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DOE STANDARD

DOCUMENTED SAFETY ANALYSIS FOR DOE REACTOR FACILITIES

U.S. Department of Energy **AREA SAFT**
Washington, DC 20585

FOREWORD

1. This Department of Energy (DOE) Standard (STD) has been approved to be used by DOE, including the National Nuclear Security Administration, and their contractors.

2. Beneficial comments (recommendations, additions, and deletions), as well as any pertinent data that may be of use in improving this document, should be emailed to nuclearsafety@hq.doe.gov or addressed to:

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3. Title 10 of the Code of Federal Regulations (C.F.R.) Part 830, *Nuclear Safety Management*, Subpart B, *Safety Basis Requirements*, establishes requirements for the documented safety analyses (DSA) for nuclear facilities. This Standard provides an acceptable methodology for meeting the 10 C.F.R. Part 830 requirements for the preparation of DSAs for reactor facilities and constitutes an approved "successor document" for the method for reactor facilities described in Appendix A of Subpart B of 10 C.F.R. 830.

4. This Standard is not required to be used for the development of new DOE reactor DSAs or the revision of existing DOE reactor DSAs. However, if a facility, site, or program office chooses to use this Standard for revising an existing DSA, then this Standard requires implementation in its entirety (i.e., all applicable "shall" statements are met) if it is used as the safe harbor.

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1.0 INTRODUCTION

1.1 Purpose

This Department of Energy (DOE) Standard (STD) describes an acceptable methodology for preparing a Documented Safety Analysis (DSA) for a reactor facility. This Standard is a DOE-approved methodology for meeting the requirements in Title 10 of the Code of Federal Regulations (C.F.R.) Part 830, *Nuclear Safety Management,* Subpart B, *Safety Basis Requirements*.

A DSA is required for each DOE reactor in accordance with 10 C.F.R. 830, *Nuclear Safety Management*, Subpart B, *Safety Basis Requirements*. Table 2 of Appendix A to Subpart B provides a listing of acceptable DSA methodologies (also known as safe harbor methods[1]) for various types of nuclear facilities, including DOE reactors. The existing Table 2 lists the following for DOE reactors:

> "*Using the method in U.S. Nuclear Regulatory Commission Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, or successor document.*"

Nuclear Regulatory Commission (NRC) Regulatory Guide (Reg. Guide) 1.70, Rev. 3, was issued in February 1972 and last revised in November 1978. In September 2014, NRC staff recommended withdrawal of NRC Reg. Guide 1.70 and it is slowly being replaced with NRC Reg. Guide 1.206, *Applications for Nuclear Power Plants*. Since DOE will continue to operate research reactors and has plans to design, build, and operate new reactors, DOE has developed this Standard to describe a pre-approved, acceptable method for satisfying the requirements of 10 C.F.R. 830 for DOE reactors. This method can be maintained by DOE without reliance on an external organization.

1.2 Applicability

This Standard applies to DOE reactors as defined by 10 C.F.R. Part 830, *Nuclear Safety Management*. The DOE definition of reactors in 10 C.F.R. Part 830 is relatively broad and includes "research, test, and power reactors, and critical and pulsed assemblies and any assembly that is designed to perform subcritical experiments that could potentially reach criticality." This

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[1] Safe harbor methods (or safe harbors) are methods for developing DSAs that have already been determined by DOE to be acceptable for use. This concept and usage is described in the supplementary material to the rulemaking for 10 C.F.R. 830, Subpart B; see Interim Final Rule published on October 10, 2000 (Comment L, Section 830.204, page 65 FR 60300). The safe harbor concept is further discussed in DOE Guide 421.1-2A, *Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 C.F.R. 830.*

Standard is intended to be technology-neutral (i.e., its applicability is independent of the specific type of technology used in the reactor facility).

1.3 Background/Use of this Methodology

Title 10 C.F.R. Part 830, Subpart B, Section 830.204(a) requires that, "The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must obtain approval from DOE for the methodology used to prepare the documented safety analysis for the facility unless the contractor uses a methodology set forth in Table 2 of Appendix A to this Part." This Standard provides a DOE-approved methodology used to prepare a DSA for a DOE reactor facility. If the contractor decides to use this methodology, all applicable requirement statements in this Standard shall be met. If the contractor is unable or unwilling to meet all applicable requirements in this Standard, the contractor may develop an alternate methodology for DOE review and approval in accordance with DOE-STD-1083-2009, *Processing Exemptions to Nuclear Safety Rules and Approval of Alternative Methods for Documented Safety Analyses.*

The methodology described in this Standard has been used since 2001 at DOE reactors. DSAs for DOE reactors have consistently augmented NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants* – the established safe harbor method for reactors identified in Appendix A of Subpart B to 10 C.F.R. Part 830– with additional industry and DOE technical standards as needed to address identified hazards and meet the DSA requirements of 10 C.F.R. Part 830, Subpart B. This Standard codifies and formalizes current practice, and better describes the methods used to establish DSAs for DOE reactor facilities.

