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Preclosure Criticality Analysis Process Report

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10. Remarks

This revision of the technical report was undertaken to incorporate the licensing requirements for analyses of spent nuclear fuel and high-level radioactive wastes in the preclosure period for the geologic repository at Yucca Mountain identified in the U.S. Nuclear Regulatory Commission's 10 CFR Part 63 rule and to update the documentation describing the approach used for the preclosure criticality methodology.

Change History

11. Revision No.	12. Description of Change
01	Revision 01 This is a complete revision of and supersedes *Preclosure Criticality Analysis Process Report*, TDR-DS0-NU-000001 REV 00 to incorporate comments from DOE.
00	Initial Issue This is a complete revision of and supersedes *Preclosure Criticality Analysis Process Report*, TDR-EBS-NU-000004 REV 04.

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EXECUTIVE SUMMARY

The preclosure criticality analysis process described in this technical report provides a systematic approach for evaluating the effectiveness of the criticality control mechanisms during the preclosure period of the Monitored Geologic Repository at Yucca Mountain, Nevada. This process is appropriate for analyses of the surface and subsurface facility systems consisting of (a) waste form, canister, and waste package handling, (b) waste form aging prior to disposal, (c) waste form and package preparation for final disposal, and (d) waste form emplacement in the drifts and retrieval prior to permanent closure. This report describes the approach, performance criteria, and process applications used for preclosure criticality analyses. This process will be used to demonstrate that preclosure criticality is prevented by design for normal conditions, and controlled for off-normal conditions such that no credible (Category 1 or Category 2) event sequence will result in an end-state configuration that violates the configuration-specific upper subcritical limit (Section 3).

The preclosure criticality analysis process complies with the U.S. Nuclear Regulatory Commission's 10 CFR Part 63 rule, *Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada*[a]. This process also meets the criticality safety specific guidance found in *Yucca Mountain Review Plan, Final Report*[b] including the discussion in Appendix A, which describes the use of a risk-informed, performance-based process combined with deterministic analyses (e.g., effective neutron multiplication factor calculational methods). The U.S. Department of Energy will use this process in facility and process specific reports (i.e., criticality design and safety analyses) developed in support of licensing activities for the Monitored Geologic Repository at Yucca Mountain, Nevada, to demonstrate the acceptability of proposed systems and facilities for preventing and controlling preclosure criticality.

Figure 3-1 provides an overview of the criticality analysis process. The starting point for the preclosure criticality analysis process is to define criticality design and operational criteria based on review and analysis of waste forms, waste packages and canister designs, facility designs and characteristics, and the operational sequences in the various handling facilities. The types of event sequences important for criticality that must be considered in the identification of hazard and initiating events, event sequence analyses and subsequent categorization of event sequences are those that result in unanticipated moderation, loss of neutron absorber, geometric changes, or administrative errors in loading of the waste package. The specific events to be considered must be based on the review of each system's design, as discussed in Section 3.

Parameters that affect criticality are identified for each end-state configuration, and the ranges of values for these parameters are established for particular waste forms. These parameters may include the amounts of fissionable material, neutron absorber, moderators, and reflectors. If an event sequence important for criticality cannot be screened out as beyond Category 2 (less than one chance in 10,000 during the preclosure period), criticality evaluations are performed for those end-state configurations over the range of parameters that characterize the event sequence. If the maximum effective neutron multiplication factor for the end-state configurations is less than the configuration-specific upper subcritical limit, then criticality safety is demonstrated for the particular event sequence. Configurations that have a maximum effective neutron multiplication factor which exceeds the configuration-specific upper subcritical limit for the waste form are also acceptable provided the overall estimated probability of occurrence of the extended/refined event sequence end-state configuration is less than the Category 2 screening criterion (Section 3.6). This extended/refined event sequence probability includes the additional probability of occurrence of parameters important for criticality such that the particular configuration whose k_{eff} exceeds the configuration-specific upper subcritical limit occurs. If the probability of an extended/refined event sequence end-state configuration exceeds the Category 2 screening criterion, design or operational requirements will be imposed to reduce the probability of the event sequence end-state to below the Category 2 screening criterion.

The analysis process is continued until all event sequences have been identified and evaluated as acceptable. The analysis proceeds through all facility areas and through all facilities. The surface and subsurface facility designs are acceptable with respect to criticality when: (a) each event sequence important for criticality has been shown to have a probability less than the Category 2 screening criterion or (b) the maximum effective neutron multiplication factor of end-state configurations of all credible event sequences is less than the configuration-specific upper subcritical limit.

[a] 10 CFR 63. 2005 Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. ACC: MOL.20050405.0118 [DIRS 173273].

[b] NRC (U.S. Nuclear Regulatory Commission) 2003. *Yucca Mountain Review Plan, Final Report.* NUREG-1804 Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards. TIC: 254568 [DIRS 163274].

CONTENTS

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FIGURES

Page

TABLES

Page

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ACRONYMS AND ABBREVIATIONS

ANSI	American National Standards Institute
ANS	American Nuclear Society
BWR	boiling water reactor
CL	critical limit
CSNF	commercial spent nuclear fuel
DOE	U.S. Department of Energy
DPCs	dual purpose canisters
HLW	high-level waste
ISFSI	independent spent fuel storage installation
ISG	Interim Staff Guidance
k_{eff}	effective neutron multiplication factor for a system
MWd/MTU	megawatt days per metric ton of uranium
NNPP	Naval Nuclear Propulsion Program
NRC	U.S. Nuclear Regulatory Commission
PWR	pressurized water reactor
ROA	range of applicability
ROP	range of parameters
SAR	Safety Analysis Report
SSC	structures, systems, and components
SNF	spent nuclear fuel
TAD	Transportation, Aging and Disposal
USL	upper subcritical limit
WP	waste package

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1. INTRODUCTION

The U.S. Congress charged the U.S. Department of Energy (DOE) with managing the geologic disposal of spent nuclear fuel (SNF) and high-level radioactive waste (HLW) through the Nuclear Waste Policy Act of 1982, as amended, and the Energy Policy Act of 1992 (10 CFR Part 63 [DIRS 173273], Subpart A, Section 1). A primary objective of the geologic disposal concept is keeping the fissionable material in a condition such that there is no credible foreseen potential for a self-sustaining nuclear chain reaction (criticality) to occur. This technical report documents the process for achieving this objective for the preclosure period. The methodology for the postclosure period is documented in the *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003, [DIRS 165505]).

The means to prevent and control criticality must be addressed as part of the preclosure safety analysis required for compliance with 10 CFR Part 63 [DIRS 173273], where the preclosure period covers the time prior to and during permanent closure activities. Even though the preclosure period is expected to be 100 years, the most important part of that period for criticality concerns is the estimated 50-year period for waste emplacement in the repository (BSC 2005 [DIRS 176199], Section 3.1). One of the objectives of the preclosure safety analysis as stated in 10 CFR Part 63 [DIRS 173273] is to perform:

> "…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 required in this paragraph must include, but not necessarily be limited to, consideration of-- …
> (6) Means to prevent and control criticality…" {10 CFR Part 63 [DIRS 173273], Subpart E, Section 112(e)}.

As stated, the referenced objective of such analyses is to identify and describe the controls that are being relied upon to limit the occurrence of event sequences important for criticality or to mitigate their consequences. These analyses also identify measures taken to ensure the availability of safety systems. Criticality accidents are included among the numerous events to be identified and controlled. Thus, event sequences important for criticality need to be identified and analyzed for preventing or minimizing the probability of occurrence of criticality accidents during the preclosure period.

The preclosure criticality analysis process complies with the U.S. Nuclear Regulatory Commission's 10 CFR Part 63 rule, *Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada* [DIRS 173273]. This process also meets the criticality safety specific guidance found in *Yucca Mountain Review Plan, Final Report* [DIRS 163274] including the discussion in Appendix A, which describes the use of a risk-informed, performance-based process combined with deterministic analyses (e.g., effective neutron multiplication factor calculational methods).

The use of risk-informed, performance-based analyses in regulatory matters is consistent with the U.S. Nuclear Regulatory Commission (NRC) policy statement 60 FR 42622 [DIRS 103662]. It is likewise consistent with correspondence among the NRC commissioners on risk-informed, performance-based regulation (Jackson 1998 [DIRS 150737]). Thus, the regulations described in 10 CFR Part 63 [DIRS 173273] and the *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]) are risk-informed, performance-based to the extent practical (NRC 2003 [DIRS 163274], Abstract).

The guidance for developing and documenting a risk-informed, performance-based preclosure safety analysis is presented in *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274], Section 2.1.1) which specifically discusses criticality as part of the preclosure safety analysis. Criticality safety analysis components noted there address the following:

1) "A systematic examination of …the design; the potential hazards; initiating events, and their consequences…[and]…considers the probability of potential hazards….[and]… identifies and describes the controls that are relied upon to prevent potential event sequences from occurring or to mitigate their consequences…" (Section 2.1.1)

2) An event sequence frequency analysis that provides the means to evaluate the likelihood of such occurrences and to demonstrate whether or not they are credible. "Determination of frequency or probability of occurrence of hazards and initiating events…" (Section 2.1.1.3.1)

3) Criticality analyses ("…for conditions under which available fissionable material could pose a criticality hazard…" [Section 2.1.1.3.2]) that provide design constraints whose purpose is to prevent or control criticality and to verify that sub-criticality is maintained during the occurrence of event sequences that are important for criticality and have at least one chance in 10,000 of occurring prior to repository closure (10 CFR Part 63 [DIRS 173273], Subpart A, Section 2).

