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Preclosure Criticality Analysis Process Report

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No document review in accordance with *Document Review*, PA-PRO-0601, is required because this technical report does not affect any discipline or organization other than the originating organization.

Change History

11. Revision No.	12. Description of Change
03	Revision 03 This is a minor revision of and supersedes *Preclosure Criticality Analysis Process Report,* TDR-DS0-NU-000001 REV 02. The changes are marked by vertical lines on the right hand side of each page that was revised.
02	Revision 02 This is a complete revision of and supersedes *Preclosure Criticality Analysis Process Report*, TDR-DS0-NU-000001 REV 01 to incorporate external review comments and to update the process based on design and operational changes.
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 determining the need for criticality controls and for evaluating their effectiveness 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 including (1) waste form, canister, and waste package handling, (2) waste form aging prior to disposal, (3) waste package preparation for final disposal, and (4) waste package emplacement in the drifts 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 for normal conditions and for off-normal conditions such that no event sequence with a mean probability of occurrence greater than or equal to one chance in 10,000 during the preclosure period 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*[1]. This process also addresses the criticality safety specific review methods and satisfies the acceptance criteria found in *Yucca Mountain Review Plan, Final Report*[2]. The U.S. Department of Energy will use this process in facility and process specific reports (i.e., nuclear criticality calculations 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, canister designs, facility designs and characteristics, and the operational sequences in the various handling facilities. In order to determine the criticality potential for each specific waste form and associated facility and handling operations, effective neutron multiplication factor (k_{eff}) sensitivity calculations are performed. These calculations evaluate the impact on system reactivity of variations in each of the parameters important to criticality during the preclosure period, which are waste form characteristics, reflection, interaction, neutron absorbers (fixed and soluble), geometry, and moderation. The criticality calculations in this process step determine the sensitivity of k_{eff} to variations in any parameter(s) as a function of the other parameters and provide guidance to event sequence development, quantification, and categorization analyses on whether each parameter (or its effect): (1) is bounded and does not need to be controlled; (2) needs to be controlled if another parameter is not controlled (conditional control); or (3) needs to be controlled because it is the primary criticality control parameter.

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[1] 10 CFR 63. 2005 *Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada*. ACC: MOL.20050405.0118 (Ref. 5.2.1).

[2] 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 (Ref. 5.1.7).

Based on internal and external hazards identification and screening analyses and event sequence development and quantification analyses, the event sequences that impact these criticality control parameters are identified, developed, quantified and categorized. If an event sequence important to 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 k_{eff} for the end-state configurations is less than the configuration-specific upper subcritical limit, then criticality safety is demonstrated for the particular event sequence. For end-state configurations where the maximum k_{eff} value exceeds the upper subcritical limit, and the probability of occurrence of the end-state configuration exceeds the Category 2 criterion, the event sequence is further extended or refined to credit additional design features or procedural safety controls such that the event sequence probability is reduced. The probability of the extended or refined event sequence may include the additional probability of occurrence of parameters important to criticality, such as degree of moderation, extent of fuel rearrangement, and fuel basket geometric reconfiguration. The end-state configuration is acceptable provided that the probability of occurrence of the extended or refined event sequence does not exceed the Category 2 screening criterion. If the probability of an extended or refined event sequence exceeds the Category 2 screening criterion, design or operational requirements will be imposed to reduce the probability of the event sequence to below the Category 2 screening criterion.

The analysis process is continued until all facilities and waste forms have been evaluated, criticality control parameters have been established, and event sequences important to criticality have been identified and evaluated as acceptable. The surface and subsurface facility designs are acceptable with respect to criticality when: (1) each event sequence important to criticality has been shown to have a probability less than the Category 2 screening criterion or (2) the maximum effective neutron multiplication factor of end-state configurations of all Category 1 and Category 2 event sequences important to criticality is less than the configuration-specific upper subcritical limit.

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ACRONYMS AND ABBREVIATIONS

ANSI	American National Standards Institute
ANS	American Nuclear Society
CSNF	commercial spent nuclear fuel
DOE	U.S. Department of Energy
FCSS	Fuel Cycle Safety and Safeguards
GROA	geologic repository operations area
HAZOP	Hazard and Operability Study
HLW	high-level radioactive waste
ISFSI	independent spent fuel storage installation
ISG	Interim Staff Guidance
ITS	important to safety
k_{eff}	effective neutron multiplication factor
LBTL	Lower Bound Tolerance Limit
LWR	light water reactor
MRS	monitored retrievable storage
NNPP	Naval Nuclear Propulsion Program
NRC	U.S. Nuclear Regulatory Commission
PCSA	preclosure safety analysis
SAR	safety analysis report
SFPO	Spent Fuel Project Office
SSCs	structures, systems, and components
SNF	spent nuclear fuel
TAD	transportation, aging and disposal
USL	upper subcritical limit
WHF	Wet Handling Facility

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1. INTRODUCTION

1.1 BACKGROUND

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*, Subpart A, Section 1 (Ref. 5.2.1)). A primary objective of the geologic disposal concept is keeping the fissionable material in a condition such that there is no credible potential for a self-sustaining nuclear chain reaction (criticality) to occur. This technical report documents the process for achieving this objective during the period prior to permanent closure of the Monitored Geologic Repository. The methodology for the postclosure period is documented in the *Disposal Criticality Analysis Methodology Topical Report* (Ref. 5.1.9).

1.2 SAFETY REQUIREMENTS

The means to prevent and control criticality must be addressed as part of the preclosure safety analysis (PCSA) required for compliance with *10 CFR Part 63* (Ref. 5.2.1), 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 (*Evaluation of Alternative Design Concepts for the Critical Decision-1 Revision*, Section 3.1 (Ref. 5.1.1)). The only preclosure criticality technical requirement in 10 CFR Part 63 *Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada* is to perform:

> "…An analysis of the performance of the structures, systems, and components (SSCs) 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*, Subpart E, Section 112(e) (Ref. 5.2.1)).

