	NL(B) (g/m²)	Li (mg/L)	NC(Li) (g/L)	NL(Li) (g/m²)	Na mg/L	NC(Na) g/L	NL(Na) g/m²
HF-1	5.00	0.347	0.175	5.90	0.363	0.183	38.70	0.407	0.206
HF-2	5.00	0.347	0.175	5.95	0.366	0.185	39.20	0.412	0.208
HF-3	4.77	0.331	0.167	5.70	0.351	0.177	37.60	0.395	0.200
HF-4	4.98	0.346	0.175	5.95	0.366	0.185	40.10	0.422	0.213
average	—	0.343	0.173	—	0.361	0.183	—	0.409	0.207
std dev	—	0.008	0.004	—	0.007	0.004	—	0.011	0.006

	Si (mg/L)	NC(Si) (g/L)	NL(Si) (g/m²)	U (mg/L)	NC(U) (g/L)	NL(U) (g/m²)		pH(RT)	
HF-1	47.80	0.212	0.107	2.92	0.143	0.072		10.48	
HF-2	47.10	0.209	0.105	2.81	0.138	0.069		10.51	
HF-3	45.20	0.200	0.101	2.74	0.134	0.068		10.35	
HF-4	47.30	0.210	0.106	2.90	0.142	0.072		10.35	
average	—	0.208	0.105	—	0.139	0.070		10.42	
std dev	—	0.005	0.003	—	0.004	0.002		0.08	

NOTE: HF = high fissile (glass)
std dev = standard deviation

Table B-5. Calculated Elemental Mass Fractions in Nominal and High Fissile Glasses

Oxide wt%		
Oxide	Nominal Glass	High Fissile Glass
B_2O_3	4.98	4.64
Li_2O	3.69	3.50
Na_2O	12.70	12.82
SiO_2	48.78	48.24
U_3O_8	1.52	2.41

Elemental Mass Fraction		
Element	Nominal Glass	High Fissile Glass
B	0.01547	0.01441
Li	0.01714	0.01626
Na	0.09422	0.09510
Si	0.22801	0.2255
U	0.01289	0.02044

The solution concentrations generated in a PCT depend on the dissolution kinetics of the glass, time, surface area of glass, and volume of solution. The solution mass and time are fixed in PCT-A (demineralized water is used as the leachant and the density usually estimated to be 1.00 g/cm^3). The size fraction and mass of crushed glass are also fixed, but the surface area of glass depends on the specific surface area of glass particles and density of the glass. The glass surface area is typically calculated by assuming the particles of crushed glass are spherical with a diameter equal to the arithmetic average of the bounding sieve openings for the size fraction and the bulk glass density. Using the calculated geometric surface area is recommended in ASTM C1285 instead of the more accurate Brunauer–Emmett–Teller (BET) surface area because the surface area is used to relate the volume of glass that dissolves to the solution concentration that is generated. For crushed glass in the −100 +200 mesh size fraction used in these PCTs, a diameter of d = 112 µm is used to calculate the specific surface area of a typical particle as

$$S_{sp} = \frac{6}{\rho \cdot d} \qquad \textit{Equation B-2}$$

where ρ is the density of the glass. The surface area of glass available in a test is calculated as the product of the specific surface area and mass of glass added to the test vessel. Therefore, the glass surface area provided by the fixed water/glass mass ratio called for in PCT-A depends on the density of the glass. A glass having a higher density provides less surface area than the same mass of a glass having a lower density. The higher fission product loading in ABD glasses is expected to increase the glass density. Therefore, the effect of the different densities of the nominal and high fissile glasses on the PCT responses were taken into account. The densities of the nominal and high fissile glasses calculated by Crawford et al. (2021) using a composition-based model were reported to be 2.6947 and 2.7055 g/cm^3, respectively. The small effect of the different glass densities on the surface area available in the PCTs is within the testing uncertainty.

The PCT-A water/glass mass ratio specification in ASTM C1285-14 cited in Crawford et al. (2021) was 10.0 ± 0.5 cm^3/g. The specification was revised to 10.0 ± 0.1 cm^3/g in ASTM C1285-21 due to the significant effect on the test response. Crawford et al. (2021) reported that "fifteen milliliters of ASTM-Type I water were added to 1.5 gm of glass," but the exact water/glass mass ratio used in each test was not reported. Therefore, it was assumed that each test was conducted with a water/glass mass ratio of 10.0.

The normalized elemental mass loss values were calculated by using Equation B-3

$$NL(i) = \frac{c(i)}{\left({}^{S}\!/_{V}\right) \cdot f(i)} = \frac{m(i)}{S \cdot f(i)} \qquad \textit{Equation B-3}$$

where *S* is the exposed surface area of glass, *V* is the solution volume, *c*(*i*) is the measured concentration of element *i* in the solution, and *m*(*i*) is the calculated mass of element *i* in solution. Values of *NL*(*i*) are usually expressed in units of grams glass per square meter glass surface area. The *NC*(*i*) and *NL*(*i*) values are related as

$$NL(i) = \frac{NC(i)}{(S/V)} \qquad \text{Equation B-4}$$

The glass dissolution rate equation in the DHLW glass degradation model includes explicit terms for the solution pH and temperature and bounding values representing the effects of the glass and solution composition.

$$rate_G = k_E \, 10^{\eta \, pH} \, exp(-E_a/RT) \qquad \text{Equation B-5}$$

Separate values of the coefficient k_E are used to quantify lower and upper bounds for the effects of the glass composition and solute concentrations in acidic and alkaline solutions. Values of *NL*(B) measured in PCT-A with eight reference HLW glasses were used to determine the maximum value of k_E for glass degradation in alkaline solutions. The average rate determined using *NL*(B) measured in a PCT-A test was used to quantify $rate_G$ as

$$rate_G = NL(B)/7 \equiv NR(B). \qquad \text{Equation B-6}$$

The effects of pH and temperature on the rate were taken into account separately using parameter values of η= 0.49 and E_a = 69 kJ/mol, and values of $\log_{10}(k_E)$ were determined by solving Equation B-5 as

$$log_{10}(k_E) = log_{10}\{NR(B)\} - 0.49 \text{ x } pH - log_{10}\{exp(-69/RT)\} \qquad \text{Equation B-7}$$

The average value of $\log_{10}(k_E)$ determined from replicate PCT-A results for the eight glasses plus two standard deviations (which is $\log_{10}(k_E)$ = 4.54) is used to calculate the dissolution rate in the DHLW glass degradation model. Note that this value is higher than the value determined from results of PCT-A with the EA glass that are used as a benchmark for acceptable HLW glasses; the value for EA glass is $\log_{10}(k_E)$ = 4.18. This result means the model will bound all HLW glasses meeting the Waste Acceptance Product Specifications (WAPS) criteria.

B-1.2 Suitability of HLW Glass Degradation Model for ABD Glasses

Values of $\log_{10}(k_E)$ were calculated from the results of PCT-A conducted with the nominal and high fissile glasses to determine if dissolution rates of those glasses would also be within the range calculated with the HLW glass degradation model. The average value of *NL*(B) for replicate PCT-A with high fissile glass is 0.173 ± 0.004 g/m^2. The mean plus two standard deviations is 0.181 g/m^2. Solving Equation B-7 with pH = 10.42, T = 363 K, and *NL*(B) = 0.181 g/m^2 yields $\log_{10}(k_E)$ = 3.24. This value is lower than the maximum value used in the DHLW glass degradation model and within the uncertainty range. The average value of *NL*(B) for replicate PCT-A with the nominal glass is 0.191 ± 0.005 g/m^2. The mean plus two standard deviations is 0.202 g/m^2. Solving Equation B-7 with pH = 10.30, T = 363 K, and *NL*(B) = 0.202 g/m^2 yields $\log_{10}(k_E)$ = 3.34. This value is also lower than the maximum value used in the DHLW glass degradation model.