The *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility* (NRC 2002 [DIRS 159567], Introduction) also contains an acknowledgement of the risk-informed performance requirements in *Domestic Licensing of Special Nuclear Materials* (10 CFR Part 70 [DIRS 173315], Subpart A) and requires license applicants to conduct an Integrated Safety Analysis where *Integrated Safety Analysis* means:

> "…a systematic analysis to identify facility and external hazards and their potential for initiating accident sequences, the potential accident sequences, their likelihood and consequences,…the NRC requirement is limited to consideration of the effects of all relevant hazards on radiological safety, prevention of nuclear criticality accidents,…" (10 CFR Part 70 [DIRS 173315], Subpart A, Section 4).

These requirements in 10 CFR 70, Subpart A, are consistent with and applicable to the process described in this report.

The U.S. Department of Energy will use this process in facility and process specific reports (i.e., criticality design and safety analyses) developed in support of licensing activities for the

Monitored Geologic Repository at Yucca Mountain, Nevada, to demonstrate the acceptability of proposed systems and facilities for preventing and controlling preclosure criticality.

DOE O 420.1B [DIRS 176666], *Facility Safety,* establishes facility and programmatic safety requirements including criticality safety for DOE facilities. Section 3(c) *Exclusions* of this order states

> "Requirements in this Order that overlap or duplicate requirements of the Nuclear Regulatory Commission (NRC) related to radiation protection, nuclear safety, (including quality assurance), and safeguards and security of materials, do not apply to the design, construction, operations, and decommissioning of DOE facilities. This exclusion does not apply to requirements for which the NRC defers to DOE or does not exercise regulatory authority."

Since preclosure criticality safety is governed under 10 CFR 63 [DIRS 173273], DOE O 420.1B [DIRS 176666], is considered not applicable.

1.1 PURPOSE

The purpose of this technical report is to present, within the context of the regulatory requirements, a risk-informed, performance-based approach to the process of performing criticality analyses of waste packages and canisters, waste forms, and repository facilities for the time period beginning with waste form receipt at the surface facility up to permanent closure of the subsurface facility. In addition, this report provides a single reference for the preclosure criticality analysis process. The information presented in this report is not design information that can be used to support procurement, fabrication, or construction.

The preclosure criticality analysis process described in this technical report provides a systematic approach for evaluating the effectiveness of the criticality control mechanisms in the repository surface and subsurface facilities. Application of this preclosure criticality analysis process will result in facility designs such that the probability of occurrence of any foreseen preclosure event sequence that could result in a criticality accident will be below the Category 2 screening criterion.

A discussion of applicable NRC regulations and the regulatory framework, e.g., ANSI/ANS-8 standards, within which this technical report has been developed, is provided in Section 2. The process is discussed in Section 3 and conclusions are given in Section 4.

Processes, criteria, codes, and methodology for Naval spent nuclear fuel, will be provided in the NNPP Technical Support Document portion of the License Application.

1.2 SCOPE

The scope of this technical report is the complete process for performing preclosure criticality design and safety analyses for various configurations of waste forms that could occur during the preclosure period as a result of normal loading, staging, and placement operations or from event sequences representing off-normal conditions. These conditions may include unanticipated moderation, loss of neutron absorber, geometric changes, or administrative errors in waste form placement (loading) of the waste packages. The particular waste forms anticipated for receipt at the repository include but are not limited to CSNF, HLW, and DOE SNF. With a focus on safety requirements, the analyses will be performed for all processes starting with the receipt of canisters and/or transportation casks, and continuing with the transfer of bare CSNF assemblies into canisters, remediation activities, aging, loading of waste packages for closure and emplacement in the subsurface, and waste package residence in the subsurface facilities up to the time of permanent closure of the repository.

1.3 APPLICATION

Application of this process to preclosure criticality analyses will address applicable design criteria as discussed in Section 2 and provide input to the preclosure safety analysis that will demonstrate that the repository will meet its overall performance objectives for operations, including criticality, up to permanent repository closure. Operation of the repository involves a number of distinct but interrelated waste form activities and functions including receiving, handling, aging, and packaging for disposal of SNF and HLW that may be either in a canistered form or an individual (bare) assembly form.

The preclosure analysis process will be applied to design calculations and preclosure safety analyses. Using event tree/fault tree and reliability analyses in conjunction with validated effective neutron multiplication factor calculational methods, criticality design and safety analyses will demonstrate compliance with criticality design criteria to ensure that preclosure criticality is prevented for normal and for credible off-normal conditions.

Criticality analyses with respect to preclosure considerations include the evaluation of processes beginning with the receipt of transportation casks, extending through operations concerning the loading of canisters into waste packages for closure, and ending with the subsurface emplacement of waste packages.

1.4 QUALITY ASSURANCE

This technical report describes the process for performing preclosure criticality analyses for waste forms and repository facilities prior to permanent closure of the repository. This activity is subject to *Quality Management Directive* (BSC 2006 [DIRS 177655], Section 3.1.B) and the development of this report is controlled by PA-PRO-0313, *Technical Reports*.

1.5 USE OF COMPUTER SOFTWARE

No computer software subject to *Quality Management Directive* (BSC 2006 [DIRS 177655]) was used in the development of this report.

1.6 ASSUMPTIONS

There are no assumptions associated with this process report.

2. REGULATORY PERSPECTIVE

As stated in Section 1.1, the purpose of this report is to present, within the context of the regulatory requirements, a risk-informed, performance-based approach for performing criticality analyses of waste packages and canisters, waste forms, and repository facilities for the preclosure time period. This section discusses the regulatory perspectives with respect to this process.

2.1 WASTE ACCEPTANCE REQUIREMENTS FOR CRITICALITY SAFETY

The repository requirements relating to criticality safety at the time of SNF and/or HLW receipt are given in *Waste Acceptance System Requirements Document* (DOE 2002 [DIRS 158873]) as follows:

> DOE SNF Canister Criticality Requirement
> A. Preclosure. The calculated effective multiplication factor (k_{eff}) at the time of delivery to the CRWMS [*Civilian Radioactive Waste Management System*] shall be shown to not exceed 0.95 unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to criticality safety. The calculated k_{eff} must be sufficiently below unity to show at least a 5 percent margin, after allowance for bias and uncertainty in the experiments used to validate the method of calculation (DOE 2002 [DIRS 158873], Section 4.3.12).
>
> Naval SNF Canister Criticality Requirement
> A. Preclosure. The calculated effective multiplication factor (k_{eff}) at the time of delivery to the CRWMS shall be shown to not exceed 0.95 unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to criticality safety. The calculated k_{eff} must be sufficiently below unity to show at least a 5 percent margin, after allowance for bias and uncertainty in the experiments used to validate the method of calculation (DOE 2002 [DIRS 158873], Section 4.4.13).
>
> Disposable Commercial-Origin DOE SNF Canister Criticality Requirement
> A. Preclosure. The calculated effective multiplication factor (k_{eff}) at the time of delivery to the CRWMS shall be shown to not exceed 0.95 unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to criticality safety. The calculated k_{eff} must be sufficiently below unity to show at least a 5 percent margin, after allowance for bias and uncertainty in the experiments used to validate the method of calculation (DOE 2002 [DIRS 158873], Section 4.5.13).
>
> Criticality Requirement for Canisters Containing HLW
> The calculated k_{eff} for individual canisters at the time of acceptance into CRWMS shall be shown to be 0.95 or less under credible water-moderated conditions most likely to cause criticality, after allowance for bias in calculational methods and uncertainty in empirical data (DOE 2002 [DIRS 158873], Section 4.8.12).

The requirements listed in *Waste Acceptance System Requirements Document* (DOE 2002 [DIRS 158873]) pertaining to preclosure criticality are applicable to the potentially critical configurations of waste forms at the time of receipt by the repository. Subsequent criticality analyses (after receipt of waste) for repository operations are governed by the process described in this report.

The requirements listed above are expected to be revised in the upcoming revision to *Waste Acceptance System Requirements Document* (DOE 2002 [DIRS 158873]).

2.2 REPOSITORY REQUIREMENTS FOR CRITICALITY SAFETY

The regulatory requirements for criticality safety for the Yucca Mountain project are established in 10 CFR 63 ([DIRS 173273], Subpart E, Section 112(e)) as follows:

> "…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 required in this paragraph must include, but not necessarily be limited to, consideration of-- …(6) Means to prevent and control criticality;…."

Repository requirements relating to criticality safety are given in *Project Operational and Performance Requirements* (Curry 2006 [DIRS 176634], Section 2.6.2) as follows:

> "The repository shall provide means to ensure and demonstrate acceptable criticality control. WPs, site-specific casks, transfer staging area racks, and similar areas where SNF is held outside the licensed transportation casks shall be designed to maintain criticality safety."

The requirements from *Project Operational and Performance Requirements* (Curry 2006 [DIRS 176634], Section 2.6.2) are applicable to the preclosure period and are in accord with 10 CFR Part 63 ([DIRS 173273], Subpart E, Section 112(e)(6)), i.e., preclosure safety analyses must include the consideration of the *means to prevent and control criticality*.

2.3 CRITICALITY CONTROL CRITERIA AND GUIDANCE

Since the regulatory requirements of 10 CFR Part 63 ([DIRS 173273], Subpart E, Section 112(e)(6)) for control of criticality are not specific, the design criteria, NRC guidance, computational methods, operations, and industry standards for criticality safety applicable to preclosure criticality analyses are described in this section.