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

Repository requirements relating to criticality safety are given in *Project Design Criteria Document*, Section 4.10.2.1.1 (Ref. 5.1.2) as follows:

> "SSCs shall be designed such that adequate controls and procedures can be effectively implemented to**:** prevent criticality and institute controls that are relied on to limit or prevent potential event sequences or mitigate their consequences during processing, handling, transfer, or transport of the waste form or waste package in the preclosure period…"

These requirements are applicable to the preclosure period and are in accord with *10 CFR Part 63*, Subpart E, Section 112(e)(6) (Ref. 5.2.1)).

1.3 PROCESS REQUIREMENTS

The preclosure criticality analysis process described in this report complies with *10 CFR Part 63* (Ref. 5.2.1). This process also meets the criticality safety specific review methods and acceptance criteria found in NUREG-1804, *Yucca Mountain Review Plan, Final Report* (Ref. 5.1.7) including the discussion in Appendix A, which describes the use of a risk-informed, performance-based process combined with deterministic analyses.

The review methods and acceptance criteria for a risk-informed, performance-based PCSA are presented in NUREG-1804, *Yucca Mountain Review Plan, Final Report* (Ref. 5.1.7), which specifically discusses criticality as part of the PCSA. 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) "Verify that the appropriate properties and factors are considered in determining the adequacy of the hazard and initiating event identification, such as: (… conditions under which available fissionable material could pose a criticality hazard…" (Section 2.1.1.3.2)

3) "An Adequate List of Structures, Systems, and Components Identified as Being Important to Preclosure Radiological Safety…. (1) The analysis and classification of structures, systems, and components for the geologic repository operations area uses results of the hazard assessment, identification of event sequences, and consequence analyses as a basis to identify those structures, systems, and components that are important to safety; and (2) The analyses used to identify structures, systems, and components important to safety, safety controls, and measures to ensure the availability and reliability of the safety systems, include adequate consideration of:… (f) Means to prevent or control criticality;.." (Section 2.1.1.6.3)

4) "Verify that criticality design criteria are developed based on the consequence analysis results from the preclosure safety analysis. Confirm that criticality design criteria are factored into models and assumptions used for criticality analysis…." (Section 2.1.1.7.2.1).

1.4 WASTE ACCEPTANCE REQUIREMENTS

The repository acceptance requirements relating to criticality safety for canistered DOE SNF, Naval Nuclear Propulsion Program (NNPP) SNF, and HLW are given in *Waste Acceptance System Requirements Document* (WASRD), Sections 4.3.8, 4.4.8, and 4.8.11 (Ref. 5.1.4). In the case of commercial SNF (CSNF) and its associated packaging, waste acceptance criteria will be in place upon approval to receive and possess. The preclosure criticality requirement for all canistered and uncanistered SNF is that the SNF and canister designs, in conjunction with the facility SSCs, shall provide the basis for ensuring subcriticality at the time of delivery to the geologic repository and during all subsequent handling operations, including all event sequences that are important to criticality and have at least one chance in 10,000 of occurring before permanent closure. To provide assurance of subcriticality, the WASRD specifies that the methodology will account for the biases and uncertainties in both the calculations and experimental data used in the development of the effective neutron multiplication factor (k_{eff}), and will also include a justified administrative margin (Δk_m).

1.5 PURPOSE AND SCOPE

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 forms, canisters, waste packages 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 scope of this technical report is the complete process for performing preclosure criticality calculations 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. The particular waste forms anticipated for receipt at the repository include but are not limited to commercial SNF, HLW, and DOE SNF. With a focus on the safety requirements, the analyses will be performed for all processes starting with the receipt of canisters and/or transportation casks, the transfer of bare CSNF assemblies into canisters, aging of canisters, loading of canisters into waste packages, waste package emplacement, and waste package residence in the subsurface facilities up to the time of permanent closure of the repository. The detailed process is discussed in Section 3.

A discussion of 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. Conclusions are given in Section 4.

1.6 APPLICATION

Application of this process to preclosure criticality analyses will provide input to the PCSA that will demonstrate that the repository will meet its overall performance objectives for operations, including criticality, up to permanent repository closure.

The preclosure analysis process will be applied to criticality calculations and safety analyses. Using event tree/fault tree and reliability analyses in conjunction with validated effective neutron multiplication factor calculational methods, criticality calculations and safety analyses will demonstrate compliance with criticality design criteria to ensure that preclosure criticality is prevented for normal and for category 1 and category 2 event sequences.

1.7 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 the *Quality Management Directive* (Ref. 5.1.3), and the records designator for this report is noted as QA:QA. The development of this report is controlled by PA-PRO-0313, *Technical Reports* (Ref. 5.2.20).

1.8 USE OF COMPUTER SOFTWARE

No computer software subject to *Quality Management Directive* (Ref. 5.1.3) was used in the development of this report.

1.9 ASSUMPTIONS

There are no assumptions associated with this process report.

2. REGULATORY PERSPECTIVES

As stated in Section 1.4, 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 forms, canisters, waste packages and repository facilities for the preclosure time period. This section discusses the regulatory perspectives with respect to this process.

2.1 CRITICALITY SAFETY GUIDANCE

Since the regulatory requirements of *10 CFR Part 63*, Subpart E, Section 112(e)(6) (Ref. 5.2.1) for control of criticality are not specific, regulatory guidance and industry standards for criticality safety applicable to preclosure criticality are described in this section.

2.1.1 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 and NRC technical documents (NUREG series). The NRC documents reviewed in conjunction with the development of the preclosure criticality process are discussed briefly in this section. Unless explicitly stated, the cited guidance documents are applicable in whole without exceptions. Items that are further discussed are provided for clarity or amplification purposes.

NRC Regulatory Guide 3.71 Revision 1, *Nuclear Criticality Safety Standards for Fuels and Material Facilities* (Ref. 5.2.19)

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 regulations including 10 CFR Part 70 (Ref. 5.2.2). This regulatory guide endorses 11 ANSI/ANS-8 standards without exceptions and 4 ANSI/ANS-8 standards subject to specified NRC exceptions. These NRC exceptions and their applicability to this process report are discussed for each standard in Section 2.1.2. The approach presented in this report applies Regulatory Guide 3.71 to the same extent it applies the ANSI/ANS standards discussed in Section 2.1.2.