B-1.3 Additional Insights

The boron concentrations measured in the PCTs were used to quantify the glass dissolution rates for comparison with the degradation model, but the normalized release values based on other soluble glass constituents provide insights to the high fissile glass dissolution behavior. As is commonly seen in tests with borosilicate waste glasses, the alkali metals Li and Na are also released to a greater extent than boron in the PCTs with the nominal and high fissile glasses and both Si and U are released to a lesser extent. Alkali metals on the glass surface are released through ion exchange reactions with hydronium ion (which drives the solution pH higher) and through hydrolysis reactions with water in neutral and alkaline solutions. Hydrolysis of B−O bonds to release boron occurs faster than the hydrolysis reactions required to release Si or U. The preferential release of alkali metals typically occurs during the first several hours of a PCT then wanes as the surface becomes depleted and alkali metal release becomes limited by mass transport. The increased pH promotes the hydrolysis reactions, but the solution concentrations of released Si and U species become affected by solubility limits within the seven-day test interval and do not provide reliable measures of the extent of glass dissolution. Boron remains highly soluble throughout the test and provides the most representative measure of the glass dissolution kinetics. The relative values of NL(B), NL(Li), NL(Na), NL(Si), and NL(U) determined in PCTs with the nominal and high fissile glasses are consistent with those measured in PCTs with DWPF glasses and other HLW glasses represented by the DHLW glass degradation model.

The solution compositions generated in the PCTs provide additional insights into the effects of ABD glass dissolution on the composition of seepage water in a breached waste package. Table B-6 provides the averages of solution concentrations measured in replicate PCT with each glass, although the concentrations of many constituents were below the analytical detection limits. The most significant result is the surprisingly high aluminum concentration in tests with the high fissile glass. High aluminum concentrations are often found to trigger the precipitation of zeolites that lead to increased glass dissolution rates referred to as Stage III behavior; zeolite formation also requires sufficient dissolved alkali metal and silica concentrations and high pH. The maximum rate in the DHLW glass degradation rate was shown to provide an upper bound to the Stage III dissolution rates of glasses during model development. The PCT method was purposefully developed to generate test solutions approaching saturation by using crushed glass to provide high S/V ratios. Although the goal was to ensure the test response would be related to a characteristic glass property, the resulting solution represents seepage water compositions after long degradation times relevant to anticipated disposal conditions in a confined volume with low seepage flows. The observation that gadolinium concentrations were below the detection limit suggests Gd added to the glass as a neutron absorber is either being retained in the glass or as an insoluble oxide.

Table B-6. Averages of Measured PCT Solution Concentrations (mg/L)

Element	Nominal Glass	High Fissile Glass	Element	Nominal Glass	High Fissile Glass
Ag	<	<	Mn	0.87	0.53
Al	7.60	15.33	Mo	<	<
B	5.87	4.94	Na	38.25	38.90
Ba	<	<	Ni	<	<
Be	<	<	P	0.42	0.62
Ca	<	<	Pb	<	<
Cd	<	<	S	<	<
Ce	<	<	Sb	<	<
Co	<	<	Si	63.03	46.85
Cr	<	<	Sn	<	<
Cu	<	<	Sr	<	<
Fe	3.29	2.00	Th	<	<
Gd	<	<	Ti	<	<
K	<	<	U	2.59	2.84
La	<	<	V	<	<
Li	6.99	5.88	Zn	<	<
Mg	<	<	Zr	<	<

NOTE: "<" = below detection limit

B-2. ASSESSMENT OF THERMAL ASPECTS OF MULTISCALE THERMAL-HYDROLOGIC MODEL

The report *Multiscale Thermohydrologic Model* (SNL 2008b, Section 6.1.5) describes design factors affecting thermal behavior in a repository. These factors include the average areal-heat-generation density of the waste inventory over the heated repository footprint and the average lineal-heat-generation density along the drifts (also called the line-averaged thermal load). According to the report these two parameters constrain the distance between drifts and the size of the required repository footprint. The subsections below examine the potential effect of the reduced CDSP inventory on these two parameters. Other topics addressed include the estimation of the limiting waste string and the postclosure thermal limits specified in the repository design. Note that the analyses in Appendix B-2 focus on the thermal effects of implementing the ABD Program on the multiscale thermal-hydrologic model, with the exception of Appendix B-2.4.3, which considers the role of hydrologic processes with respect to meeting the mid-pillar peak temperature limit. A discussion of how the coupled thermal-hydrologic aspects of the multiscale thermal-hydrologic model or other models with a thermal-hydrologic component may be affected by implementing the ABD Program is provided in Section 3.3.2.

B-2.1 Line-Averaged Thermal Load

B-2.1.1 As implemented in Yucca Mountain License Application

The line-averaged thermal load is defined as total repository heat load divided by total length of the emplacement drift as documented in *Multiscale Thermohydrologic Model* (SNL 2008b, Figure 6.1-1). Thus,

- Total repository heat load of all waste packages = 83,300 kW
- Total length of emplacement drift = 57.48 km
- Initial line-averaged thermal load = 83,300 kW/ 57.48 km = 1.45 kW/m

The TSPA also uses 1.45 kW/m as the initial average line load (SNL 2010a).

Note that the line-averaged thermal load is just a ratio of total heat load to total length of emplacement drift. As a result, the parameter does not involve waste package size, spacing, or staggering of hotter and cooler waste package types. Those issues are addressed further in Appendix B-2.3 and Appendix B-2.4.

B-2.1.2 Possible Changes due to Implementing the ABD Project

As discussed in Section 3.1.1, implementing the ABD Program will result in disposing of about 900 fewer CDSP waste packages, thereby reducing the overall heat load in the repository. For the analysis described below, the 900 CDSP waste packages were assumed to be either 5-DHLW/DOE Short or 5-DHLW/DOE Long. It is also assumed that the contents of a CDSP waste package are the same as in the Yucca Mountain license application (DOE 2008). The length and initial heat generation rate for the CDSP waste packages are given in Table B-7. Removing 900 CDSP waste packages reduces the heat load. Likewise, removing 900 CDSP waste packages reduces the total length of waste packages. To quantify the effect of the reduction in the heat load and the total length of waste packages on the initial line-averaged thermal load, the following scenarios were considered:

- **Constant Emplacement Drift Length—**In this scenario, the emplacement drift length remains the same. The waste packages are therefore spaced further apart than what was originally planned (equivalent to Design Example 2 presented in Section 3.1.2).

- **Reduced Emplacement Drift Length—**In this scenario, the waste packages are placed about 10 cm apart, as originally planned, thereby reducing the emplacement drift length needed to emplace all the waste packages. Note that this scenario represents a different possible design option to accommodate the reduction in CSDP waste packages than the design examples discussed in Section 3.1.2.
- **Maintain Original Line-Averaged Heat Load—**In this scenario, waste packages are arranged such that the original initial line-average thermal load is maintained at 1.45 kW/m. Note that this scenario represents a different possible design option than the design examples discussed in Section 3.1.2.

The results of the analysis for the specified scenarios are shown in Table B-8 and Table B-9 assuming 5-DHLW/DOE Short waste packages and 5-DHLW/DOE Long waste packages, respectively. As shown in Table B-7, the 5-DHLW/DOE Long waste packages are longer than the 5-DHLW/DOE Short waste packages but with much reduced heat load. The results shown in Table B-8 and Table B-9 reflect these differences. When the emplacement drift length is left constant, reducing the heat load for the 5-DHLW/DOE Short case results in a line-average thermal load of 1.39 kW/m (Table B-8). The corresponding line-average thermal load for the 5-DHLW/DOE Long case is 1.44 kW/m (Table B-9). For this scenario using 5-DHLW/DOE Long waste packages has minimum effect on the line-average thermal load. When the emplacement drift is reduced because of the reduction of 900 CDP waste packages, the resulting changes are larger for the 5-DHLW/DOE Long case. Reducing the heat load and drift length for the 5-DHLW/DOE Short case results in a line-average thermal load of 1.48 kW/m (Table B-8). The corresponding line-average thermal load for the 5-DHLW/DOE Long case is 1.57 kW/m (Table B-9). For this scenario using 5-DHLW/DOE Short waste packages has minimal effect on the line-average thermal load. In summary, reducing the emplacement drift length has the greatest effect. However, these changes are within the defined design thermal constraints.

Table B-7. CDSP Waste Packages Inventory and Thermal Data

Waste Package Type	Length (m)	Initial Heat (kW)
5-DHLW/DOE Long	5.2280	0.407
5-DHLW/DOE Short	3.6814	3.620

NOTE DHLW = DOE high-level radioactive waste
DOE = Department of Energy (in this case, shorthand for DSNF or DOE spent nuclear fuel.)