2.3.1 Design Criteria

The design criteria for waste form storage and handling include confirmation that analyses used to identify SSCs important to safety, safety controls, and measures to ensure the availability of the safety systems, and include adequate consideration of means to prevent and control criticality, such as complying with provisions in American National Standards Institute (ANSI)/American Nuclear Society (ANS) nuclear criticality safety standards. The standards applicable to nuclear criticality safety are listed in Section 2.3.3. The term “waste form” will be applied to either the canistered (i.e., CSNF and DOE SNF disposable canisters) or uncanistered form (e.g., CSNF received in dual purpose canisters (DPCs).

2.3.2 Regulatory Guidance

Guidance from the NRC pertaining to nuclear criticality safety analysis is contained in several publications issued by the NRC or under NRC direction. These publications include Regulatory Guides, Interim Staff Guidance (ISG) documents, NRC technical documents (NUREG series), and technical documents issued by NRC contractors (NUREG/CR series). The NRC documents reviewed in conjunction with the development of the preclosure criticality process are listed in Table 2-1 and discussed briefly in this section.

Table 2-1. NRC Guidance Document Applicability

Guidance Document	Description
Regulatory Guide 3.60 (1987), *Design of an Independent Spent Fuel Storage Installation (Dry Storage)* [DIRS 103468]	Provides guidance acceptable to the NRC staff for use in design of dry fuel staging areas and surface aging facilities.
Regulatory Guide 3.71 (2005), *Nuclear Criticality Safety Standards for Fuels and Material Facilities* [DIRS 176331]	Discusses acceptance of and exceptions to the ANSI/ANS-8 standards.
SFPO-ISG-1[a], Revision 1, *Damaged Fuel* (NRC 2002 [DIRS 164018])	Provides a definition of damaged fuel and outlines how damaged fuel is considered in storage or transportation analyses.
FCSS-ISG-3, Revision 0, *Nuclear Criticality Safety Performance Requirements and Double Contingency Principle* (NRC 2005 [DIRS 179445])	Describes the relationships between the 10 CFR 70.61 performance requirements and the double contingency principle of 10 CFR 70.64 [DIRS 173315]).
SFPO-ISG-8, Revision 2, *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks* (NRC 2002 [DIRS 161448])	Provides additional clarification to existing guidance in the NRC Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (NUREG-1617) on the use of burnup credit (NRC 2002 [DIRS 161448]).
FCSS-ISG-10[b], Revision 0, *Justification of Minimum Margin of Subcriticality* [DIRS 178606]	Provides guidance concerning the justification of minimum margin of subcriticality
SFPO-ISG-11, Revision 3, *Cladding Considerations for the Transportation and Storage of Spent Fuel* (NRC 2003 [DIRS 170332])	Provides review guidance and acceptance criteria for analyses of potential fuel reconfigurations involving cladding considerations during handling and storage operations.
SFPO-ISG-15, *Materials Evaluation* (NRC 2001 [DIRS 161724])	Provides review guidance for evaluation of material performance of components important to safety of an independent spent fuel storage installation.
Kopp 1998, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants” [DIRS 137583]	Although directed at nuclear power plants, this guidance is useful to the Yucca Mountain Project because it includes several clarifications and it documents the current NRC positions regarding the storage of SNF racks.
NUREG-1520, *Standard Review Plan for the Review of a License Application (LA) for a Fuel Cycle Facility* (NRC 2002 [DIRS 159567])	This guidance includes information regarding integrated safety analyses to support safe operation of the facility, as required by 10 CFR Part 70 [DIRS 173315].

Table 2-1. NRC Guidance Document Applicability (continued)

Guidance Document	Description
NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997 [DIRS 101903])	This guidance includes information regarding criticality design and analysis related to spent nuclear fuel handling, packaging, transfer, and storage procedures for normal, off-normal, and accident conditions pertaining to 10 CFR Part 72 [DIRS 173336].
NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 2000 [DIRS 149756])	This review plan builds on the guidance from NUREG-1536 [DIRS 101903] and includes information regarding requirements for maintaining subcritical configurations of fissile material in independent dry storage facilities under normal, off-normal, and accident conditions during all operations, transfers, and storage at the site as pertaining to applicable portions of 10 CFR 72 [DIRS 173336].
NUREG-1804, *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274])	This guidance is the review plan for the Yucca Mountain project and pertains to 10 CFR Part 63 [DIRS 173273].
NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages* (Lichtenwalter 1997 [DIRS 106574])	This guidance includes information regarding the benchmark experiment selection process and methods for calculating upper subcritical limits.
[a] SFPO – Spent Fuel Project Office (Office of Nuclear Material Safety and Safeguards, NRC)	
[b] FCSS – Fuel Cycle Safety and Safeguards	

Each guide is briefly discussed in the following paragraphs.

NRC Regulatory Guide 3.60 Revision 0 [DIRS 103468], *Design of an Independent Spent Fuel Storage Installation (Dry Storage).*

The regulatory basis for Regulatory Guide 3.60 is provided in NRC rule 10 CFR Part 72 [DIRS 173336] *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*. This regulatory guide endorses ANSI/ANS-57.9-1984 for use in the design of an independent spent fuel storage installation (ISFSI) subject to specified exceptions that do not directly affect requirements for prevention of criticality accidents. ANSI/ANS-57.9-1984 has been superseded by ANSI/ANS-57.9-1992 [DIRS 176945] that was reaffirmed in 2000.

10 CFR Part 72 [DIRS 173336] Subpart F, Section 124 provides a detailed list of criteria relating to criticality safety, which will be used for general guidance; specifically,

> "(a) *Design for criticality safety*. Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

> (b) *Methods of criticality control*. When practicable, the design of an ISFSI *[independent spent fuel storage installation]* or MRS *[monitored retrievable storage]* must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that a significant degradation of the neutron absorbing materials cannot occur over the life of the facility…." (10 CFR Part 72 [DIRS 173336], Subpart F, Section 124).

NRC Regulatory Guide 3.71 Revision 1 [DIRS 176331], *Nuclear Criticality Safety Standards for Fuels and Material Facilities.*

Regulatory Guide 3.71 provides licensees and applicants with guidance concerning criticality safety standards that the NRC has endorsed for use with nuclear fuels and material facilities. This guide describes methods that the NRC staff considers acceptable for complying with the NRC's regulation in 10 CFR Parts 70 and 76. This regulatory guide endorses 11 ANSI/ANS-8 standards without exceptions and 4 ANSI/ANS-8 standards subject to specified exceptions. These exceptions and their applicability to this process report are discussed for each standard in Section 2.3.3. The approach presented in this report is consistent with Regulatory Guide 3.71 to the same extent it is consistent with the ANSI/ANS standards discussed in Section 2.3.2.

SFPO-ISG-1 Revision 1 [DIRS 164018], *Damaged Fuel.*

SFPO-ISG-1 provides definitions of damaged fuel, outlines how damaged fuel is to be considered in storage or transportation analyses, and provides guidance for classifying spent fuel as either damaged or intact. The process presented in this report is consistent with this ISG.

FCSS-ISG-3 Revision 0 [DIRS 179445], *Nuclear Criticality Safety Performance Requirements and Double Contingency Principle.*

FCSS-ISG-3 describes the relationships between the 10 CFR 70.61 performance requirements and the double contingency principle of 10 CFR 70.64 [DIRS 173315]. The double contingency principle as described in this ISG is compatible with the risk-informed, performance-based methodology described in 10 CFR Part 63 [DIRS 173273] in terms of minimizing the likelihood of criticality accidents. The *Integrated Issue Resolution Status Report* (NRC 2005 [DIRS 175566]) states that

> "While the double-contingency principle, which has been used historically in designing criticality control systems for facilities, storage, and transportation packages, may not require the licensee to quantify the probability of the unlikely events, under 10 CFR Part 63, events must be identified, their probabilities quantified, and designations assigned as Category 1 or 2 events. ... Therefore, DOE has indicated that the repository preclosure structures, systems, and components will be designed to prevent criticality under normal operation and Categories 1 and 2 events…" (§ Section 4.1.7.2.3.7).

SFPO-ISG-8 Revision 2 [DIRS 161448], *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks*.

SFPO-ISG-8 provides guidance on the limits of licensing bases, code validation, model assumptions, use of loading curves, assignment of burnup values, and estimates of reactivity margins as they pertain to burnup credit. The process described in this report does not include validation of a burnup credit methodology. For operations with bare commercial SNF, the surface facilities are designed on the basis of the fresh fuel assumption (no burnup credit). For transportation casks or canisters that credit burnup, the repository takes the same burnup credit granted for transportation, without additional verification or confirmation, as long as the SNF is within the boundary of the transport cask or canister. However, once fuel is removed from the transport cask or canister, such fuel is treated as fresh fuel (no burnup credit). When the fuel is subsequently repackaged in a canister other than the original transport canister, the associated repository criticality safety analyses will assume fresh fuel.

FCSS-ISG-10 Revision 0 [DIRS 178606], *Justification of Minimum Margin of Subcriticality*

FCSS-ISG-10 provides guidance on the justification for the chosen minimum margin of subcriticality. The guidance given in FCSS-ISG-10 will be applied in determining the administrative margin as part of the configuration-specific upper subcritical limit (USL) calculations.

SFPO-ISG-11 Revision 3 [DIRS 170332], *Cladding Considerations for the Transportation and Storage of Spent Fuel.*

SFPO-ISG-11 does not address criticality concerns directly but discusses cladding considerations during storage that affect end-state configurations resulting from preclosure event sequences important for criticality. The guidance given in SFPO-ISG-11 will be applied for preclosure criticality analyses to determine end-state configurations, where cladding performance needs to be considered.

SFPO-ISG-15 [DIRS 161724], *Materials Evaluation*.