NUREG-1804, *Yucca Mountain Review Plan, Final Report* (Ref. 5.1.7)

The review methods and acceptance criteria provided in NUREG-1804 address 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* (Ref. 5.2.1), there is specific guidance for criticality design criteria in Section 2.1.1.7 of the NUREG-1804, namely:

1) Confirm that criticality design criteria are factored into models and assumptions used for criticality analysis. These criteria should be consistent with those given in NUREG-1567 *Standard Review Plan for Spent Fuel Dry Storage Facilities*

(Ref. 5.1.5) 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 *Nuclear Criticality Safety Standards for Fuels and Material Facilities* (Ref. 5.2.19).

2) Incorporate criticality design bases and criteria that include geometry, moderators, and k_{eff} limits, to ensure that nuclear fuel remains subcritical during handling, transfer, repackaging, storage, and retrieval.

3) 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.

NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (Ref. 5.1.5)

The guidance provided in NUREG-1567 *Standard Review Plan for Spent Fuel Dry Storage Facilities*, Glossary, Sections 4.5.3.5 and 8 (Ref. 5.1.5), addresses the NRC approach for reviewing criticality safety analyses for independent spent fuel storage installations (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 criteria applicable to the preclosure criticality process 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 (monitored retrievable storage) 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 (safety analysis report) 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…." (*Standard Review Plan for Spent Fuel Dry Storage Facilities*, Section 8.4.1.1 (Ref. 5.1.5).

2.1.2 Industry Standards

Several ANSI/ANS standards that are applicable to nuclear criticality safety have been reviewed for applicability to preclosure criticality safety in the geologic repository operations area (GROA). 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* (Ref. 5.2.19)) relating to nuclear criticality safety. Note that some of the standards have more recent reaffirmation dates than those listed in Regulatory Guide 3.71 (Ref. 5.2.19). Several ANSI/ANS standards are determined to be applicable to preclosure criticality safety in the GROA. Unless explicitly stated, the cited ANSI/ANS standard is applicable in whole without exceptions. Items that are further discussed are provided for clarity or amplification purposes.

ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, Section 2 (Ref. 5.2.3) 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 applies the guidance for prevention of criticality accidents provided in this standard. In addition, the single-parameter (such as mass, enrichment, volume, and concentration) and multi-parameter limits in the standard may be applicable to some waste forms and operations.

Regulatory Guide 3.71 *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, Section 2 (Ref. 5.2.19) provides further sufficiency clarification 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-8.1-1998 (Ref. 5.2.3), Section 4.2.2 states:

> "Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible."

The Double Contingency Principle has always been recognized as a guide to the proper degree of protection against a criticality accident. Section 4.1.2 of ANSI/ANS-8.1-1998 (Ref. 5.2.3) provides the following overarching requirement, i.e., a "shall statement":

> "Before a new operation with fissionable materials is begun or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions."

Both this "requirement" and the Double Contingency Principle "guidance" play prominent roles in the DOE and NRC criticality safety regulations. In all cases the goal is accident prevention and the Double Contingency Principle provides important guidance in achieving this goal. The design of the surface facilities is based on the principles of double contingency; whereas, the quantitative event sequence-based analysis demonstrates compliance with 10 CFR Part 63 (Ref. 5.2.1) (i.e., all operations shall be determined to be subcritical for normal operations and for individual event sequences with a mean probability of occurrence greater than or equal to one chance in 10,000 prior to permanent closure).

ANSI/ANS-8.3-1997 (Reaffirmed 2003), *Criticality Accident Alarm System*, Section 2 (Ref. 5.2.4) 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 (*Criticality Accident Alarm System*, Section 3.3 (Ref. 5.2.4)) or criticality accidents are demonstrated to be incredible. Regulatory Guide 3.71 *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, Section 2 (Ref. 5.2.19) 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."

The determination of the need for a criticality alarm system is a conclusion of the safety analysis; it cannot be determined a priori. Therefore, the criticality safety analysis will contain a thorough evaluation of the design and operations with fissile materials. This evaluation will be based on 10 CFR Part 63 (Ref. 5.2.1) preclosure criticality requirements, prescriptive NRC regulations for similar applications, and guidance from ANSI/ANS-8.3-1997 *Criticality Accident Alarm System* (Ref. 5.2.4). The results of the criticality safety analysis will determine the need for a criticality accident alarm system.

ANSI/ANS-8.5-1996 (Reaffirmed 2002), *Use of Borosilicate-Glass Raschig Rings as Neutron Absorber in Solutions of Fissile Material*, Section 1 (Ref. 5.2.5) 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), *Safety in Conducting Subcritical Neutron Multiplication Measurements in Situ*, Section 2 (Ref. 5.2.6) 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), *American National Standard for Nuclear Criticality Safety in the Storage of Fissile Materials*, Section 2 (Ref. 5.2.7) 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."

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. All preclosure operations will be determined to be subcritical for normal operations and for individual event sequences with a mean probability of occurrence greater than or equal to one chance in 10,000 prior to permanent closure using an appropriate effective neutron multiplication factor calculational method and a comparison of the maximum credible k_{eff} value to the configuration-specific upper subcritical limit (USL). Therefore, this standard is not applicable to repository operations.

ANSI/ANS-8.10-1983 (Reaffirmed 2005), *American National Standard Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement,* Section 2 (Ref. 5.2.8) 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."

All preclosure operations will be determined to be subcritical for normal operations and for individual event sequences with a mean probability of occurrence greater than or equal to one chance in 10,000 prior to permanent closure. Therefore, there will not be required reliance on shielding to protect against a criticality accident. Therefore, this standard is not applicable to repository operations.

ANSI/ANS-8.12-1987 (Reaffirmed 2002), *American National Standard for Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors*, Section 2 (Ref. 5.2.9) 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, *Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors***,** Section 2 (Ref. 5.2.10) 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 is applicable to this preclosure criticality process since soluble neutron absorbers are used in the Wet Handling Facility (WHF) pool for criticality control.