Source: SNL 2008b, Table 6.2-6[a].

Table B-8. Line-average Thermal Load Scenarios for Reduction of 900 CDSP Waste Packages (assuming the reduced waste packages are all 5-DHLW/DOE Short)

Scenario	Total Heat Load (kW)	Reduction in Heat Load (%)	Total Waste Package Length (km)	Reduction in Drift Length (%)	Initial Line-Average Thermal Load (kW/m)
License Application	83,300	—	57.48	—	1.45
Constant Emplacement Drift Length	80,042	3.91	57.48	0	1.39
Reduced Emplacement Drift Length	80,042	3.91	54.17	5.76	1.48
Original Line Average Heat Load	80,042	3.91	55.20	3.97	1.45

Table B-9. Line-Average Thermal Load Scenarios for Reduction of 900 CDSP Waste Packages (assuming the reduced waste packages are all 5-DHLW/DOE Long)

Scenario	Total Heat Load (kW)	Reduction in Heat Load (%)	Total Waste Package Length (km)	Reduction in Drift Length (%)	Initial Line-Average Thermal Load (kW/m)
License Application	83,300	—	57.48	—	1.45
Constant Emplacement Drift Length	82,934	0.44	57.48	0	1.44
Reduced Emplacement Drift Length	82,934	0.44	52.77	8.19	1.57
Original Line Average Heat Load	82,934	0.44	57.2	0.49	1.45

B-2.2 Areal Heat Generation

In addition to the line-averaged thermal load, other parameters such as areal heat generation are also important. The effect of the ABD glass process on this parameter is evaluated below.

B-2.2.1 As Implemented in Yucca Mountain License Application

The areal heat generation is discussed in *Multiscale Thermohydrologic Model* (SNL 2008b, Section 6.2.4). Areal power density is defined as line-average thermal load divided by the drift spacing. Thus,

- Drift spacing = 81 m (SNL 2008b, Table 4.1-1).
- Initial line-averaged thermal load = 1.45 kW/m (SNL 2008b, Figure 6.1-1).
- Initial areal power density = 1.45 kW/m / 81 m = 17.9 W/m^2

The multiscale thermal-hydrologic model (SNL 2008b, Section 6.2.4) provides the calculation sequence and the use of areal mass loading. Based on total drift length of about 58 km and drift spacing of 81 m, the areal mass loading is about 55 MTU/acre. Thus, 55 MTU/acre corresponds to 17.9 W/m^2 (i.e., 1 MTU/acre ~ 0.327 W/m^2). The multiscale thermal-hydrologic model (SNL 2008b, Section 6.2.4) has also shown the effect of using different areal mass loading values (66, 55, 37, 27, 14, and 7).

B-2.2.2 Possible Changes due to ABD Glass Program

The discussion about line-average thermal load in Appendix B-2.1 also applies to areal heat generation analysis. To quantify the effect of the reduction in the heat load and the total length of waste packages on the initial areal power density, the same scenarios as in Appendix B-2.1 were considered. Table B-10 and Table B-11 show the results for areal power density for the scenarios affecting the initial line-average thermal load. As also shown in Appendix B-2.1, reducing the emplacement drift length has the greatest effect.

Varying the drift spacing will also affect the areal power density and the areal mass loading. The changes related to the ABD glass process are within the output of the different areal mass loading values considered.

Table B-10. Areal Heat Generation Scenarios for Reduction of 900 CDSP Waste Packages (assuming the reduced waste packages are all 5-DHLW/DOE Short)

Scenario	Initial Line-Average Thermal Load (kW/m)	Drift Spacing (m)	Areal Power Density (W/m^2)
License Application	1.45	81	17.90
Constant Emplacement Drift Length	1.39	81	17.16
Reduced Emplacement Drift Length	1.48	81	18.27

Table B-11. Areal Heat Generation Scenarios for Reduction of 900 CDSP Waste Packages (assuming the reduced waste packages are all 5-DHLW/DOE Short)

Scenario	Initial Line-Average Thermal Load (kW/m)	Drift Spacing (m)	Areal Power Density (W/m^2)
License Application	1.45	81	17.90
Constant Emplacement Drift Length	1.44	81	17.78
Reduced Emplacement Drift Length	1.57	81	19.38

B-2.3 Estimated Limiting Waste String

The multiscale thermal-hydrologic model (SNL 2008b) uses the seven-waste package sequence to incorporate waste package-to-waste package variability in heat generation into the simulations (Figure B-2). The figure shows the length of the waste packages, waste package spacing, and overall unit cell length. The heat output of each of the seven waste packages is given in SNL (2008b, Table 6.2-6[a]).

The repeating sequence has six full waste packages with a half waste package on either end. The following waste package types are included in the sequence: 21-PWR, 44-boiling water reactor (BWR), 5-DHLW/DOE Short, and 5-DHLW/DOE Long. The 5-DHLW/DOE Short and 5-DHLW/DOE Long represent the hottest and coldest CDSP waste packages, respectively. The heat output of the 5-DHLW/DOE Short is based on loading the waste package with five SRS short glass canisters, and the heat output of the 5-DHLW/DOE Long is based on loading the waste package with five Hanford long glass canisters (DTN: MO0702PASTREAM.001). As discussed in Section 1.4, the thermal output of the ABD glass waste is not expected to be substantially different from the thermal output of the glass that would otherwise have been produced. Thus, implementing the ABD Program does not affect the selection of these two waste packages as the CDSP waste package types to include in the seven-waste package sequence. Note also that, while incorporating waste package-to-waste package heat generation variability is important, the influence on multiscale thermal-hydrologic model results is smaller than the influence of the combined host-rock thermal conductivity–percolation flux uncertainty (SNL 2008b, Section 6.3.16[a]).

The representation (i.e., relative amounts) of the different waste package types in the seven-waste package sequence was selected based on the inventory of the waste string. For example, representation of the CDSP waste packages is shown below.

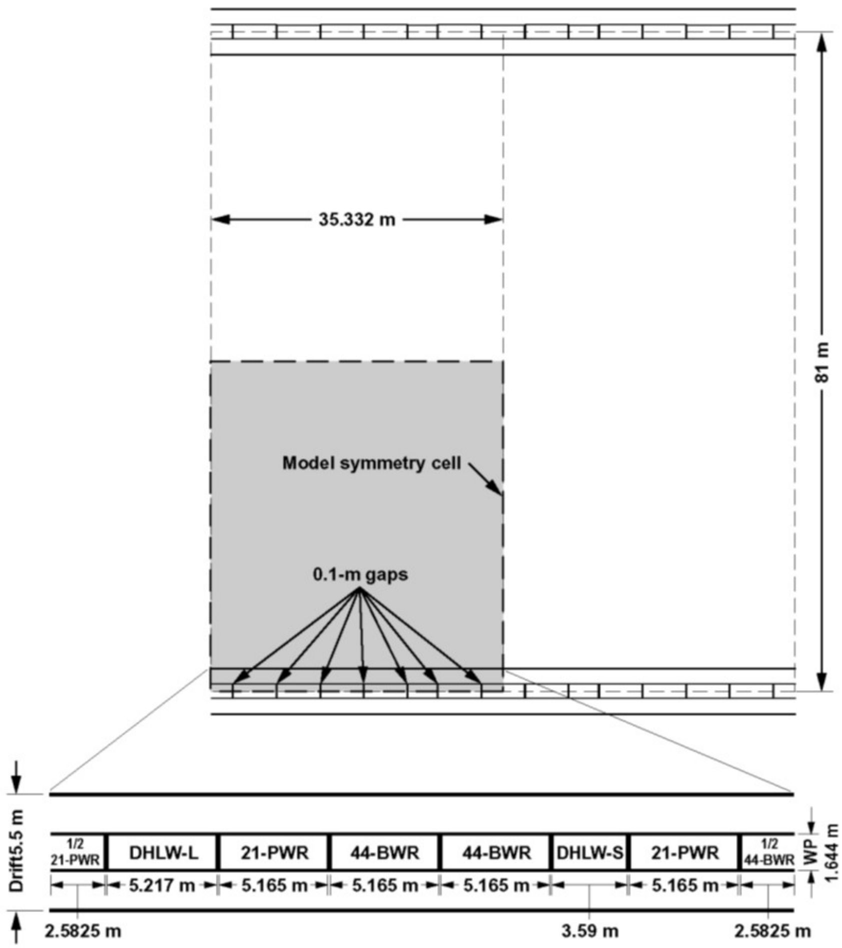
- Inventory of CDSP waste packages (excluding DSNF MCO): 5-DHLW long (1940) + 5-DHLW short (1,257) = 3,197
- Total number of all canisters (excluding DSNF MCO) = 10,820

Thus, the percentage of CDSP waste packages in the inventory is (3,197/10,820) × 100 = 29.55%. This value is equivalent to the representation of the two 5-DHLW/DOE waste packages in the seven-waste package sequence (i.e., 2/7 × 100 = 28.57%).