SFPO-ISG-15 provides guidance on materials evaluation, specifically those relied upon to demonstrate criticality control. It directs incorporation of this guidance into NUREG-1536 [DIRS 101903] and NUREG-1567 [DIRS 149756]. The approach used in the preclosure process is consistent with this ISG in relying upon neutron absorber material for criticality control.

Kopp [DIRS 137583], *Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants*.

This document lists the NRC guidance for assuring criticality safety in the storage of both fresh and irradiated fuel at light-water reactor power plants and is applicable to CSNF. The guidance in this document for criticality analysis methods includes evaluation of cross section libraries and benchmarking analyses against critical experiments to establish the bias and uncertainty, all of which are part of the preclosure criticality process.

NUREG-1520 [DIRS 159567], *Standard Review Plan for the Review of a License Application (LA) for a Fuel Cycle Facility*

The guidance provided in Chapter 3 of NUREG-1520 [DIRS 159567] addresses the NRC approach for reviewing integrated safety analysis requirements for spent nuclear fuel handling and storage facilities. Chapter 5 of NUREG-1520 discusses the review of the nuclear criticality safety program to ensure the adequacy of the controls and barriers to prevent criticality accidents. The guidance given in this document is directly applicable, since some fraction of the commercial SNF assemblies will be handled bare in the surface facilities.

NUREG-1536 [DIRS 101903], *Standard Review Plan for Dry Cask Storage Systems*

The guidance provided in Chapter 6 of NUREG-1536 [DIRS 101903], addresses the NRC approach for reviewing criticality safety analyses for dry cask storage systems. Criteria from NUREG-1536 consistent with the preclosure criticality safety approach include:

> "When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period." (NRC 1997 [DIRS 101903], Chapter 6, Section IV.3)

> "Criticality safety of the cask system should not rely on use of the following credits… fuel-related burnable neutron absorbers…[or] more than 75 percent for fixed neutron absorbers when subject to standard acceptance tests [for greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed]…" (NRC 1997 [DIRS 101903], Chapter 6, Section IV.4)

NUREG-1567 [DIRS 149756], *Standard Review Plan for Spent Fuel Dry Storage Facilities*

The guidance provided in NUREG-1567 [DIRS 149756], Glossary, Sections 4.5.3.5 and 8, addresses the NRC approach for reviewing criticality safety analyses for ISFSIs. The NRC criticality review guidance in NUREG-1567 presumes that the method for evaluating the maximum k_{eff} includes the bias and uncertainties in the k_{eff} value. Criticality design criteria consistent with this preclosure criticality process include:

> "…include:...no more than 75 percent credit for fixed neutron absorbers, unless comprehensive fabrication acceptance tests capable of verifying the presence and uniformity of the neutron absorber are implemented…determination and use of optimum (i.e., most reactive) moderator density.... The multiplication factor limit on k_{eff}, must be met for all conditions and events while at the ISFSI and MRS. This does not require determination of k_{eff} for every situation. However, it must be demonstrated that the situations that have the highest k_{eff} have been analyzed and

that thereby the normal, off-normal, and accident and conditions with the lowest margins of safety have been analyzed; or are enveloped by the analyses conducted and included in the SAR and its supporting documentation (ANSI/ANS 8.17-1984)… Criticality safety of the design must be based on favorable geometry (preferred), permanent fixed neutron absorbing materials (poisons), or both…Where solid neutron-absorbing materials are used, the design must provide a means to verify their initial efficacy, such as manufacturer's data or in-situ measurements (ANSI/ANS 8.21). Chapter 6 of NUREG-1536 provides a basis for accepting the 20-year continued efficacy of fixed neutron poisons…Unless it is shown that all spent fuel to be stored will be contained within completely intact cladding, the occurrence of pinholes and cracks in the cladding (and water fill of the voids within the cladding) must be assumed for the criticality analysis if it results in a higher k_{eff}. The water fill in the fuel-to-cladding gap should be assumed to be unborated since this is conservative from a criticality safety viewpoint…." (NRC 2000 [DIRS 149756], Section 8.4.1.1)

NUREG/CR-6361 [DIRS 106574], *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*

Guidance from NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages* (Lichtenwalter 1997 [DIRS 106574]), is used in selecting benchmark experiments to validate the criticality computational methods used in this process and in establishing configuration-specific USLs.

NUREG-1804 [DIRS 163274], *Yucca Mountain Review Plan, Final Report*

The guidance provided in NUREG-1804, *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]), addresses the NRC approach for reviewing preclosure criticality design and analyses for the Yucca Mountain repository. While there are no specific design criteria for preclosure criticality control in 10 CFR Part 63 [DIRS 173273], there is specific guidance for criticality design criteria in Section 2.1.1.7 of the *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]), namely:

- Confirm that criticality design criteria are factored into models and assumptions used for criticality analysis. These criteria should be consistent with those given in *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 2000 [DIRS 149756]) and those American National Standards Institute/American Nuclear Society–8 nuclear criticality standards adopted by the U.S. Nuclear Regulatory Commission as listed in Regulatory Guide 3.71 [DIRS 176331].

- Incorporate criticality design bases and criteria that include geometry, neutron absorbers, moderators, reflectors, and effective neutron multiplication factor limits, to ensure that nuclear fuel remains subcritical during handling, transfer, repackaging, storage, and retrieval.

- Confirm that criticality design criteria are consistent with those used in model calculations that support the design, and that isotopic enrichment of waste is properly characterized for these models. Verify that the model configurations are appropriate for the postulated repository environments, and that appropriate computer models are used in design calculations.

The preclosure criticality analysis process described in this report is consistent with this guidance.

2.3.3 Industry Standards

Several ANSI/ANS standards that are applicable to nuclear criticality safety have been reviewed with respect to development of this preclosure process. These standards have also been cited in various NUREG and Regulatory Guidance documents (specifically Regulatory Guide 3.71 *Nuclear Criticality Safety Standards for Fuels and Material Facilities*) relating to nuclear criticality safety. Note that some of the standards have more recent reaffirmation dates than those listed in Regulatory Guide 3.71 [DIRS 176331]. Each standard is briefly discussed in the following paragraphs with the identification of exceptions being taken to particular provisions in the standards together with the rationale for the exceptions.

ANSI/ANS-8.1-1998 [DIRS 123801], *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, Section 2 states:

> "This standard is applicable to operations with fissionable materials outside nuclear reactors, except for the assembly of these materials under controlled conditions, such as in critical experiments. Generalized basic criteria are presented, and limits are specified for some single fissionable units of simple shape containing ^{233}U, ^{235}U, or ^{239}Pu, but not for multiunit arrays. Requirements are stated for establishing the validity and areas of applicability of any calculational method used in assessing nuclear criticality safety. This standard does not include the details of administrative controls, the design of processes or equipment, the description of instrumentation for process control, nor detailed criteria to be met in transporting fissionable materials."

The process described in this report for preclosure criticality analyses either uses or is consistent with much of the guidance for prevention of criticality accidents provided in this standard. However, the single parameter (such as mass, enrichment, volume, and concentration) and multi-parameter limits in the standard are generally applicable to sites that modify the characteristics of fissionable materials (such as fuel fabrication facilities, waste treatment facilities, post-irradiation examination facilities, etc.). In contrast, operations relevant to the surface facility concern only the transfer, staging and repackaging of SNF. In addition, the surface facility is to be licensed according to performance-based acceptance criteria, rendering the mentioned control parameters (or at least what they were intended for) not directly applicable.

Regulatory Guide 3.71 [DIRS 176331], Section 2 takes the following exception to this standard:

> "The guidance on validating calculational methods for nuclear criticality safety, as specified in ANSI/ANS-8.1-1998, provides a procedure that is acceptable to

> the NRC staff for establishing the validity and applicability of calculational methods used in assessing nuclear criticality safety. However, it is not sufficient to merely refer to this standard in describing the validation of a method. Rather, a licensee or applicant should provide the details of validation (as stated in Section 4.3.6 of the standard) to (1) demonstrate the adequacy of the margins of subcriticality relative to the bias and criticality parameters, (2) demonstrate that the calculations embrace the range of variables to which the method will be applied, and (3) demonstrate the trends in the bias upon which the licensee or applicant will base the extension of the area of applicability. In addition, the details of validation should state computer codes used, operations, recipes for choosing code options (where applicable), cross-section sets, and any numerical parameters necessary to describe the input."

The detailed validation of the computational methods used in the application of this process report will be provided to the extent described in this exception.

ANSI/ANS-8.3-1997 (Reaffirmed 2003) [DIRS 176884], *Criticality Accident Alarm System*, Section 2 states:

> "This standard is applicable to all operations involving fissionable materials in which inadvertent criticality can occur and cause personnel to receive unacceptable exposure to radiation."

Criticality accident alarm systems per this standard are not required in repository facilities provided either an adequate demonstration is shown that the dose consequence at personnel locations is less than 0.12 gray (12 rads) (definition of excessive dose {ANSI/ANS-8.3-1997; R2003 [DIRS 176884], Section 3.3}) or the probability of occurrence of any foreseen preclosure event sequence that could result in a criticality accident will be below the Category 2 screening criterion. However, a criticality detection and alarm system will be added as a part of the Radiological Monitoring System (BSC 2004 [DIRS 177194]) and will be installed in the surface process facilities where fissile material is handled, stored, and packaged. Its purpose is to detect a nuclear criticality accident and produce an immediate evacuation signal. The system will be designed in accordance with this standard and the applicable guidance from Regulatory Guide 3.71 [DIRS 176331], Section 2, which states:

> "The guidance on criticality accident alarm systems, as specified in ANSI/ANS-8.3-1997 (reaffirmed in 2003), is generally acceptable to the NRC staff. An exception is that 10 CFR 70.24, "Criticality Accident Requirements," requires criticality alarm systems in each area in which special nuclear material is handled, used, or stored, whereas Section 4.2.1 of the standard merely requires an evaluation for such areas. Another exception is that 10 CFR 70.24 and 10 CFR 76.89, "Criticality Accident Requirements," require that each area must be covered by two detectors, whereas Section 4.4.1 of the standard permits coverage by a single reliable detector. Finally, 10 CFR 70.24 and 10 CFR 76.89 require a monitoring system capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute."