ANSI/ANS-8.15-1981 (Reaffirmed 2005), *Nuclear Criticality Control of Special Actinide Elements*, Section 2 (Ref. 5.2.11) states:

> "This standard is applicable to operations with the following:
>
> ${}^{237}_{93}$Np, ${}^{238}_{94}$Pu, ${}^{240}_{94}$Pu, ${}^{241}_{94}$Pu, ${}^{242}_{94}$Pu, ${}^{241}_{95}$Am, ${}^{242m}_{95}$Am, ${}^{243}_{95}$Am, ${}^{243}_{96}$Cm, ${}^{244}_{96}$Cm, ${}^{245}_{96}$Cm, ${}^{247}_{96}$Cm, ${}^{249}_{98}$Cf and ${}^{251}_{98}$Cf
>
> 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 (Ref. 5.2.3), and that may be encountered in sufficient quantities to be of concern for criticality. The repository will not be storing separate isolated units of the special actinide absorbers detailed in the standard. Therefore, this standard is not applicable to repository operations.

ANSI/ANS-8.17-2004, *American National Standard, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*, Section 2 (Ref. 5.2.12) states:

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

ANSI/ANS-8.17-2004, Section 5 Criteria to Establish Subcriticality (Ref. 5.2.12) is applicable to the process described in this report. The process for crediting neutron absorbers applies the guidance in this standard. NRC regulations and guidance documents have conservatively prescribed 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. Fixed neutron absorbers, such as borated aluminum alloys and borated stainless steel alloys, will be relied upon to ensure subcriticality in the surface facilities. These alloys have been used and accepted by the NRC to ensure criticality safety in spent fuel pools and transportation casks for decades.

Regulatory Guide 3.71 *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, Section 2 (Ref. 5.2.19), 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. Preclosure criticality safety will be demonstrated on the basis of the fresh fuel assumption (i.e., no burnup credit). Therefore, the exception to this standard is not applicable to the process described in this report.

ANSI/ANS-8.19-2005, *American National Standard, Administrative Practices for Nuclear Criticality Safety*, Section 2 (Ref. 5.2.13) 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.”

During the detailed design and construction of the repository, the nuclear criticality safety design functions are performed by the preclosure safety organization with close coordination with the engineering organization. This ensures nuclear criticality safety is integrated into the design process. Prior to the receipt and handling of waste forms, the criticality organization will include operational components. This nuclear criticality safety organization will be responsible for development and implementation of administrative practices, procedures, and training for nuclear criticality safety. The nuclear criticality safety organization will also be responsible for planning and implementing emergency response actions for nuclear criticality hazards or events, and thus applying ANSI/ANS-8.19-2005 (Ref. 5.2.13). Therefore, 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), *American National Standard, Nuclear Criticality Safety Training*, Section 2 (Ref. 5.2.14) 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), *American National Standard for the Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*, Section 2 (Ref. 5.2.15) 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 WHF 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, *American National Standard for Nuclear Criticality Safety Based on Limiting and Controlling Moderators*, Section 2 (Ref. 5.2.16) 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 materials."

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. Nuclear criticality safety practices as they relate to administrative and process evaluations for limitation and control of moderators will apply the guidance described in Section 4 of ANSI/ANS-8.22-1997 (Ref. 5.2.16). Engineered practices for moderator control will apply the guidance given in Section 5 of ANSI/ANS-8.22-1997 (Ref. 5.2.16).

ANSI/ANS-8.23-1997 (Reaffirmed in 1998), *American National Standard for Nuclear Criticality Accident Emergency Planning and Response*, Section 2 (Ref. 5.2.17) 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-8.24-2007, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, Section 2 (Ref. 5.2.18) states:

> "This standard provides requirements and recommendations for validation, including establishing applicability, of neutron transport calculational methods used in determining critical or subcritical conditions for nuclear criticality safety analyses."

This standard is applicable to the process described in this report for validating neutron transport calculational methods as described in Section 3.4.1.

2.1.3 Interim Staff Guidance

Interim Staff Guidance (ISG) documents reviewed in conjunction with the development of the preclosure criticality process are discussed briefly in this section. These documents are not directly applicable to preclosure design and operations; however they are used as guidance to ensure consistency with precedents and accepted practices in criticality safety.

Spent Fuel Project Office (SFPO)-ISG-1 Revision 1, *Interim Staff Guidance-1. Damaged Fuel* (Ref. 5.1.6)

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 specific guidance for storage given in SFPO-ISG-1 that will be applied for preclosure criticality analyses where damaged fuel needs to be considered is presented in the following two quotes:

> "**Discussion**: A criticality analysis for canned damaged fuel is typically performed by assuming a non-mechanistic redistribution of the fuel pellets into the most reactive geometry…"

> "(**Section) 2.4.2**: A fuel assembly with missing fuel pins shall be classified as damaged unless criticality analyses demonstrate an acceptable value of k_{eff} with the fuel pins missing."

Fuel Cycle Safety and Safeguards (FCSS)-ISG-10 Revision 0, *Fuel Cycle Safety and Safeguards – Interim Staff Guidance -10. Justification of Minimum Margin of Subcriticality for Safety* (Ref. 5.1.8)

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 USL calculations. As prescribed in FCSS-ISG-10, the following considerations are taken into account in the justification of an administrative margin for preclosure criticality safety analyses:

1) Validation results
2) Conservatisms in the calculational model
3) Likelihood of abnormal conditions
4) System sensitivity
5) Knowledge of neutron physics.