DOE Guide (G) 421.1-2A, *Implementation Guide for Use in Developing Documented Safety Analysis to Meet Subpart B of 10 C.F.R. 830*, recognizes the practice of augmenting NRC Regulatory Guide 1.70 to provide a more complete method for developing DOE reactor DSAs:

> "*Most DOE large reactors use [NRC] Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. There is an ANSI/ANS standard that provides guidance for small research reactors (ANSI/ANS-15.21, I). NUREG-1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, also provides guidance for non-power reactors. However, none of these reactor formats was written for DOE reactors and each has left out several topics that should be included. For DOE reactors, in addition to the topics discussed in Regulatory Guide 1.70, hazard analysis and categorization of the facility and applicable facility design codes and standards should be added. DOE-STD-3009 provides specific guidance for the content and organization DOE expects for these additional topics. DSAs for reactors often use different safety classification terminology (e.g., conforming*

to NRC Regulatory Guide 1.70) rather than that identified in 10 C.F.R. 830. [DOE G 421.1-2A, Section 4.2.1]"

DOE Order (O) 413.3B, *Program and Project Management for the Acquisition of Capital Assets,* provides requirements for delivering major DOE projects that meet safety requirements. This Order includes requirements for developing DSAs and Preliminary DSAs prior to critical decision points in the project life-cycle.

1.4 Existing DOE Reactors with Approved DSAs

Table 2 in Appendix A to Subpart B of 10 C.F.R Part 830allows use of NRC Regulatory Guide 1.70 as a general safe harbor method for DOE reactor facilities. Contractors with existing reactor facilities that have DOE-approved DSAs may continue operations under those DSAs. There is no requirement or need for an existing, approved reactor DSA to be revised or revisited in light of the issuance of this Standard. However, if a DOE Program Office chooses to use this Standard as a safe harbor for upgrading an existing DSA, then it shall be implemented in its entirety, (i.e., all applicable "shall" statements are met). (Note: this Standard is not required to be used for new or existing reactors).

1.5 Overview of the Standard

Section 2 describes the terminology used in the Standard, including acronyms and abbreviations, requirements and recommendation statements, and definitions.

Section 3 provides an approach and requirements for DSA development that first categorizes a current or planned reactor facility into a general reactor type to define a pre-approved safety basis strategy.

Section 4 provides an approach and requirements for DSA development that is based on first developing a Safety Basis Strategy using a structured set of requirements. Either Section 3 or Section 4 is required to establish DSA development requirements.

Section-5 provides specific requirements for developing the DSA once the DSA development requirements have been established.

Section 6 provides references.

2.0 TERMINOLOGY

2.1 Shall, Should, and May

The word "shall" denotes a requirement (i.e., actions to be performed to satisfy the method defined by this Standard); the word "should" denotes a recommendation; and the word "may" denotes permission, neither a requirement nor a recommendation.

2.2 Applicability of Requirements in Standards

This Standard identifies other industry and DOE technical standards that are required to be used "as applicable." The meaning of this term "as applicable" is that the design and safety analysis requirements of an identified standard shall be used if they are relevant to the specific facility (i.e., they provide design and safety analysis requirements that ensure that desired functions of structures, systems, and components (SSC) are achieved, and these requirements are appropriate for the design materials, configuration, and service conditions). The stated applicability of industry codes and standards should not be used to narrowly interpret relevancy for SSC design and safety analysis.

2.3 Acronyms and Abbreviations

CDNS	Chief of Defense Nuclear Safety
C.F.R.	Code of Federal Regulations
CNS	Chief of Nuclear Safety
DOE	Department of Energy
DSA	Documented Safety Analysis
G	Guide
HDBK	Handbook
HC	Hazard Category
MAR	Material at Risk
NRC	Nuclear Regulatory Commission
O	Order
SMP	Safety Management Program
SAC	Specific Administrative Control
SBS	Safety Basis Strategy
SDS	Safety Design Strategy
STD	Standard
SSCs	Structures, Systems, and Components
TSR	Technical Safety Requirement

2.4 Definitions

The definitions presented below are provided for understanding and consistency among the various safe harbor methodologies. The origins of the definitions are indicated by references shown in square brackets [ ]. If no reference is listed, the definition originates in this Standard.