A criticality monitoring system will not be used in the subsurface facilities, thus taking an exception to Regulatory Guide 3.71.

ANSI/ANS-8.5-1996 (Reaffirmed 2002) [DIRS 158941], *Use of Borosilicate-Glass Raschig Rings as Neutron Absorber in Solutions of Fissile Material*, Section 1 states:

> **"**This standard provides guidance for the use of borosilicate-glass Raschig rings as a neutron absorber for criticality control in ring-packed vessels containing solutions of ^{235}U, ^{239}Pu, or ^{233}U."

The repository operations are designed to handle only solid SNF and HLW, thus this standard for criticality control of fissile solutions is not applicable to repository operations.

ANSI/ANS-8.6-1983 (Reaffirmed 2001) [DIRS 158942], *Safety in Conducting Subcritical Neutron Multiplication Measurements in Situ*, Section 2 states:

> **"**This standard provides safety guidance for conducting subcritical neutron-multiplication measurements where physical protection of personnel against the consequences of a criticality accident is not provided."

This standard is not applicable to repository operations.

ANSI/ANS-8.7-1998. (Reaffirmed 1999) [DIRS 158943], *American National Standard for Nuclear Criticality Safety in the Storage of Fissile Materials*, Section 2 states:

> **"**This standard is applicable to the storage of fissile materials. Mass and spacing limits are tabulated for uranium containing greater than 30 wt % ^{235}U, for ^{233}U and for plutonium, as metals and oxides."

This standard is not applicable to repository operations because canisters, casks, and waste packages for each specific waste form are designed to ensure subcriticality through limiting fissile mass, geometry, and incorporation of neutron absorbers. The surface facility will handle existing waste forms without the ability to modify their characteristics to allow compliance with the tabulated limits given in this standard. Subcriticality will be demonstrated for each waste form for all credible configurations in the surface facility using an appropriate effective neutron multiplication factor calculational method and a comparison of the maximum credible k_{eff} value to the configuration-specific USL.

ANSI/ANS-8.10-1983 (Reaffirmed 2005) [DIRS 176885], *American National Standard Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement*, Section 2 states:

> "This standard is applicable to operations with ^{235}U, ^{233}U, ^{239}Pu, and other fissile and fissionable materials outside of nuclear reactors in which shielding and confinement are provided for protection of personnel and the public, except the assembly of these materials under controlled conditions, such as in critical experiments. Criteria are provided that may be used for criticality control under these conditions."

This standard is not applicable to the preclosure criticality analysis process since subcriticality will be demonstrated for all normal configurations and end-state configurations of Category 1 and Category 2 event sequences. In addition, certain operations involving fissionable material may not be performed remotely, and so would not include shielding for a hypothetical criticality accident.

ANSI/ANS-8.12-1987 (Reaffirmed 2002) [DIRS 176895], *American National Standard for Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors*, Section 2 states:

> "This standard is applicable to operations with plutonium-uranium oxide fuel mixtures outside nuclear reactors, except for the assembly of these materials under controlled conditions, such as in critical experiments. Basic criteria are presented for plutonium-uranium fuel mixtures in single units of simple shape containing no more than 30 wt% plutonium combined with uranium containing no more than 0.71 wt% ^{235}U."

This standard is not applicable to the preclosure criticality analysis process since CSNF received at the repository is not expected to be in the simple geometric forms posited by this standard.

ANSI/ANS-8.14-2004 [DIRS 178573], *Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors***,** Section 2 states:

> "This standard provides guidance for the use of soluble neutron absorbers for criticality control. This standard addresses neutron absorber selection, system design and modifications, safety evaluations, and quality control programs."

This standard will be applicable to this preclosure criticality process if soluble neutron absorbers are used for criticality control.

A**NSI/ANS-8.15-1981 (Reaffirmed 2005)** [DIRS 176886], *Nuclear Criticality Control of Special Actinide Elements*, Section 2 states:

> "This standard is applicable to operations with the following:
>
> Np-237, Pu-238, Pu-240, Pu-241, Pu-242, Am-241, Am-242m, Am-243, Cm-243, Cm-244, Cm-245, Cm-247, Cf-249 and Cf-251.
>
> Subcritical mass limits are presented for isolated fissionable units. The limits are not applicable to interacting units."

This standard addresses control of isotopes of the actinide elements that are capable of supporting a chain reaction, other than those isotopes addressed in ANSI/ANS-8.1-1998, and that may be encountered in sufficient quantities to be of concern for criticality. It addresses these isotopes in a similar manner to that used in ANSI/ANS-8.1-1998 for ^{233}U, ^{235}U, and ^{239}Pu. The process described in this technical report for preclosure criticality analyses either uses or is consistent with the guidance for prevention of criticality accidents provided in this standard to the same extent as documented for ANSI/ANS-8.1-1998. However, the repository will not be storing separate isolated units of the special actinide absorbers detailed in the standard. Thus, this standard is not directly applicable.

ANSI/ANS-8.17-2004 [DIRS 176225], *American National Standard, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*, Section 2 states:

> "This standard provides nuclear criticality safety criteria for the handling, storage, and transportation of LWR fuel rods and units outside reactor cores."

This standard allows neutron absorbers to be relied on for controlling criticality and, in addition, it allows credit to be taken for burnup through reactivity measurements or through analysis and verification of exposure history. It also provides criteria to establish subcriticality by including an administrative margin for calculated multiplication factors to ensure subcriticality is maintained.

Regulatory Guide 3.71 DIRS [176331], Section 2, takes an exception to this standard, namely:

> "The general safety criteria and criteria to establish subcriticality, as specified in ANSI/ANS-8.17-2004, provide guidance that is acceptable to the NRC staff for preventing nuclear criticality accidents in handling, storing, and transporting fuel assemblies at fuel and material facilities. The only exception is that licensees and applicants may take credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored."

The process described in this report does not include validation of a burnup credit methodology. For operations with bare commercial SNF, the surface facilities are designed on the basis of the fresh fuel assumption (no burnup credit). For transportation casks that credit burnup, the repository takes the same burnup credit granted for transportation, without additional verification or confirmation, as long as the SNF is within the boundary of the transport cask or canister. However, once fuel is removed from the transport cask or canister, such fuel is treated as fresh fuel (no burnup credit). When the fuel is subsequently repackaged in a canister other than the original transport canister, the associated repository criticality safety analyses will assume fresh fuel. Thus, the exception to this standard is not applicable to the process described in this report.

The process used for criticality analyses and the approach for establishing neutron absorber credit are consistent with the guidance in this standard. In addition, the guidance in Criteria to Establish Subcriticality (ANSI/ANS-8.17-2004 [DIRS 176225], Section 5) is used in establishing the USL applicable to configurations evaluated with the preclosure process. The approach for establishing subcriticality prescribed in Section 5.1 of this standard is similar to the approach recommended in NUREG/CR-6361 *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages* (Lichtenwalter et al. 1997 [DIRS 106574]) for establishing subcriticality.

ANSI/ANS-8.19-2005 [DIRS 176226], *American National Standard, Administrative Practices for Nuclear Criticality Safety*, Section 2 states:

> **"**This standard provides criteria for the administration of a nuclear criticality safety program for outside-of-reactor operations in which there exists a potential for nuclear criticality accidents. Responsibilities of management, supervision, and the nuclear criticality safety staff are addressed. Objectives and characteristics of operating and emergency procedures are included."

This standard is not used for preclosure criticality analyses, but will be implemented by the repository criticality safety program.

ANSI/ANS-8.20-1991 (Reaffirmed 2005) [DIRS 178572], *American National Standard, Nuclear Criticality Safety Training*, Section 2 states:

> **"**This standard provides criteria for nuclear criticality safety training for personnel associated with operations outside reactors where a potential exists for criticality accidents. It is not sufficient for the training of nuclear criticality safety staff."

This standard is not used for preclosure criticality analyses, but will be implemented by the repository criticality safety program.

ANSI/ANS-8.21-1995 (Reaffirmed 2001) [DIRS 176893], *American National Standard for the Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*, Section 2 states:

> **"**This standard provides guidance for the use of fixed neutron absorbers as an integral part of nuclear facilities and fissionable material processing equipment outside reactors, where such absorbers provide criticality safety control."

The process described in this report makes use of fixed absorbers as described in this standard. This standard is applicable to the in-service verification and inspection of fixed neutron absorbers in the spent fuel staging racks in the Wet Handling Facility pool. However, the in-service verification and inspection requirements for absorber effectiveness cannot be implemented in sealed canisters. The guidance in this standard is applicable for the installation and verification of fixed absorber material prior to loading and sealing of these canisters.

ANSI/ANS-8.22-1997 [DIRS 158946], *American National Standard for Nuclear Criticality Safety Based on Limiting and Controlling Moderators*, Section 2 states:

> **"**This standard applies to limiting and controlling moderators to achieve criticality safety in operations with fissile materials in a moderator control area. This standard does not apply to concentration control of fissile material."