3. PROCESS APPROACH/METHOD

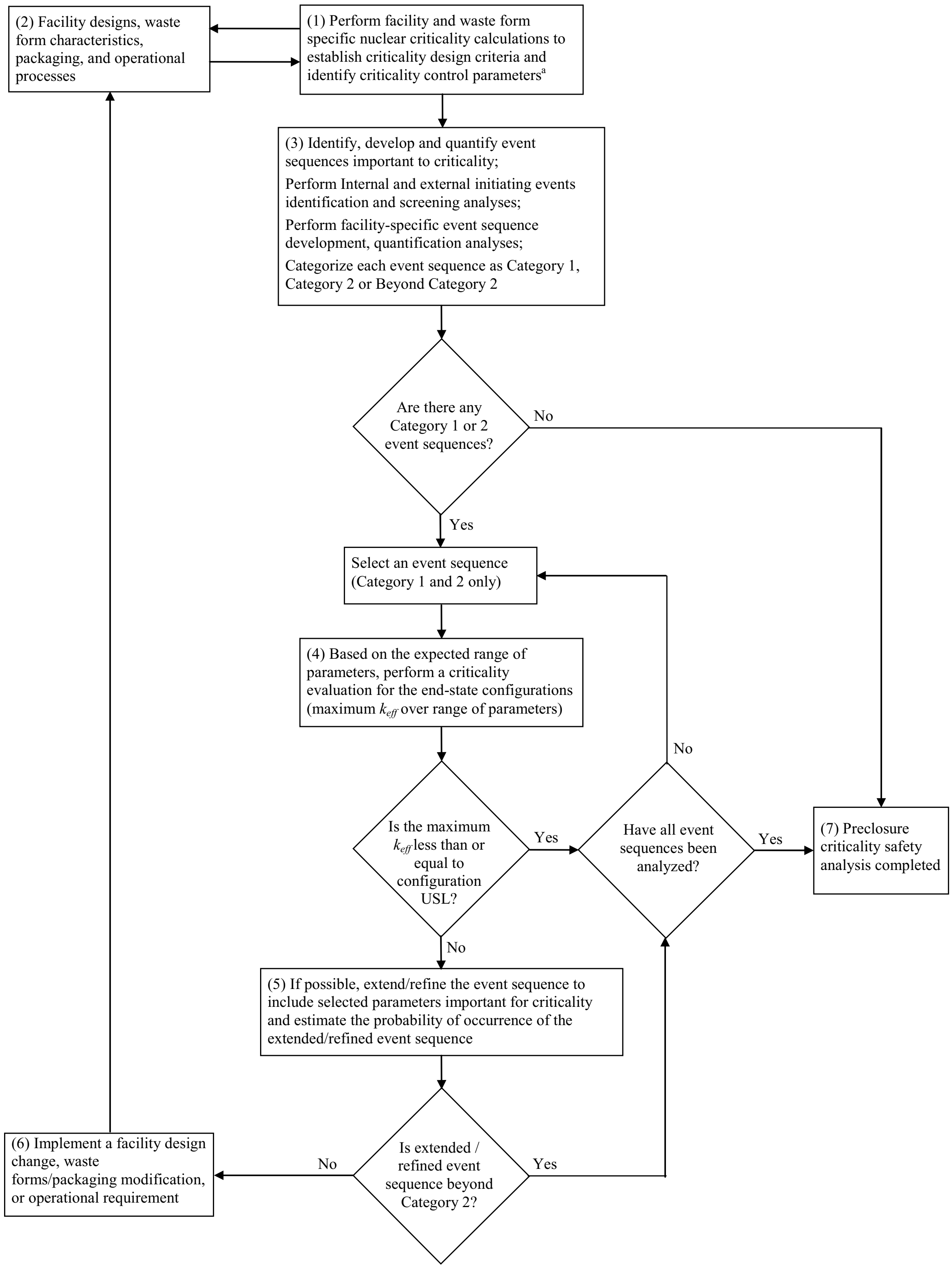
An overview of the principal elements of the preclosure criticality analysis process is given in Figure 3-1. The preclosure criticality analysis process is an iterative process among three primary disciplines:

1) Engineering - facility designs, waste form characteristics and operational processes;

2) Preclosure safety – initiating events and event sequence development, quantification and categorization; and

3) Criticality safety – criticality design criteria and control parameters.

The following subsections describe in detail each of the process steps, feed elements, and decision points depicted in Figure 3-1. The discussion includes the specific process steps, the required input needed by each process step, and the expected outputs to be generated by each process step.



[a] May include evaluation against single- and multi-parameter limits

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

3.1 BOX (1): NUCLEAR CRITICALITY CALCULATIONS

The purpose of these calculations is to establish criticality design and operational criteria and to identify criticality control parameters and their limits associated with the handling of each specific waste form in the surface and subsurface facilities. The criticality design criteria are established to ensure subcriticality for normal conditions and potential end states of Category 1 and Category 2 event sequences important to criticality.

In order to determine the criticality potential for each specific waste form and associated facility and handling operations, k_{eff} sensitivity calculations are performed. These calculations evaluate the impact on system reactivity of variations in each of the parameters important to criticality during the preclosure period. Given that the repository will handle existing waste forms without the ability to alter their form or packaging (for canistered waste forms), there are six parameters important to criticality that are evaluated as part of this process step:

1) **Waste form characteristics:** These criticality calculations will use either bounding or representative fuel characteristics. If waste form characteristics are not bounded in the calculations and the system being evaluated is subject to potential misloads based on the availability of more reactive SNF that can be mistakenly handled, then this parameter will be identified as needing to be controlled.

2) **Reflection:** These criticality calculations will cycle through all potential reflection conditions as a function of the other five parameters. These calculations will determine the impact of reflection on system reactivity and determine when reflection needs to be controlled.

3) **Moderation:** These criticality calculations will determine optimum moderation conditions (e.g, type, mass, volume, density) that maintain subcriticality as a function of all other five parameters. These calculations will determine the impact of moderation on system reactivity and the extent to which moderation needs to be controlled.

4) **Interaction – neutronic coupling**: These criticality calculations will cycle through all potential interaction conditions with the same or other waste form as a function of the other five parameters. These calculations will determine the impact of interaction on system reactivity and when interaction needs to be controlled.

5) **Neutron absorber (fixed and soluble):** These criticality calculations will determine minimum neutron absorber characteristics (e.g, type, loading, concentration) that maintain subcriticality as a function of all other five parameters. These calculations will determine the impact of neutron absorber on system reactivity and the extent to which neutron absorbers need to be controlled.

6) **Geometry:** These criticality calculations will cycle through all potential geometrical reconfiguration conditions as a function of the other five parameters. These calculations will determine the impact of geometry on system reactivity and when geometry needs to be controlled.

The criticality calculations in this process step determine the sensitivity of k_{eff} to variations in any parameter(s) as a function of the other parameters and provide guidance to event sequence development, quantification, and categorization analyses on whether each parameter (or its effect):

a) Does not need to be controlled because it is bounded (i.e., its analyzed value is greater than or equal to the design limit) or its effect is bounded,
b) Needs to be controlled if another parameter is not controlled (conditional control), or
c) Needs to be controlled because it is the primary criticality control parameter.