For the ABD glass analysis, a reduction of 900 CDSP waste packages reduces the inventory of CDSP to 2,297 (i.e., 3,197 – 900). The new percentage of CDSP waste packages in the inventory is the following: 2,297/(10,820 − 900) × 100 = 23.16%. The reduced percentage of CDSP waste packages in the inventory means that CDSP waste packages are slightly over-represented in the seven-waste package sequence. This outcome would potentially reduce the thermal load of the limiting waste string. Alternatively, the number of waste package types represented in the limiting waste string could be changed to maintain the thermal load. One such exercise is shown below. Note that this exercise is for illustration purposes only.

Table B-12 includes the multiscale thermal-hydrologic model calculations to estimate representative waste packages in the limiting waste string that give initial line-averaged thermal load of about 1.45 kW/m. A similar analysis was done for the ABD glass case that maintains the initial line-averaged thermal load of about 1.45 kW/m. Table B-13 shows an alternative arrangement of waste packages for the case with 900 fewer CDSP waste packages. In this example the limiting waste string contains 21 waste packages, with the quantity of each waste type as shown in the table. Thus, the initial line-averaged thermal load of about 1.45 kW/m is maintained with this new waste string. The length of the new limiting waste string is about 120 m. This value differs from the length of the original limiting waste string (i.e., about 39 m over 7 waste packages). Using an alternative limiting waste string may have implications for downstream users of the results of the multiscale thermal-hydrologic model. However, as discussed in Appendix B-2.1 and Appendix B-2.4, limited variations in the initial line-averaged thermal load would have minimal effect.

Both the original and alternative limiting waste string calculations assumed a waste package spacing of 0.1 m. Different waste package spacing values could be considered for the alternative calculations. Such calculations would result in different arrangement of the waste package types in the waste string. That choice would also have implications for downstream use of the results. An analysis of increased waste package spacing is given in Appendix B-2.4.2.



NOTE: BWR = boiling water reactor
DHLW-L = DOE high-level radioactive waste long (shorthand version of 5-DHLW/DOE Long)
DHLW-S = DOE high-level radioactive waste short (shorthand version of 5-DHLW/DOE Short)
PWR = pressurized water reactor
WP = waste package

Source: SNL 2008b, Figure 6.2-2.

Figure B-2. Diagram Showing Drift Spacing, Waste Package Lengths, and Waste Package Spacing in Plan View Considered in the Multiscale Thermal-Hydrologic Model Calculations for the TSPA Base Case

Table B-12. Representative Waste Packages Included in the Limiting Waste String Calculations

Waste Package Type	Inventory	% in Inventory	Number Represented	% Represented	Unit Length (m)	Unit Initial Heat (kW)	Total Length (m)	Total Initial Heat (kW)
21-PWR TAD	4586	42.38	3	42.86	5.8501	12.33	17.55	36.99
44-BWR TAD	3037	28.07	2	28.57	5.8501	7.704	11.70	15.408
5-DHLW/DOE Long	1940	17.93	1	14.29	5.2280	0.407	5.228	0.407
5-DHLW/DOE Short	1257	11.61	1	14.29	3.6814	3.62	3.6814	3.62
Total	10820	100	7	100.01			38.86	56.425

NOTE: Initial average line load: 56.425 kW/38.86 m = 1.452 kW/m
BWR = boiling water reactor
DHLW = DOE high-level radioactive waste
PWR = pressurized water reactor
TAD = transportation, aging, and disposal

Source: SNL 2008b, Table 4.1[a] and 6.2-6[a].

Table B-13. Representative Waste Packages Included for an Alternative Limiting Waste String for ABD Glass Analysis

Waste Package Type	Inventory	% in Inventory	Number Represented	% Represented	Unit Length (m)	Unit Initial Heat (kW)	Total Length (m)	Total Initial Heat (kW)
21-PWR TAD	4586	46.23	10	47.62	5.8501	12.33	58.501	123.3
44-BWR TAD	3037	30.61	6	28.57	5.8501	7.704	35.10	46.22
5-DHLW/DOE Long	1940	19.56	4	19.05	5.2280	0.407	20.91	1.63
5-DHLW/DOE Short	357	3.60	1	4.76	3.6814	3.62	3.68	3.62
Total	9920	100	21	100			120.295	174.77

NOTE: Initial average line load: 174.77 kW/120.295 m = 1.453 kW/m
BWR = boiling water reactor
DHLW = DOE high-level radioactive waste
PWR = pressurized water reactor
TAD = transportation, aging, and disposal

B-2.4 Postclosure Thermal Limits

The postclosure temperature limits are the following:

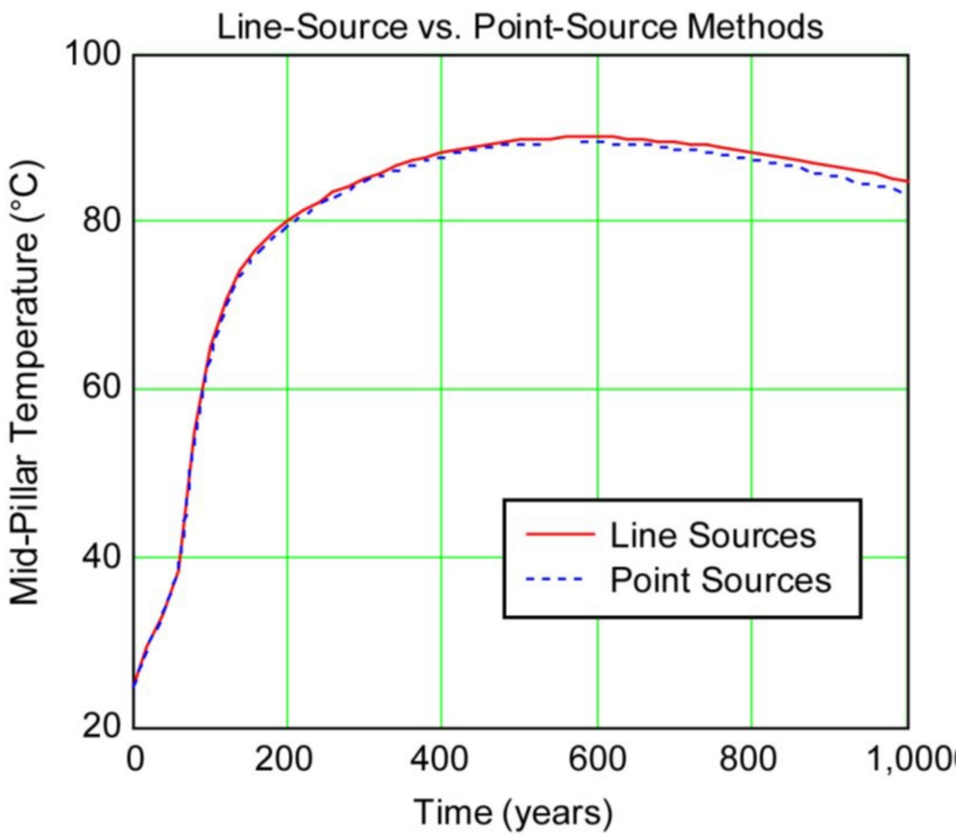
- Mid-pillar temperature limit of 96°C (approximate boiling temperature of water at the repository elevation) to facilitate drainage of percolation water and condensate through the repository horizon. This value of the mid-pillar temperature limit is documented in *Yucca Mountain Project Conceptual Design Report* (DOE 2006, Section 4.6.5). The mid-pillar limit is used in evaluating unsaturated zone FEPs (DTN: MO0706SPAFEPLA.001) and is called out in *Postclosure Modeling and Analyses Design Parameters* (BSC 2008e, Table 1, Parameter 05-03). The purpose of the mid-pillar limit is to preserve pathways for drainage of percolation flux and condensate between every pair of adjacent drifts in the repository. The 96°C limit corresponds to the approximate boiling temperature for water at the repository elevation.
- Peak postclosure drift wall temperature of 200°C to limit thermomechanical effects on drift opening stability
- Waste package outer wall temperature limit of 300°C for 500 years, followed by 200°C for 9,500 years, to reduce Alloy 22 corrosion from certain metallurgical processes
- Maximum CSNF cladding temperature of 350°C to limit degradation of cladding integrity due to thermal creep rupture

Among all the postclosure temperature limits, the mid-pillar temperature limit poses the most significant constraint on thermal loading (SNL 2007l, Section 7; see Section 6.3 for discussion of limiting temperature criteria). Thus, for this analysis, only the peak mid-pillar temperature is considered.