The guidance given in this standard is applicable to the preclosure criticality analysis process to demonstrate criticality safety in areas where moderator control is credited.

ANSI/ANS-8.23-1997 (Reaffirmed in 1998) [DIRS 158947], *American National Standard for Nuclear Criticality Accident Emergency Planning and Response*, Section 2 states:

> **"**This standard provides guidance for minimizing risks to personnel during emergency response to a nuclear criticality accident outside reactors. This standard applies to those facilities for which a criticality accident alarm system…is in use."

This standard is not used in the criticality analysis process, but will be implemented by the repository criticality safety program.

ANSI/ANS-57.7-1988 (Reaffirmed 1997) [DIRS 177851], *American National Standard, Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type),* Section 1.2 states:

> "This standard provides design criteria for systems and equipment of a facility for the receipt and storage of spent fuel from light water reactors. It contains requirements for the design of major buildings and structures including the shipping cask unloading and spent nuclear fuel storage pools, cask decontamination, unloading and loading areas, and the surrounding buildings which contain radwaste treatment, heating, ventilation, and air conditioning, and other auxiliary systems. It contains requirements and recommendations for spent fuel storage racks, special equipment and area layout configurations, the pool structure and its integrity, pool water cleanup, ventilation, residual heat removal, radiation monitoring, fuel handling equipment, cask handling equipment, prevention of criticality, radwaste control and monitoring systems, quality assurance requirements, materials accountability, and physical security."

The guidance given in this standard is applicable to the preclosure criticality analysis process to demonstrate criticality safety in the pool in the Wet Handling Facility.

ANSI/ANS-57.9-1992 (Reaffirmed 2000) [DIRS 176945], *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*, Section 1.1 states:

> "This standard is intended to be used by the owner and operator of a dry storage-type independent spent fuel storage installation (ISFSI) in specifying the design requirements and by the designer to meet the minimum requirements of such installations. The standard includes requirements for the following: the design of major buildings and structures, shipping cask unloading and handling facilities, cask decontamination, loading and unloading areas, spent fuel storage areas and racks, fuel handling equipment, radiation shielding, special equipment and area layout configurations, air of gas quality, storage area integrity, air or gas cleanup, fuel inspection, ventilation, residual heat removal, radiation monitoring, prevention of criticality, radwaste control and monitoring systems, provisions to facilitate decommissioning, quality assurance, materials accountability, and physical security."

This standard provides design criteria for an independent spent fuel storage installation for light water reactor spent fuel that incorporates one or more of the dry storage concepts that include three major types, i.e., cask (silo), drywall (caisson), and vault (canyon). The standard provides performance and design requirements along with general guidelines that will assist in the design and licensing requirements. This standard invokes ANSI/ANS-8.17-1984 (Reaffirmed 1989) for general nuclear criticality safety requirements and ANSI/ANS-8.1-1998 for general criticality safety practices. The guidance provided in this standard is applicable to the preclosure criticality analysis process to the extent that many of the operations in the surface and subsurface facilities during the preclosure period are quite similar to those in an independent spent fuel storage installation

3. PROCESS APPROACH

An overview of the principal elements of the preclosure criticality analysis process is given in Figure 3-1. The following subsections describe in detail each of the process blocks, feed elements and decision points depicted in Figure 3-1. The discussion includes the specific process taking place as well as the required level of input needed by the process and the expected outputs to be generated by the specific process block.

The analysis process begins with design, waste form, and operational detail sufficient to perform design analyses and to identify event sequences important for criticality. If the end-state of the event sequence for the particular waste form and containment has a probability of occurrence below the Category 2 screening criterion, then further analysis of this end-state is not required, i.e., this event sequence is screened out, based upon low probability. However, if the end-state has a probability of occurrence at or above the Category 2 screening criterion, criticality analyses of end-state configurations are performed to assess the reactivity (maximum k_{eff}) of these configurations. If the reactivity of the analyzed configurations has a maximum k_{eff} over the range of configurations that does not exceed the USL, then the end-state configurations are considered to be subcritical and can be screened from further analyses. For end-state configurations where the maximum k_{eff} value over the range of configurations exceeds the USL (and the probability of occurrence of the end-state exceeds the Category 2 criterion), the end-state will be acceptable provided that the probability of occurrence of the extended/refined event sequence end-state does not exceed the Category 2 screening criterion (Section 3.6). The probability of the extended/refined event sequence includes the additional probability of occurrence of parameters important for criticality such that the particular configuration whose k_{eff} exceeds the configuration-specific upper subcritical limit occurs. If the probability of an extended/refined event sequence end-state configuration violates the Category 2 screening criterion, design or operational requirements will be imposed to reduce the probability of the event sequence end-state to below the Category 2 screening criterion.

The analysis process is continued until all event sequences have been identified and evaluated as acceptable. The analysis proceeds through all facility areas and through all facilities. The surface and subsurface facility designs are acceptable with respect to criticality when all facilities with event sequences important for criticality have no event sequence that fails to satisfy the Category 2 screening criterion or the maximum effective neutron multiplication factor of end-state configurations of all credible event sequences is less than the configuration-specific upper subcritical limit.

The preclosure criticality analysis process is described in Sections 3.1 through 3.6. Section 3.7 discusses the application of the preclosure criticality analysis process.

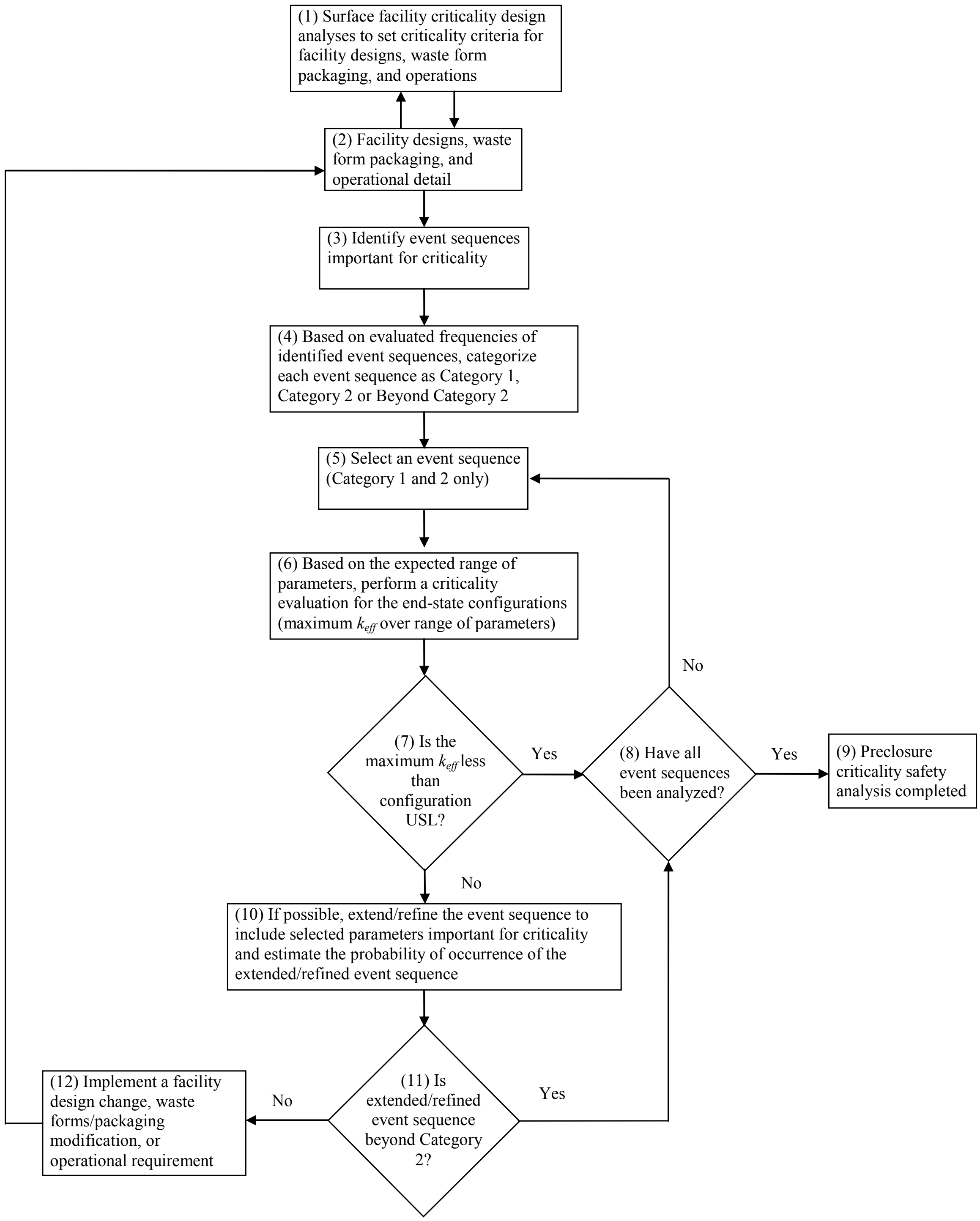


Figure 3-1. Overview of the Preclosure Criticality Analysis Process

3.1 FACILITY CRITICALITY DESIGN ANALYSES (BOX 1)

The purpose of these analyses is to define criticality design and operational criteria such that criticality is prevented for normal and credible off-normal conditions during the preclosure period. These analyses are performed during the initial stages of the design and may require several iterations until the design and operational criteria are established. The established criticality design and operational criteria are then reflected in the Preclosure Nuclear Safety Design Basis and the facility-specific Basis of Design documents.