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

There are six facilities where waste forms will be handled, packaged, or stored:

1) Receipt Facility
2) Canister Receipt and Closure Facility
3) Initial Handling Facility
4) WHF
5) Intrasite operations including the Aging Facility
6) Subsurface Facility

The waste forms expected for receipt, handling, packaging and emplacement in the repository are:

1) Commercial SNF in sealed transportation, aging and disposal canisters (TADs), dual-purpose canisters or transportation casks

2) DOE SNF in sealed disposable canisters

3) HLW glass in sealed disposable canisters

4) NNPP SNF in sealed disposable canisters

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 with handling canisters
2) Operations with handling individual assemblies
3) Operations with handling waste packages.

These design, waste form and operational details are used in the criticality calculations described in Section 3.1 [Box (1)]. Given that this is an iterative process, design and operational detail may change based on the results of these calculations.

3.3 BOX (3): IDENTIFICATION AND CATEGORIZATION OF CRITICALITY EVENT SEQUENCES

Event sequences important to criticality include actions and/or occurrences within the repository operational facilities that could potentially lead to a criticality accident. An event sequence has 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.

Based on the criticality calculations described in Section 3.1 [Box (1)], the identification of event sequences will focus on those that impact any of the parameters identified as needing to be controlled. Some event sequences do not need to be identified, quantified or categorized (i.e., they are systematically screened out based on the criticality sensitivity calculations described in Section 3.1 [Box (1)]). These event sequences are those associated with parameters (or their effects) that are bounded and do not need to be controlled. The following provides a guide for how these parameters are considered in the event sequence evaluation:

1) **Waste form characteristics:** If waste form characteristics are not bounded in the calculations described in Section 3.1 [Box (1)] and the system being evaluated is subject to potential misloads, then event sequences that result in a misloaded system will be identified and quantified.

2) **Reflection:** If reflection is a parameter that is identified as needing to be controlled, then event sequences that introduce changes in reflection conditions will be identified and quantified. Based on the calculations described in Section 3.1 [Box (1)], reflection may need to be conditionally controlled. For example, reflection may need to be controlled only if moderator enters a breached canister, and therefore, event sequences that do not introduce moderator into a breached canister need not investigate potential for changes in reflection conditions.

3) **Moderation:** If moderation is a parameter that is identified as needing to be controlled, then event sequences that introduce moderator into the system being evaluated (e.g., breached canister) will be identified and quantified. For most canisters, moderator cannot be present in the canister without canister breach. For a few DOE SNF types, such as TRIGA SNF, the fuel matrix is self-moderated. Based on the calculations described in Section 3.1 [Box (1)], moderation may be the only parameter that needs to be controlled. For example, event sequences that introduce moderator into a breached TAD canister need not investigate potential impacts on other parameters; meaning that the event sequence results in criticality potential without any changes to reflection, neutron absorbers, geometry, waste form characteristics, or interaction.

4) **Neutron absorber (fixed and soluble):** If neutron absorber is a parameter that is identified as needing to be controlled, then event sequences that affect neutron

absorber effectiveness (e.g., dilution of soluble boron in the WHF pool, or reduction of fixed neutron absorber effectiveness due to damage or misplacement) will be identified and quantified. Based on the calculations described in Section 3.1 [Box (1)], neutron absorber may need to be conditionally controlled. For example, neutron absorber may need to be controlled only if moderation is present, and therefore, event sequences that do not result in moderation need not investigate potential for reduction in neutron absorber effectiveness.

5) **Interaction – neutronic coupling:** If interaction is a parameter that is identified as needing to be controlled, then event sequences that alter interaction conditions will be identified and quantified. Based on the calculations described in Section 3.1 [Box (1)], interaction conditions may not need to be controlled. For example, if the criticality calculations are performed with mirror or periodic boundary conditions, then event sequences that alter interaction conditions do not need to be investigated.

6) **Geometry:** If geometry is a parameter that is identified as needing to be controlled, then event sequences that alter geometry will be identified and quantified. Based on the calculations described in Section 3.1 [Box (1)], geometry may need to be conditionally controlled. For example, geometry may need to be controlled only if moderation is present, and therefore, event sequences that do not result in moderation need not investigate potential for geometrical reconfiguration.

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 PCSA. The performance of the SSCs and implementation of operational requirements are reviewed to verify that all sequences important to criticality have been identified. These reviews identify and describe the controls and procedures that are relied upon to prevent (i.e., limit the likelihood of) event sequences important to criticality from leading to a criticality accident. The analyses 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 to criticality will be included in the identification of event sequences important to safety. However, the list of event sequences provided in categorization analyses may not necessarily be strictly specific to criticality safety. The PCSA is a systematic examination of the site; the design; and the potential initiating events caused by underlying hazards. The PCSA is centered on the identification of internal and external initiating events and the event sequences emanating from them, which may result in potential radiological exposures to workers and the public or potential reactivity increases that might lead to criticality. Naturally occurring and human-induced initiating events that could occur at the GROA are systematically identified. A comprehensive list of internal and external initiating events is developed. External initiating events are initially screened to determine whether they are applicable to the repository. Both internal and external initiating events are screened. Possible event sequences initiated by only screened in initiating events and internal hazards are analyzed to determine whether they cause an event sequence.

Figure 3-2 illustrates the PCSA process as a flow diagram. It illustrates the interrelationship of various analyses integrated into the PCSA, including the interfaces with design. The PCSA applies elements of probabilistic risk analysis that are embedded in the structured, multi-tiered individual analyses of internal and external initiating events, event sequences, radiological consequences, and potential criticality. The PCSA is structured around three questions:

1) **What can happen?** The answer to this question concerns identification of event sequences. These begin with initiating events, which are a departure from normal operation, and from which pivotal events emanate. Pivotal events represent SSC and operational responses to initiating events. End states are the termini of event sequences.

2) **How likely is it?** The answer to this question concerns the identification of the number of expected occurrences over the preclosure period. This can also be expressed as a probability over the preclosure period. The mean number of occurrences over the preclosure period is compared to the Category 1 and Category 2 threshold values defined in 10 CFR 63, Subpart A, Section 2 (Ref. 5.2.1).