The *Postclosure Analysis of the Range of Design Thermal Loadings* report (SNL 2008c) provides an analysis of mid-pillar temperature behavior. For the analysis an analytical solution was used based on a thermal decay function fitted to the average line load decay history for the postclosure thermal reference case (DTN: MO0705SUPPCALC.000, folder: \Other Supporting Files, file: *Reference Line Load Fit.xls*).

The mid-pillar thermal analysis uses the line-averaged thermal load, given in *Initial Radionuclide Inventories* (SNL 2007g, Table 7-5[a]). SNL (2008c, Section 6.1.3) gives a thermal analysis of the mid-pillar temperature using the line-averaged thermal load and other relevant input. Figure B-3 shows part of the results. The peak mid-pillar temperature is within the 96°C limit. As shown in Appendix B-2.1.2, implementing the ABD Project would affect the line averaged thermal load (Table B-8 and Table B-9). However, the changes are within acceptable limits as discussed in Appendix B-2.4.1. Thus, changes to the mid-pillar temperature would be minimal.

The above analysis was done using thermal-only models. Thermal-hydrologic models would reduce the mid-pillar temperature as shown in Appendix B-2.4.2.



Source: SNL 2008c, Figure 6.1-6.

Figure B-3. Comparison of Mid-Pillar Temperature Histories Calculated Using Line-Source and Point-Source Methods

B-2.4.1 Use of Higher Line-Averaged Thermal Load

The *Post-closure Analysis of the Range of Design Thermal Loadings* report (SNL 2008c) describes 3D thermal analysis that was originally developed in *Repository Twelve Waste Package Segment Thermal Calculation* (BSC 2006b). The purpose of the analysis is to produce high-resolution estimates of the peak postclosure temperatures of the drift wall and waste package, for comparison to the respective 200°C and 300°C limits. The analysis approach uses ANSYS v8.0, a finite-element simulator, to implement thermal conduction in solids, and thermal radiation across the air spaces between the waste package and drip shield and between the drip shield and the drift wall.

For this exercise two calculation cases reported in BSC (2006b) are used to show the effect of raising the line-averaged thermal load from 1.45 kW/m to 1.75 kW/m.

B-2.4.1.1 Case 1 (Base Case) Description

Case 1 involves a line-averaged thermal load of 1.45 kW/m with a maximum waste package heat load of 11.8 kW and a preclosure ventilation time of 50 years. It is run to a maximum time of 10,000 years.

For this case, the composition of the 12-waste package segment is determined based on relative percentages of the total waste package population, as shown in BSC (2006b, Table 4). As noted earlier, based on the percentages shown in BSC (2006b, Table 4), the segment consists of four and a half 21-PWRs, three 44-BWRs, two and a half 5-DHLW Longs, one 5-DHLW Short, and one “other” type. In this case, the “other” type is chosen to be the 12-PWR waste package because it is relatively hot. BSC (2006b, Table 39 and Figure 4) provides the waste package emplacement order, initial heat outputs, and lengths modeled. Lengths are taken from Table 5 of BSC (2006b). Heat output for all waste packages is given in in BSC (2006b, Table 33).

For the postclosure period the results show that the peak waste package and drift wall temperatures are 193°C and the 160°C, respectively. These values are lower than the limits set for each location.

B-2.4.1.2 Case 6 Description

Case 6 has the same configuration as Case 1. The heat loads are uniformly raised to produce a line-averaged thermal load of 1.75 kW/m. The calculation of line-averaged thermal load can be found in Attachment IV, file: calulation_cases.xls in BSC (2006b, Table 40), which provides the waste package emplacement order, initial heat outputs, and lengths modeled for Case 6.

As expected, temperatures are generally hotter than those in Case 1. In the postclosure period, waste package surface and drift wall temperatures are both up to 32°C warmer compared to Case 1 temperatures. The Case 6 values are still lower than the limits set for each location. Thus, for this analysis, raising the line-averaged thermal load from 1.45 kW/m to 1.75 kW/m resulted in peak temperatures below the specified thermal limits for the waste package and the drift wall.

B-2.4.2 Effect of Changing Drift Spacing and Waste Package Spacing

The 3D thermal analysis reported in *Repository Twelve Waste Package Segment Thermal Calculation* (BSC 2006b) also includes cases in which the drift spacing is reduced from 81 m to 75 m (Case 13) and the waste package spacing is increased from 0.1 m to 0.5 m (Case 20), with everything else being the same as the base case (Case 1). The results of the reduced drift spacing case (Case 13) show temperatures up to 2°C warmer at the waste package surfaces and up to 4°C warmer at the drift wall surfaces when compared to the base case (Case 1). The results of the increased waste package spacing case (Case 20) showed that predicted temperatures are cooler than the base case (Case 1) temperatures. For Case 20, waste package surface temperatures are up to 12°C cooler and drift wall temperatures are up to 10°C cooler.

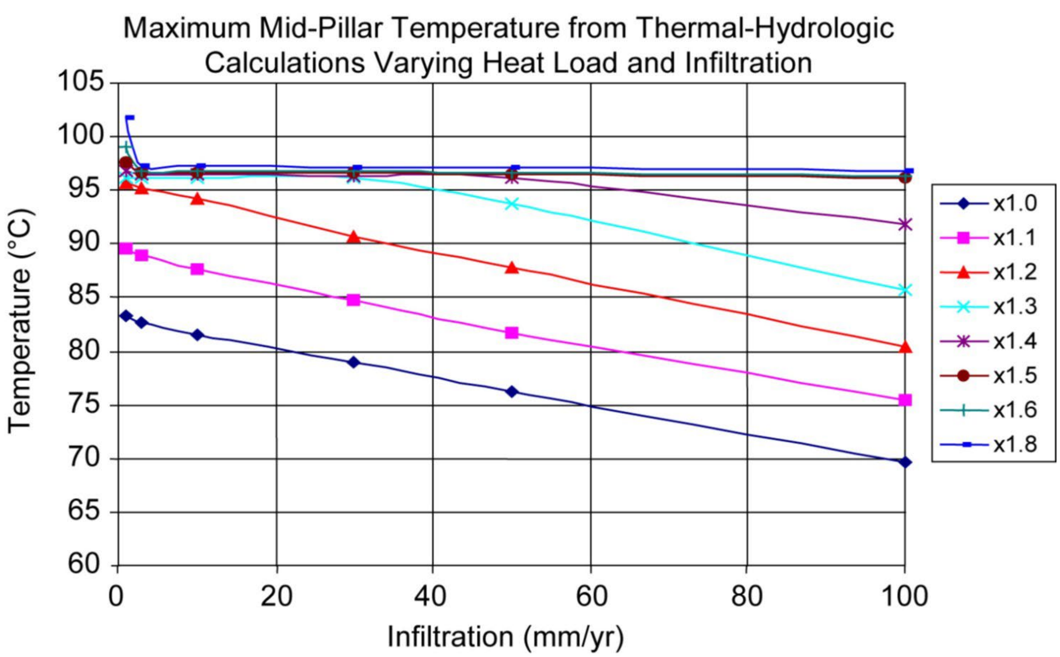
B-2.4.3 Thermal-Hydrologic Margin on Mid-Pillar Peak Temperature from Hydrologic Effects

The analyses described in Appendix B-2.4, Appendix B-2.4.1, and Appendix B-2.4.2 were developed for thermal conduction only. Consideration of thermal-hydrologic effects provides additional benefits in lowering thermal conditions. SNL (2008c, Section 6.2) describes an analysis of the effect of thermal hydrology. It evaluates cases that provide additional assurance that the mid-pillar temperature limit will be met. The cases are (1) heat dissipation effects at the edge of the repository layout, and (2) the hydrological effects of percolation flux. The analysis to determine coupled thermal-hydrologic effects is summarized below.