3.2 FACILITY DESIGN, WASTE FORM, OPERATIONAL DETAIL (BOX 2)

Operation of the repository involves a number of distinct but interrelated waste form activities. These activities include receiving, handling, aging, and packaging SNF and HLW for disposal. These waste forms may be received either in a canistered or an individual (bare) assembly form. The operations performed in the surface and subsurface facilities during these activities are:

1. Operations in the carrier/cask handling system
2. Operations in the assembly wet transfer system
3. Operations in the canister transfer system and aging facility
4. Operations in the waste package handling system
5. Emplacement of waste packages.

For each waste form type (i.e., CSNF, DOE SNF, and HLW), design values or technical specification limits for fissionable isotope concentrations will be used in establishing the initial isotopic content of the waste form. When fissile isotope production during reactor operations leads to a higher reactivity (e.g., breeder fuel), adjustments, based on waste custodian fuel characteristics reports, will be made to the design values to account for the increase in fissionable isotopic content, which is conservative with respect to criticality.

Criticality analyses for DOE SNF will use the most reactive fuel state (i.e., fresh fuel assumption with no burnup credit for non-breeder reactor fuel, or calculated most reactive state for breeder reactor fuel). The high degree of variability in the DOE SNF inventory with respect to the SNF parameters (Radulescu, et al. 2004 [DIRS 165482], Section 3.1.2), and the level of uncertainty in the parameter values in general, and for burnup values in particular, preclude any practical use of burnup credit for these waste forms.

The process described in this report does not include validation of a burnup credit methodology for commercial SNF. For operations with bare commercial SNF, the surface facilities are designed on the basis of the assumption of fresh fuel (no burnup credit). For transportation casks that credit burnup, the repository takes the same burnup credit granted for transportation, without additional verification or confirmation, as long as the SNF is within the boundary of the transport cask or canister. However, once fuel is removed from the transport cask or canister, such fuel is treated as fresh fuel (no burnup credit). When the fuel is subsequently repackaged in a canister other than the original transport canister, the associated repository criticality safety analyses will assume fresh fuel.

For CSNF, a sensitivity analysis (BSC 2005 [DIRS 175046], Section 6.6) has shown that there is no unique PWR or BWR assembly design that is most reactive for all configurations using the fresh fuel assumption. Thus, the choice for most reactive fuel state must be sufficiently supported by sensitivity studies that include, among other things, fuel density, enrichment, and isotopic composition. The most reactive state with respect to isotopic composition normally includes neglecting neutron absorber isotopes, e.g., ^{234}U and ^{236}U, and inclusion of fissile isotopes to maximize reactivity.

3.3 IDENTIFICATION OF CRITICALITY EVENT SEQUENCES (BOX 3)

Event sequences important for criticality are a series of actions and/or occurrences within the repository operational facilities that could potentially lead to a criticality accident. An event sequence may include one or more initiating events and any number of combinations of system component failures, including those produced by operating personnel action or inaction. The event tree process provides a systematic approach to address the scenarios identified as having event sequences with potential to increase the reactivity of their end-state configurations. This process can be used to identify and evaluate end-state configurations for the various operations with waste forms expected for receipt at the monitored geologic repository. The event tree process is used in the evaluations of event sequences to categorize those event sequences based on their frequency. The top events on the event tree are the specific events important for criticality. Branching under the top events provides a traceable path through each event sequence culminating with their respective end-state configurations.

The event sequences to be considered as part of the criticality safety analysis must be determined through review of the facility design and operations and identified as part of the preclosure safety analysis. The performance of the SSCs and implementation of operational requirements are reviewed to verify that all sequences important for criticality have been identified. These reviews will identify and describe the controls and procedures that are relied upon to limit the likelihood of or prevent event sequences important for criticality from leading to a criticality accident. The analyses will consider features designed to prevent and control criticality, and to identify measures in place to ensure the availability of safety systems.

Identification of event sequences important for criticality will be included in the identification of event sequences important to safety. However, the list of event sequences provided in categorization analyses, as described in Section 3.4, may not necessarily be strictly specific to criticality safety. For the criticality safety portion of the preclosure safety analysis, otherwise benign events must also be considered. These include events that may result in unanticipated moderation (e.g., activation of sprinkler systems, introduction of hydrogenous fluids from failed hydraulic cylinders or oil systems, or inadvertent proximity to reflector materials), formation of unanticipated geometries (e.g., drop events), or operational errors in waste form placement (e.g., canister misloads). A guide for identification of event sequences important for criticality is shown in Figure 3.2. The following is a description of the decision points in the figure that aid in the identification of event sequences important for criticality:

End State for Evaluation:

This is the end point of any event sequence (e.g., crane failure and a drop of canister) whose importance for criticality needs to be identified.

Waste Form Handling:

This decision point allows for the identification of the waste form packaging affected by the event sequence (e.g., did the event result in a canister drop or a bare assembly drop).

Waste Form Type:

This decision point allows for the identification of the waste form type affected by the event sequence (e.g., DOE SNF, CSNF, HLW or NNPP).

Moderation inside Canister:

This decision point allows for determining whether the event sequence resulted in moderation inside the canister. For most canisters, moderator cannot enter the canister without canister breach. For a few DOE SNF types, such as TRIGA SNF, the fuel matrix is self-moderated.

Impact with SNF and/or Basket Reconfiguration:

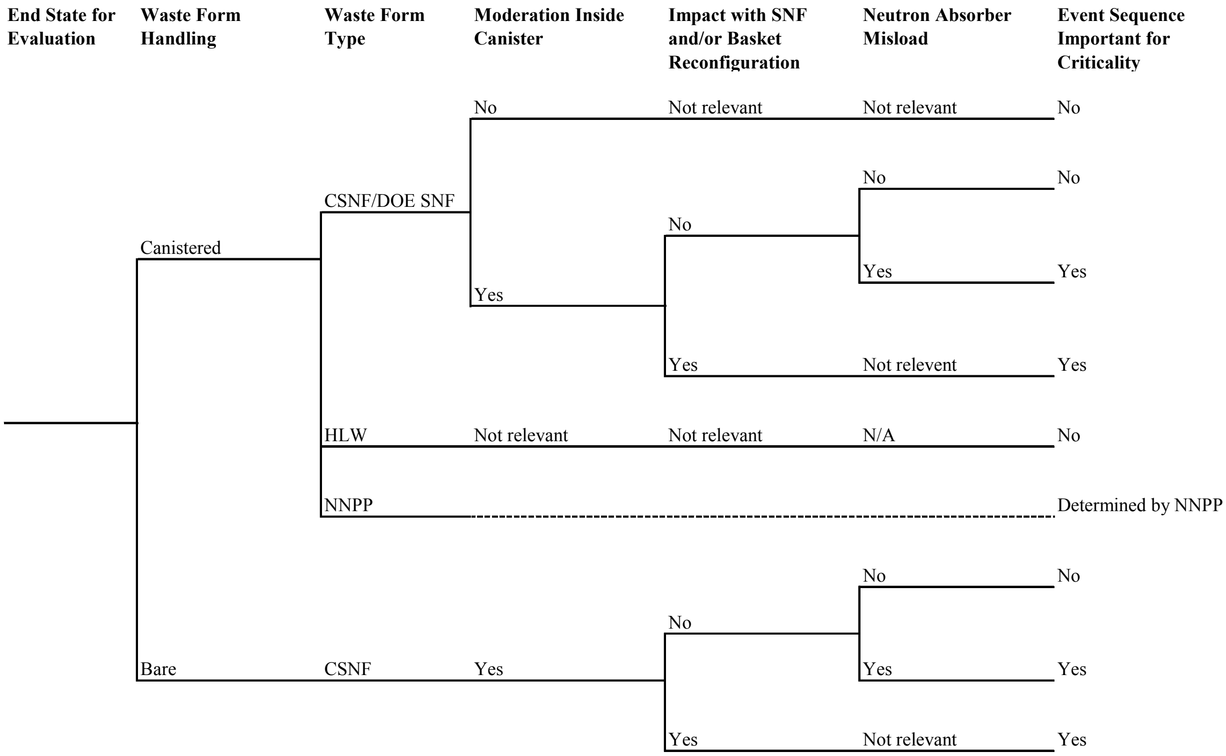
This decision point allows for determining whether the event sequence resulted in reconfiguring the basket structure and/or the waste form.

Neutron Absorber Misload:

This decision point allows for determining whether the event sequence resulted in misloading the neutron absorber required for a particular waste form. The misload evaluation considers either the absence of the neutron absorber or the presence of a reduced amount of neutron absorber.

Event Sequence Important for Criticality:

Based on the results of the decision points described above, the event sequences are categorized based on whether they are or are not important for criticality. Effective neutron multiplication factor calculations are performed for only those event sequences that are deemed important for criticality. For event sequences to be important for criticality, moderation would have to be present along with reconfiguration of the waste form, reconfiguration of the basket, or misloading of the neutron absorber.



NOTES:
"Not relevant" means that the corresponding event can occur, but it does not determine the importance to criticality of the given branch.
"N/A" (not applicable) means that the corresponding event cannot occur on that branch.

Figure 3-2. Guide for Identification of Event Sequences Important for Criticality

3.4 CATEGORIZATION OF EVENT SEQUENCES (BOX 4)

Categorization of event sequences is based on evaluated frequencies and documented in event sequence and quantification reports for each facility. These reports utilize a fully integrated probabilistic risk assessment computational tool which develops each event sequence by linking the fault trees and the initiating event. The probability distributions are propagated through the event tree/fault tree logic model to obtain the probability distribution of each event sequence. The probability distributions represent the uncertainties associated with the analysis input parameters. Categorization of event sequences is achieved by matching each mean value of the event sequence probability distribution to the Category 1 (an expected number of occurrences of at least one in the preclosure period), Category 2 (a probability greater than 1 chance in 10,000 but an expected number of occurrences less than one in the preclosure period), or beyond Category 2 (less than 1 chance in 10,000 in the preclosure period) event sequence thresholds.