3) **What are the consequences?** The answer to this question concerns calculation of potential radiological doses to workers or the public or potential reactivity increases that might violate the subcriticality criterion USL.

These questions are frequently asked as the PCSA progresses through the event sequence analyses. The same questions guide the analysis of specific safety topics, including random and passive equipment failure, internal fire, internal floods, external initiating events, and human reliability analyses.

Categorization of event sequences is based on evaluated frequencies and documented in event sequence and quantification reports for each facility. Categorization of event sequences is achieved by comparing the mean value of each event sequence probability distribution to the Category 1 (an expected number of occurrences of at least one in the preclosure period) or Category 2 (a mean probability greater than or equal to one chance in 10,000 but an expected number of occurrences less than one in the preclosure period) event sequence criteria. Thus, an event sequence is either Category 1, Category 2, or beyond Category 2. Event sequences that are categorized as beyond category 2 are considered to be screened out from the requirement to conduct criticality analysis to demonstrate compliance with the requirements of 10 CFR 63.112(e)(6) (Ref. 5.2.1).

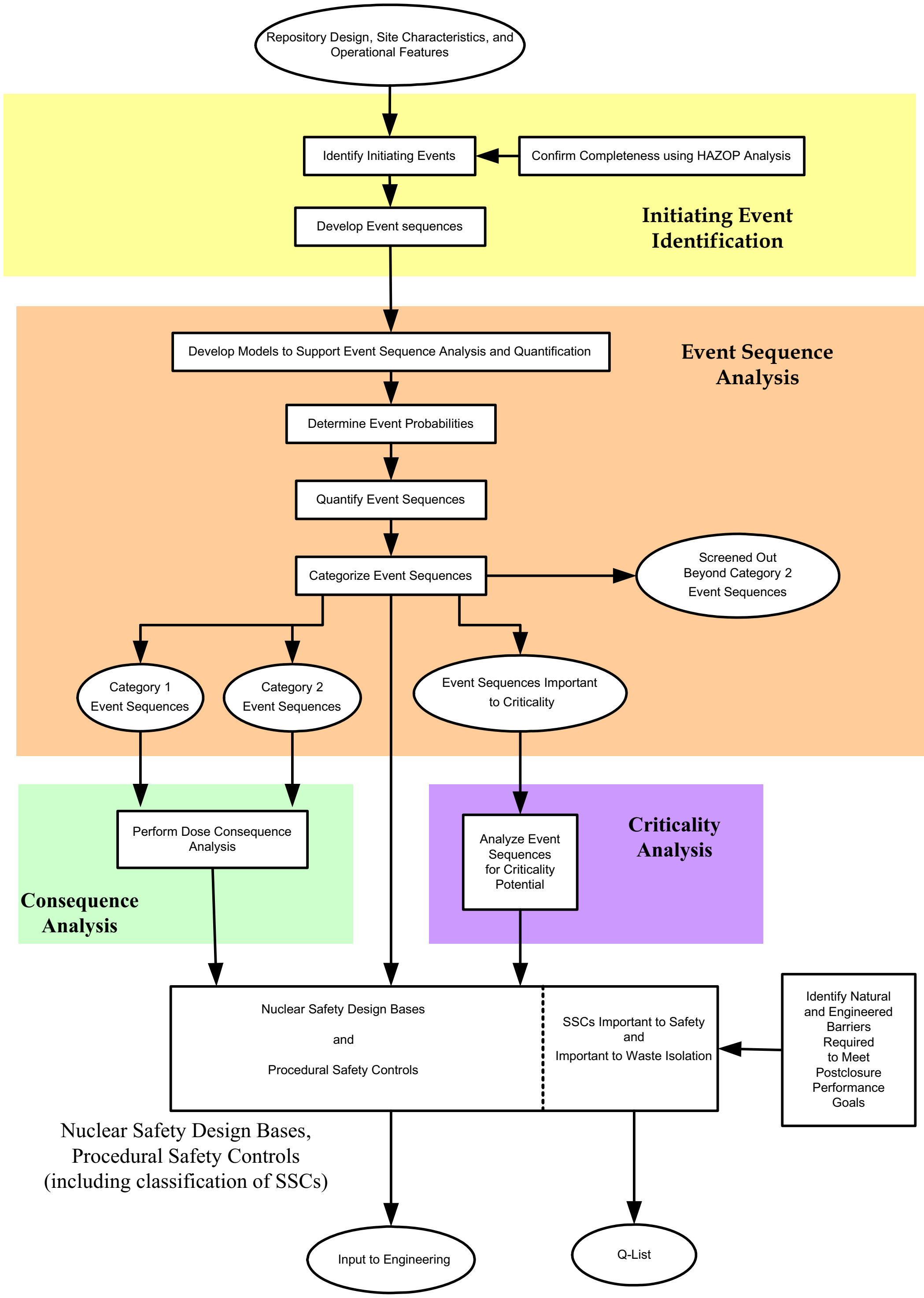


Figure 3-2. Overview of the PCSA Process

3.4 BOX (4): K-EFFECTIVE EVALUATIONS

If any of the end states resulting from event sequences important to 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 criticality evaluation process begins with the selection of parameters and parameter values that are obtained from event sequences important to 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 maximum k_{eff} plus calculational uncertainties is less than or equal to the configuration-specific USL. In equation notation, the use of the USL is:

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

$$\text{USL} = 1 - \text{sum of bias and uncertainties} - \text{administrative margin} \qquad \text{(Eq. 2)}$$

where,

k_S = calculated k_{eff} for the system

Δk_S = an allowance for:

1) 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),
2) Material and fabrication tolerances, and
3) Uncertainties due to the geometric or material representations used in the computational method. (**Note:** allowance for items (2) and (3) can be obviated by using bounding representations).

USL = an upper limit on k_{eff} characterized by statistical tolerance limits that accounts for:

1) Biases and uncertainties associated with the criticality code trending process,
2) 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
3) A justified administrative margin to ensure subcriticality.