A series of thermal-hydrologic simulations was performed varying thermal line loads and infiltration rates parametrically to evaluate how much additional thermal loading can be accommodated while maintaining the mid-pillar temperature at the 96°C limit. Thermal loading was multiplied by factors of 1.0, 1.1, 1.2, 1.3, 1.4, 1.5, 1.6, 1.8, and 2.0. Percolation flux was set to 1 mm/yr, 3 mm/yr, 10 mm/yr, 30 mm/yr, 50 mm/yr, and 100 mm/yr. All possible combinations of these settings were run (DTN: MO0707THERMHYD.000). Additional simulations evaluated the effect of decreased host-rock thermal conductivity. Comparisons between conduction-only and thermal-hydrologic simulations show the effect of percolation flux on peak mid-pillar temperature through a series of mid-pillar temperature histories. As the flux increases, mid-pillar temperature is lowered, and the temperature differences between conduction-only and thermal-hydrologic simulations increase because of the temperature hold-up near 96°C.

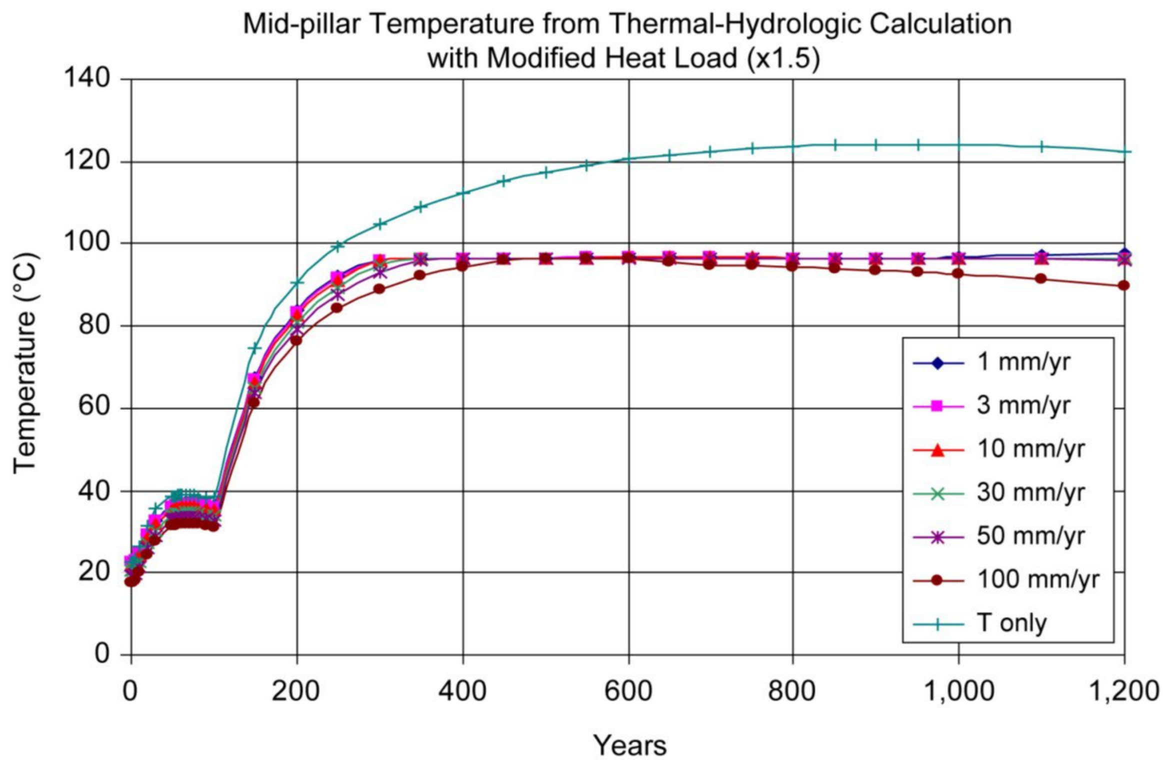
The relationship between peak mid-pillar temperature, infiltration rate, and thermal loading is summarized in Figure B-4 (for host-rock wet thermal conductivity of 1.89 W/[m·K]).

In summary, these results show that there is substantial margin to meet the mid-pillar temperature limit, for infiltration flux of 1 mm/yr and greater. Figure B-5 shows mid-pillar temperature predictions for conduction-only and thermal-hydrologic conditions for the case of 1.5 times the heat load. The results demonstrate the capability of thermal-hydrologic processes to limit mid-pillar temperature.



Source: DTN: MO0707THERMHYD.000, file: *max_min_temp.xls*; SNL 2008c, Figure 6.2-12.

Figure B-4. Summary of Peak Mid-Pillar Temperature as a Function of Thermal Load and Percolation Flux



Source: DTN: MO 0707THERMHYD.000, file: *TH1.5_P30.xls*; SNL 2008c, Figure 6.2-8.

Figure B-5. Mid-Pillar Temperature for 1.5 Times the Base Case Thermal Load, and Increasing Values of Percolation Flux

B-2.5 Summary

The analyses described in Appendix B-2 focus primarily on the thermal effects of disposing of ABD glass waste in a generic repository similar to Yucca Mountain. It is assumed that implementing the ABD Program would result in about 900 fewer CDSP short waste packages. That reduction would affect the thermal aspects of the analyses reported in the multiscale thermal-hydrologic model (SNL 2008b) and other reports, as well as downstream users. Simplified calculations were made examining the potential effects of implementing the ABD Program on the line-averaged heat load, the areal heat generation, the estimation of the limiting waste string, and the ability to meet the postclosure thermal limits. The CDSP waste packages being reduced were assumed to be either all 5-DHLW/DOE Short or all 5-DHLW/DOE Long. The results of the calculations can be summarized as follows:

- The thermal effects due to changes to the line-averaged thermal load are within the defined thermal constraints.
- The thermal effects due to changes to the areal power density and the areal mass loading are within defined thermal constraints.
- Reduction of both the 5-DHLW/DOE Short and 5-DHLW/DOE Long waste packages would affect the makeup of the waste packages represented in the limiting waste string. An alternative waste string is shown in Table B-13 for illustration purposes.

- Previous thermal calculations described in Appendix B-2.4 indicate that the postclosure thermal limits would be met.
- The postclosure thermal limits analyses described in Appendix B-2.4 and Appendix B-2.4.1 were developed for thermal conduction only. The addition of hydrologic processes provides increased benefits with respect to lowering temperatures. SNL (2008c, Section 6.2) describes an analysis that evaluates cases providing additional assurance that the mid-pillar temperature limit will be met. The cases are (1) heat dissipation effects at the edge of the repository layout, and (2) the hydrological effects of percolation flux. These results showed that there is substantial margin to meet the mid-pillar temperature limit for infiltration flux of 1 mm/yr and greater.

APPENDIX C—BASIS FOR CONCLUSION REGARDING GENERIC REPOSITORY SIMILAR TO YUCCA MOUNTAIN

The discussion below provides a semiquantitative basis for the conclusion that, for a repository similar to that proposed at Yucca Mountain, implementing the ABD Program might affect the technical basis for the license application but would not change the conclusion that a generic repository similar to Yucca Mountain would meet regulatory requirements. The basis relies on (1) the margin between modeled results and regulatory limits, (2) the contribution to modeled quantitative results from radionuclides disposed of in CSNF waste packages versus the contribution from radionuclides disposed of in CDSP waste packages, and (3) the radionuclides that contribute most to dose.