3.5 K-EFFECTIVE EVALUATIONS (BOXES 6 AND 7)

If any of the end-states resulting from event sequences important for criticality have a probability of occurrence above the Category 2 screening criterion, then k_{eff} evaluations are performed for each end-state configuration over its range of parameters. The process for performing such calculations is described in the *Criticality Model* (BSC 2004 [DIRS 168553]). The evaluation process of configurations that have potential for criticality also follows the guidance given in *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* (ANSI/ANS-8.1-1998 [DIRS 123801]).

The criticality evaluation process begins with the selection of parameters and parameter values that are obtained from event sequences important for criticality as well as the waste form(s) characteristics for the configurations. The ranges of these parameters and values represent the material composition and geometry that define configurations. A configuration is considered acceptably subcritical if the calculated k_{eff} plus calculational uncertainties is less than or equal to the configuration-specific upper subcritical limit. In equation notation, the use of the USL is:

$$k_S + \Delta k_S \leq \mathrm{USL} \qquad \text{(Eq. 1)}$$

where

k_S = calculated k_{eff} for the system

Δk_S = an allowance for:
(a) statistical or convergence uncertainties, or both in the computation of k_s [Note: bounds for k_{eff} values are typically provided at the 95% confidence level],
(b) material and fabrication tolerances, and
(c) uncertainties due to the geometric or material representations used in the computational method. [Note: allowance for items (b) and (c) can be obviated by using bounding representations].

USL = an upper limit on k_{eff} characterized by statistical tolerance limits that accounts for biases and uncertainties associated with the criticality code trending process, any uncertainties due to extrapolation outside the range of experimental data, or limitations in the geometrical or material representations used in the computational method, and an administrative margin to ensure subcriticality.

ANSI/ANS-8.17 [DIRS 176225], Section 5, *Criteria to Establish Subcriticality* states:

> "Where methods of analysis are used to predict neutron multiplication factors, the calculated multiplication factor…shall be equal to or less than an established allowable neutron multiplication factor [USL]…"

The USL is an upper limit placed on k_{eff} to ensure subcriticality with allowances made for the bias and uncertainty in the calculation model as well as an administrative criticality safety margin. The administrative criticality safety margin is the difference between a critical limit (CL) and an upper subcritical limit, i.e.,

$$CL = 1 - \text{sum of bias and uncertainties}$$

$$USL = CL - \text{administrative margin.}$$

A USL is associated with a specific type of waste form configuration and is characterized by a representative set of benchmark criticality experiments and a technically justified administrative margin following the guidance given in FCSS-ISG-10 [DIRS 178606], which specifies that the chosen value should be technically justified based on the adequacy of the validation and applicability of the benchmark experiments selected for the particular waste form and configuration. The set of criticality experiments also prescribes the basic range of applicability of the results.

The USL is represented in equation form based on ANSI/ANS-8.17-2004 [DIRS 176225], Section 5, as:

$$USL(x) = f(x) - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m \qquad \text{(Eq. 2)}$$

where

x	=	a neutronic parameter used for trending
f(x)	=	the lower-bound tolerance limit function accounting for biases and uncertainties that cause the calculational results to deviate from the true value of k_{eff} for a critical experiment, as reflected over an appropriate set of critical experiments
Δk_{EROA}	=	penalty for extending the range of applicability
Δk_{ISO}	=	penalty for isotopic composition bias and uncertainty if burnup credit is used
Δk_m	=	an administrative margin ensuring subcriticality for preclosure analyses, turning the critical limit function into an USL function

Based on a given set of critical experiments, the USL is estimated as a function of a trending parameter (x) for the experiments. Because USL(x) can vary with this parameter, the USL is expressed as a function of this parameter within an appropriate range of applicability derived from the parameter bounds.

The repository will contain various types of waste forms, the majority of which will be CSNF. Each waste form will be analyzed separately for configurations that have potential for criticality and are characterized using a set of benchmark critical experiments that span the characteristics of the particular waste form. This includes analyzing potentially critical configurations during the receiving, packaging, and emplacing operations. Analyses for each of these operations may require different sets of benchmark critical experiments since the neutronic parameters may change between operations.

The relationship of the equations in this section to those in Section 5 of ANSI/ANS-8.17-2004 [DIRS 176225] is shown in Appendix B.

3.6 EXTENSION/REFINEMENT OF EVENT SEQUENCES IMPORTANT FOR CRITICALITY (BOXES 10, 11, AND 12)

For end-state configurations where the maximum k_{eff} value over the range of configurations exceeds the USL (and the probability of occurrence of the end-state exceeds the Category 2 criterion), the end-state will be acceptable provided that the probability of occurrence of the extended/refined event sequence end-state does not exceed the Category 2 screening criterion. This extended/refined event sequence probability includes the additional probability of occurrence of parameters important for criticality such that the particular configuration whose k_{eff} exceeds the configuration-specific upper subcritical limit occurs. These parameters include but are not limited to the specific details associated with level of moderation, extent of fuel rearrangement, fuel basket geometric reconfiguration, etc. If the probability of an extended/refined event sequence end-state configuration exceeds the Category 2 screening criterion, design or operational requirements will be imposed to reduce the probability of the event sequence end-state to a value below the Category 2 screening criterion.

3.7 PROCESS APPLICATION

As stated in Section 1, the preclosure criticality analysis process described in this report and summarized in Figure 3-1 provides a systematic process for evaluating the effectiveness of the criticality control mechanisms in the repository facilities in limiting the likelihood of occurrence of event sequences important for criticality to less than 1 chance in 10,000 over the preclosure period using a combination of risk-informed, performance-based and deterministic analyses.

For a criticality to occur, multiple changes in conditions must occur (e.g., canister breach, water intrusion with retention, and/or removal of neutron absorbers). The process permits the identification and evaluation of the probability of the events and factors contributing to the likelihood for occurrence of such events. With this type of information, reasonable and feasible approaches to reducing the probability of occurrence of possible criticality events can be evaluated and implemented as necessary.

4. CONCLUSIONS

This technical report presents, within the context of the regulatory requirements, a risk-informed, performance-based approach to the process of performing criticality analyses of waste packages and canisters, waste forms, and repository facilities for the time period beginning with waste form receipt at the surface facility up to permanent closure of the subsurface facility. Application of this preclosure criticality analysis process will result in facility designs such that the probability of occurrence of any foreseen preclosure event sequence that could result in a criticality accident will be below the Category 2 screening criterion.

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APPENDIX A

GLOSSARY

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APPENDIX A

GLOSSARY

This glossary contains the meaning of the specialized terms used in the report.

Bare CSNF describes commercial SNF assemblies that are handled individually.

Canistered SNF describes SNF that is handled in a sealed canister.

Configuration-Specific USL is the upper subcritical limit with which the k_{eff} of the configuration being analyzed for criticality potential will be compared.

Credible event sequence is a Category 1 or Category 2 event sequence.

Extended/Refined Event Sequence includes the additional probability of occurrence of parameters important for criticality such that the particular configuration whose k_{eff} exceeds the configuration-specific upper subcritical limit occurs.

Safety Systems are structures, systems and components that are identified to be important to safety.

USL is an upper limit placed on k_{eff} to ensure subcriticality with allowances made for the bias and uncertainty in the calculation model as well as an administrative criticality safety margin.

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APPENDIX B

RELATIONSHIP OF SUBCRITICALITY CRITERIA

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APPENDIX B

RELATIONSHIP OF SUBCRITICALITY CRITERIA

The relationship of the equations in Section 3.5 of this report to those in Section 5 of ANSI/ANS-8.17-2004 [DIRS 176225] is shown below.

Equation in Section 3.5 of ANSI/ANS-8.17 is:

$$k_p \leq k_c - \Delta k_p - \Delta k_c - \Delta k_m \qquad \text{(Equation A-1)}$$

Moving Δk_p to the left side gives:

$$k_p + \Delta k_p \leq k_c - \Delta k_c - \Delta k_m \qquad \text{(Equation A-2)}$$

Equations 1 and 2 in Section 3.5 of this report are:

$$k_S + \Delta k_S \leq \text{USL} \qquad \text{(Equation A-3)}$$

and

$$\text{USL(x)} = \text{f(x)} - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m \qquad \text{(Equation A-4)}$$

Thus,

$$k_S + \Delta k_S \leq \text{f(x)} - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m \qquad \text{(Equation A-5)}$$

Comparing equations A-2 and A-5 results in:

$$k_p + \Delta k_p = k_S + \Delta k_S \qquad \text{(Equation A-6)}$$

and

$$\text{f(x)} - \Delta k_{EROA} - \Delta k_{ISO} - \Delta k_m = k_c - \Delta k_c - \Delta k_m \qquad \text{(Equation A-7)}$$

Δk_{ISO} only applies when there is compositional uncertainty associated with burnup credit, thus:

$$\text{f(x)} - \Delta k_{EROA} - \Delta k_m = k_c - \Delta k_c - \Delta k_m \qquad \text{(Equation A-8)}$$

The description of f(x) – Δk_{EROA} in Section 3.5 of this process report and the description of k_c - Δk_c in Section 5 of ANSI/ANS-8.17 are the same. They are the result of benchmarking the criticality codes and specific configuration to be analyzed with applicable critical benchmarks.