3.4.1 Validation (Determination of the USL)

ANSI/ANS-8.17-2004, *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*, Section 5, (Ref. 5.2.12) 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 the result of acceptable validation of the calculational methods. The criticality validation process begins with the selection of parameters and parameter values that are obtained from normal operations and event sequences important to 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. The second step is to select benchmarks (critical experiments) that include, to the extent possible; neutronic and physical characteristics as nearly comparable to those of the end-state configuration(s). The set of critical experiments prescribes the basic range of applicability of the results. In ANSI/ANS-8.1-1998 (p. 1) (Ref. 5.2.3), the term "area of applicability" means:

> "The limiting ranges of material compositions, geometric arrangements, neutron energy spectra and other relevant parameters (such as heterogeneity, leakage, interaction, absorption, etc.) within which the bias of a calculational method is established."

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 justified administrative margin. The justification follows the guidance given in FCSS-ISG-10 *Fuel Cycle Safety and Safeguards – Interim Staff Guidance – 10. Justification for Minimum Margin of Subcriticality for Safety* (Ref. 5.1.8) and ANSI/ANS-8.24-2007, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, Section 6.4 (Ref. 5.2.18).

The USL is represented in equation form based on ANSI/ANS-8.17-2004, *Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors*, Section 5 (Ref. 5.2.12), as:

$$\text{USL} = \text{LBTL} - \Delta k_{EROA} - \Delta k_m \qquad \text{(Eq. 3)}$$

where

> LBTL = the lower-bound tolerance limit 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_m = an administrative margin to ensure subcriticality
>
> (**Note:** The relationship of the equations in this section to those in Section 5 of ANSI/ANS-8.17-2004 *Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors*, Section 5 (Ref. 5.2.12) is shown in Appendix B.)

The LBTL may be expressed as a regression-based function of neutronic or physical variable(s), or both. In application, a LBTL could also be either: (1) a single value, reflecting a conservative result over the range of applicability for the waste form characterized, or (2) a function of a trending parameter (or a predictor variable) for the experiments. Because the USL can vary with this parameter, it is expressed as a function of this parameter within an appropriate range of applicability derived from the parameter bounds.

The validation as part of the application of the process described in this report applies the guidance of ANSI/ANS-8.24-2007 (Ref. 5.2.18) in whole.

3.5 BOXES (5 AND 6): EXTENSION/REFINEMENT OF EVENT SEQUENCES IMPORTANT TO CRITICALITY

For end-state configurations where the maximum k_{eff} value exceeds the USL, and the probability of occurrence of the end-state configuration exceeds the Category 2 criterion, the event sequence is further extended or refined to credit additional design features or procedural safety controls such that the event sequence probability is reduced. The probability of the extended or refined event sequence may include the additional probability of occurrence of parameters important to criticality, such as degree of moderation, extent of fuel rearrangement, and fuel basket geometric reconfiguration (Box [5]). The end-state configuration is acceptable provided that the probability of occurrence of the extended or refined event sequence does not exceed the Category 2 screening criterion. If the probability of an extended or refined event sequence exceeds the Category 2 screening criterion and the maximum k_{eff} value exceeds the configuration-specific USL, design or operational requirements will be imposed to reduce the probability of the event sequence to below the Category 2 screening criterion (Box [6]).

The analysis process is continued until all facilities and waste forms have been evaluated, criticality control parameters established, and event sequences important to criticality have been identified and evaluated as acceptable.

3.6 BOX (7): COMPLETION OF PRECLOSURE CRITICALITY ANALYSIS

The surface and subsurface facility designs are acceptable with respect to criticality when: (1) each event sequence important to criticality has been shown to have a probability less than one chance in 10,000 during the preclosure period, (2) the maximum k_{eff} for end-state configurations of all event sequences important to criticality with a mean probability greater than or equal to one chance in 10,000 during the preclosure period is less than the configuration-specific USL, or (3) the configuration meets the single- or multi-parameter limits established in Sections 5 and 6 of ANSI/ANS-8.1 1998 (Ref. 5.2.3).

The SSCs that are relied upon to demonstrate subcriticality and to ensure that event sequences that violate the identified criticality control parameters have a mean probability of occurrence less than one chance in 10,000 during the preclosure period will be identified as important to safety (ITS).

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 forms (including canisters and waste packages) 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 preclosure event sequence that could violate the established criticality safety criteria will be less than the Category 2 screening criterion.

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APPENDIX A
GLOSSARY

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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 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, with which the k_{eff} of the configuration being analyzed for criticality potential will be compared.

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

Event sequence important to criticality is an event sequence that impacts any of the criticality parameters identified as needing to be controlled

Safety systems are SSCs that are identified to be important to safety.

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APPENDIX B
RELATIONSHIP OF SUBCRITICALITY CRITERIA

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RELATIONSHIP OF SUBCRITICALITY CRITERIA

The relationship of the equations in Section 3.4 of this report to those in Section 5 of ANSI/ANS-8.17-2004, *American National Standard, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors* (Ref. 5.2.12) is shown below:

The equation in Section 5.1 of ANSI/ANS-8.17-2004 is:

$$k_p \quad k_c - \Delta k_p - \Delta k_c - \Delta k_m \qquad \text{(Eq. B-1)}$$

Moving Δk_p to the left side gives:

$$k_p + \Delta k_p \quad k_c - \Delta k_c - \Delta k_m \qquad \text{(Eq. B-2)}$$

Equations 1 and 3 in Section 3.4 of this report are:

$$k_S + \Delta k_S \leq \text{USL} \qquad \text{(Eq. B-3)}$$

and

$$\text{USL} = \text{LBTL} - \Delta k_{EROA} - \Delta k_m \qquad \text{(Eq. B-4)}$$

Thus,

$$k_S + \Delta k_S \leq \text{LBTL} - \Delta k_{EROA} - \Delta k_m \qquad \text{(Eq. B-5)}$$

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

$$k_p + \Delta k_p = k_S + \Delta k_S \qquad \text{(Eq. B-6)}$$

and

$$\text{f(x)} - \Delta k_{EROA} - \Delta k_m = k_c - \Delta k_c - \Delta k_m \qquad \text{(Eq. B-7).}$$

The description of f(x) – Δk_{EROA} in Section 3.4 of this process report and the description of $k_c - \Delta k_c$ in Section 5 of ANSI/ANS-8.17-2004 are the same. They are the result of validating the criticality codes for a specific configuration to be analyzed with applicable critical benchmarks.