C-1. MARGIN BETWEEN MODELED RESULTS AND REGULATORY LIMITS

The NRC adopted three postclosure radiation protection standards for the Yucca Mountain repository:

1. The individual protection standard after permanent closure (10 CFR 63.311), which considers dose to a reasonably maximally exposed individual for two time periods: 10,000 years and 1,000,000 years. Calculations demonstrating compliance with this standard must consider all events and processes that have more than one chance in 100,000,000 per year of occurring.
2. The individual protection standard for human intrusion (10 CFR 63.321), which considers dose to a reasonably maximally exposed individual in the event of human intrusion into a waste package, for the same two time periods. Calculations demonstrating compliance with this standard must consider all events and processes that have at least one chance in 100,000 per year of occurring.
3. Separate standards for protection of groundwater (10 CFR 63.331), which limit concentrations of and annual dose from certain radionuclides in a representative volume of groundwater for 10,000 years. Calculations demonstrating compliance with this standard must consider all events and processes that have at least one chance in 100,000 per year of occurring.

Table C-1, Table C-2, and Table C-3 give the performance demonstration results for each of these three requirements. What is apparent from the data in these tables is that, in most cases, the performance objective limits are met with a substantial margin; the projected peak mean annual dose in Table C-1 for the 10,000-year regulatory period is 0.24 mrem, which is ~60 times below the limit of 15 mrem and the projected peak median annual dose is 0.99 mrem, which is ~350 times below the limit of 350 mrem. The smallest margin between projected performance results and the regulatory limit is a factor of ~24, in the performance measure of dose to the thyroid (~0.17 mrem versus 4 mrem, Table C-3). Therefore, the performance measure limits will still be met with a large margin since implementing the ABD Program is not anticipated to result in significant changes to dose based on the analyses presented in the main body of this report and the discussion below.

Table C-1. Performance Demonstration Results for Individual Protection Standard

Time After Closure (year)	Projected Peak Mean Annual Dose (mrem)	Time of Peak Mean Annual Dose (year)	Projected Peak Median Annual Dose (mrem)	Time of Peak Median Annual Dose (year)	Limit for Annual Dose (mrem)
10,000	0.24	10,000	0.12	10,000	15 (mean)
1,000,000	2.30	1,000,000	0.99	~760,000	350 (median)

Source: SNL 2010a, Table 8.1-1.

Table C-2. Performance Demonstration Results for Human Intrusion Standard with Drilling Event at 200,000 years after Closure

Time After Closure (year)	Projected Peak Mean Annual Dose (mrem)	Limit for Annual Dose (mrem)
10,000	0	15 (mean)
1,000,000	$< 10^{-2}$	350 (median)

Source: SNL 2010a, Table 8.1-3.

Table C-3. Performance Demonstration Results for Groundwater Protection Standard

Type of Limit	Projected Peak mean Activity Concentration or Annual Dose	Natural Background Level	Limit for Activity Concentration or Annual Dose
Combined ^{226}Ra and ^{228}Ra	$<10^{-5}$ pCi/L	0.5 pCi/L	5 pCi/L
Gross Alpha Activity	$<10^{-4}$ pCi/L	0.5 pCi/L	15 pCi/L
Dose from combined Beta and Photon-Emitting Radionuclides	Whole Body ~ 0.04 mrem Thyroid ~0.17 mrem	Background level excluded in regulatory requirement	4 mrem

Source: SNL 2010a, Table 8.1-2.

The TSPA analyses (SNL 2010a) include several modeling cases, as discussed in Section 3.3.3. However, the total dose results are dominated by two modeling cases: the igneous intrusion modeling case and the seismic ground motion modeling case. The contribution to total dose of all the other modeling cases combined is approximately 2% (SNL 2010a, Section 8.2). The seismic ground motion modeling case contributes about 71% and the igneous intrusion modeling case contributes about 27% to the total mean annual dose for the 10,000-year calculations. For most of the 1,000,000-year calculations, the seismic ground motion modeling case and the igneous intrusion modeling case contribute almost equally to the peak total mean annual dose (SNL 2010a, Section 8.2).

It is also important to note the failure rates for each type of waste package (CSNF and CDSP) and waste form for each modeling case. In the igneous intrusion modeling case, all waste packages, whether CSNF or CDSP, are assumed to fail when the event occurs (SNL 2010a, Section 5.3.1). In addition, in the igneous intrusion modeling case, both the CSNF waste form and the HLW glass waste form are assumed to degrade instantaneously because of the igneous intrusion (SNL 2010a, Section 6.3.7). It should be noted that for all modeling cases, DSNF, which is in CDSP waste packages along with HLW glass, is assumed to degrade instantaneously such that radionuclides are available for transport, subject to solubility constraints, as soon as the CDSP waste package fails (SNL 2010a, Section 6.3.7.4.2). In contrast, in the seismic ground motion modeling case, because CDSP waste packages have a higher frequency of seismic-induced failure than do the CSNF waste packages prior to 10,000 years, expected annual doses for the 10,000-year time period are only from radionuclides modeled as being released from CDSP waste packages (SNL 2010a, Section 8.1.1.2). Therefore, the expected annual dose for the seismic ground motion modeling case for the 10,000-year time period is more sensitive to changes in the HLW glass radionuclide inventory and composition than it is for the igneous intrusion modeling case for both time periods or for the seismic ground motion modeling case for the post-10,000-year regulatory period.

C-2. RADIONUCLIDE INVENTORY IN CSNF VERSUS CDSP WASTE PACKAGES

For most radionuclides, the inventory of that radionuclide in a CSNF (including naval SNF) waste package is about 10 times greater than the inventory of that same radionuclide in a CDSP waste package (SNL 2010a, Table 6.3.7-5). In addition, the number of CSNF (including naval SNF) waste packages to be emplaced is about 2.4 times the number of CDSP waste packages to be emplaced (Table 3-1 and Table 3-2). Therefore, with the exception of the seismic ground motion modeling case for the 10,000-year regulatory period, most (> 90%) of the radionuclides in the TSPA model that are available for transport to the accessible environment and that can contribute to maximum average dose or another performance metric are from CSNF waste packages. Implementing the ABD Program would increase the inventory of some uranium isotopes in some canisters of HLW glass (Crawford et al. 2021, Table 5-5) but would reduce the number of CDSP waste packages by about 900 (Section 3.1.1), resulting in about 3.3 times as many CSNF waste packages being emplaced as CDSP waste packages. In addition, implementing the ABD Program would not result in a significant change in the radionuclide inventory of the DSNF that is disposed with the glass HLW in CDSP waste packages.

C-3. RADIONUCLIDES THAT CONTRIBUTE MOST TO DOSE FOR THE VARIOUS REGULATORY LIMITS

C-3.1 Individual Protection Standard (Table C-1)

For the igneous intrusion modeling case, most of the radionuclides modeled as contributing to the performance metrics would still be from CSNF waste packages because both CSNF and CDSP waste packages are assumed to fail and both the CSNF waste form and the HLW glass waste form are assumed to degrade instantly when the igneous event occurs. Therefore, implementing the ABD Program could have only a minimal effect on the performance metrics considered in the TSPA in the modeling cases for which both radionuclides released from CSNF waste packages and radionuclides released from CDSP waste packages contribute to the dose. That is, implementing the ABD Program could have only a minimal effect on the total mean annual dose for the 1,000,000-year regulatory period and for the total mean annual dose for the 10,000-year regulatory period as calculated for the igneous intrusion modeling case because radionuclides released from CSNF waste packages would still dominate (>90%) modeled releases.

As noted above, in the seismic ground motion modeling case, expected annual doses for the 10,000-year time period are primarily from radionuclides modeled as being released from CDSP waste packages (SNL 2010a, Section 8.2.4.1). The three radionuclides that account for about 99.9% of the peak of the total mean annual dose for 10,000 years after closure are ^{99}Tc, ^{14}C, and ^{129}I (SNL 2010a, Figure 8.2-12).

Implementing the ABD Program would decrease the number of CDSP waste packages emplaced but would not change the quantities of these three radionuclides per canister of HLW glass beyond the uncertainty and variability already included in TSPA (Section 3.3.2.4; Crawford et al. 2021), and it would not change the inventory of these radionuclides in DSNF disposed with the glass HLW in CDSP waste packages. In addition, possible changes in colloidal radionuclide stability and transport or radionuclide solubilities resulting from implementing the ABD Program (Section 3.3.2.5) also would not have a significant effect on dose because Tc, C, and I are not transported via colloids and releases of Tc, C, and I from the waste package are not solubility limited (SNL 2010a, Section 8.1.1.5). Therefore, any dose changes that could result from implementing the ABD Program will be minimal for this modeling case and time period as well.

For the seismic ground motion modeling case for the 1,000,000-year regulatory period, ^{99}Tc and ^{129}I account for about 90% of the mean annual dose for about 800,000 years, after which ^{242}Pu and ^{237}Np begin to contribute more significantly to dose (SNL 2010a, Figure 8.2-12). However, these late contributors are primarily from CSNF waste packages because CDSP waste packages have a higher frequency of seismic-induced failure and tend to fail and release radionuclides earlier in the simulation while seismic-induced CSNF waste package failure tends to occur later in the simulation (SNL 2010a, Figure 7.7.1-49). Implementing the ABD Glass Program would decrease the number of CDSP waste packages, would not change the quantities of ^{99}Tc and ^{129}I per CDSP waste package, would not affect the release and transport of these radionuclides, and would not affect release from CSNF waste packages. Therefore, any dose changes that could result from implementing the ABD Program would be minimal for this modeling case and time period.

C-3.2 Human Intrusion Standard (Table C-2)

The calculations demonstrating compliance with the human intrusion standard consisted of a stylized scenario that includes the assumption that a driller would drill a borehole directly through a degraded waste package. Implementing the ABD Program would affect the calculations for this standard as described in Section 3.3.3.6. However, as shown in Table C-2, there is a substantial margin between the projected peak mean annual dose and the regulatory limit; the slight changes described in Section 3.3.3.6 would have a minimal effect on this margin.

C-3.3 Groundwater Protection Standard (Table C-3)

The calculations demonstrating compliance with the groundwater protection standard included the nominal modeling case, the waste package early failure modeling case, the drip shield early failure modeling case, and the seismic ground motion modeling case (with only PGVs with an annual frequency greater than 10^{-5} per year). The time period for this standard is 10,000 years. In addition, projections for this standard are based on considering releases only from the damaged CDSP waste packages (SNL 2010a, Section 8.1.2). Implementing the ABD Program would affect the results of the calculations designed to demonstrate compliance with the combined ^{226}Ra and ^{228}Ra concentration limit in that it would reduce the number of CDSP waste packages, it would increase the weight percent of ^{238}U (one parent of ^{226}Ra) in some of the glass HLW by about 45%, it would increase the weight percent of ^{234}U (another parent of ^{226}Ra) in some of the glass HLW by about 320%, it would decrease the weight percent of ^{232}Th (the parent of ^{228}Ra) in some of the glass waste by about 2% (Crawford et al. 2021, Table 5-5), and it would not change the radionuclide inventory of DSNF that is disposed with the glass HLW in CDSP waste packages. These small changes would have an insignificant effect on the margin (over 5 orders of magnitude) between the projected concentrations of ^{226}Ra and ^{228}Ra and the concentration limit shown in Table C-3. This same line of reasoning also applies to the limit on gross alpha activity;

implementing the ABD Program would result in inventory changes that are minuscule compared to the margin between projected gross alpha activity and the limit on gross alpha activity shown in Table C-3.

With respect to the limit on dose from combined beta- and photon-emitting radionuclides, the radionuclide that dominates mean thyroid dose is ^{129}I, while ^{99}Tc dominates the mean whole-body dose (SNL 2010a, Section 8.1.2.3). Implementing the ABD Program would not change the inventory of either of these radionuclides per CDSP waste package, but it would decrease the number of CDSP waste packages emplaced. Therefore, implementing the ABD Program could cause a slight decrease in the projected dose for this standard.

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APPENDIX D—BASIS FOR DETERMINING HEAVY METAL LOADING IN HLW GLASS CANISTERS

The discussion in this appendix is adapted from Volume II of *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 2002) and *Options for Determining Equivalent MTHM for DOE High-Level Waste* (Knecht et al. 1999).

In 1985, DOE published a report in response to Section 8 of the Nuclear Waste Policy Act (of 1982) that required the Secretary of Energy to recommend to the President whether DHLW should be disposed of in a geologic repository along with CSNF. That report, *An Evaluation of Commercial Repository Capacity for the Disposal of Defense High-Level Waste* (DOE 1985), provided the basis, in part, for the President's determination that DHLW should be disposed of in a geologic repository. Given that determination, DOE decided to allocate 10% of the capacity of the first repository for the disposal of DOE materials as follows: 2,268 MTHM for DSNF, 4,667 MTHM for DHLW, and 65 MTHM for naval SNF for a total of 7,000 MTHM (Section 1.4; Dreyfuss 1995; Lytle 1995).

Calculating the MTHM quantity for SNF is straightforward. It is determined by the actual heavy metal content of the SNF. However, an equivalence method for determining the MTHM in DHLW is necessary because almost all of its heavy metal has been removed. A number of alternative methods for determining MTHM equivalence for HLW have been considered over the years. Three of those methods are described in the following paragraphs.

Historical Method—Table 1-1 of DOE (1985) provides a method to estimate the MTHM equivalence for DHLW based on comparing the radioactive (curie) equivalence of commercial HLW and DHLW. The method relies on the relative curie content of a hypothetical (in the early 1980s) canister of DHLW and a hypothetical canister of vitrified waste from reprocessing of high-burnup CSNF. Based on commercial HLW containing 2.3 MTHM equivalence per canister and DHLW estimated to contain approximately 22% of the radioactivity of a canister of commercial HLW, DHLW was estimated to contain the equivalent of 0.5 MTHM per canister. Since 1985, DOE has used this 0.5 MTHM equivalence per canister of DHLW in its consideration of the potential impacts of the DHLW disposal, including consideration of the proposed Yucca Mountain repository. With the historical method, slightly less than 50% of the total inventory of DHLW could be disposed of in the repository within the 4,667 MTHM allocation for DHLW.

Total Radioactivity Method—This method would establish equivalence based on a comparison of radioactivity inventory (curies) of DHLW to that of a standard MTHM of CSNF. For this equivalence method the standard SNF characteristics are based on PWR fuel with ^{235}U enrichment of 3.11% and 39.65 GWd/MTHM burnup. Using this method, 100% of the total inventory of DHLW inventory could be disposed of in the repository within the 4,667 MTHM allocation for DHLW.

Radiotoxicity Method—The radiotoxicity method uses a comparison of the relative radiotoxicity of DHLW to that of a standard MTHM of CSNF; it is thus considered an extension of the total radioactivity method. Radiotoxicity compares the inventory of specific radionuclides to a regulatory release limit for that radionuclide and uses these relationships to develop an overall radiotoxicity index. For this equivalence, the standard SNF characteristics are based on PWR fuel with ^{235}U enrichment of 3.11% and

39.65 GWd/MTHM burnup. Using this method, 100% of the total inventory of DHLW could be disposed of in the repository within the 4,667 MTHM allocation for DHLW.

Table D-1 provides a summary comparing the results of the three MTHM equivalence calculation methods for DHLW from SRS, Hanford, and Idaho National Laboratory. As mentioned previously, the historical method results indicate that slightly less than half of the DHLW at DOE sites can be disposed of given the 4,667 MTHM allocation limit. In contrast, results from the total radioactivity method and the radiotoxicity method suggest that all DHLW at DOE sites could be disposed of under the 4,667 MTHM allocation limit with a significant margin.

Table D-1. Total MTHM Equivalence for DHLW at DOE Sites

Method	SRS	Hanford	Idaho National Laboratory	Total MTHM Equivalence
Historical Method (0.5 MTHM per canister)	2,972	6,222	285	9,479
Total Radioactivity				
10-year decay	2,140	1,524	241	3,905
1,000-year decay	138	161	34.5	334
Radiotoxicity				
1,000-year decay	117	147	16.4	280
10,000-year decay	168	7.8	18.7	195

NOTE: West Valley HLW is not included above because it is classified as commercial HLW, which is not part of the DHLW allocation limit (DOE 1008, Section 1.2.1). Instead, it is included along with CSNF in the 63,000 MTHM commercial allocation limit.

CSNF = commercial spent nuclear fuel

DHLW = DOE high-level radioactive waste

HLW = high-level radioactive waste

MTHM = metric tons heavy metal

SRS = Savannah River Site

Source: Knecht et al. 1999, Table 1.