Category A Reactor. Those production, test, and research reactors designated by DOE based on power level (i.e., design thermal power rating of 20 megawatts steady state and higher), potential fission product inventory, and experimental capability. Category A reactors are Hazard Category 1 nuclear facilities. [DOE-STD-1027-2018, Section 2.3]

Category B Reactor. A reactor as defined by 10 C.F.R. Part 830, Section 830.3, that is not a Category A Reactor. Category B reactors are Hazard Category 2 nuclear facilities. [DOE-STD-1027-2018, Section 2.3]

Critical Assembly. Special nuclear devices designed and used to sustain nuclear reactions, which may be subject to frequent core and lattice configuration change and which frequently may be used as mockups of reactor configurations. [10 C.F.R. Part 830.3][2]

Documented Safety Analysis (DSA). A documented analysis of the extent to which a nuclear facility can be operated safely with respect to workers, the public, and the environment, including a description of the conditions, safe boundaries, and hazard controls that provide the basis for ensuring safety. [10 C.F.R. Part 830.3]

Fissionable Materials. A nuclide capable of sustaining a neutron-induced chain reaction (e.g., uranium-233, uranium-235, plutonium-238, plutonium-239, plutonium-241, neptunium-237, americium-241, and curium-244). [10 C.F.R. Part 830.3]

Graded Approach. The process of ensuring that the level of analysis, documentation, and actions used to comply with a requirement in this Standard is commensurate with:

- The relative importance to safety, safeguards, and security;
- The magnitude of any hazards involved;
- The life cycle stage of a facility;

[2] The 10 C.F.R. 830 definition is controlling for this Standard. However, the ANSI/ANS-1-2000 definition of critical assembly/experiment is complementary and provides added clarification: "Critical assembly. A device or physical system for performing critical experiments. In a critical assembly, the energy produced by fission is insufficient to require auxiliary cooling, and the power history is such that the inventory of long-lived fission products is insignificant. Critical experiment. An experiment or series of experiments performed with a fissionable material configuration which may be at or near critical. The principal purpose of the experiment is the study of neutron behavior within the critical assembly."

- The programmatic mission of a facility;
- The particular characteristics of a facility;
- The relative importance of radiological and non-radiological hazards; and
- Any other relevant factor. [10 C.F.R. Part 830.3]

Hazard. A source of danger (i.e., material, energy source, or operation) with the potential to cause illness, injury, or death to a person or damage to a facility or to the environment (without regard to the likelihood or credibility of accident scenarios or consequence mitigation). [10 C.F.R. Part 830.3]

Hazard Analysis. The identification of materials, systems, processes, and plant characteristics that can produce undesirable consequences (hazard identification), followed by the assessment of hazardous situations associated with a process or activity (hazard evaluation). Qualitative techniques are usually employed to pinpoint weaknesses in design or operation of the facility that could lead to accidents. The hazard evaluation includes an examination of the complete spectrum of potential accidents that could expose members of the public, onsite workers, facility workers, and the environment to radioactive and other hazardous materials. [DOE-STD-3009-2014]

Hazard Categorization. Evaluation of the consequences of unmitigated radiological releases to categorize facilities in accordance with the requirements of 10 C.F.R. Part 830. Note: 10 C.F.R. Part 830 requires categorization consistent with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. [DOE-STD-3009-2014]

Hazard Controls. Measures to eliminate, limit, or mitigate hazards to workers, the public, or environment, including: (1) physical design, structural, and engineering features; (2) safety structures, systems, and components; (3) safety management programs; (4) technical safety requirements; and (5) other controls necessary to provide adequate protection from hazards. [10 C.F.R. Part 830.3] Note: “hazard controls” include “specific administrative controls.” [DOE-STD-3009-2014]

Hazard Scenario. An event or sequence of events associated with a specific hazard, having the potential to result in undesired consequences identified in the hazard evaluation. [DOE-STD-3009-2014]

Mitigative Control. Any structure, system, component, or administrative control that serves to mitigate the consequences of a release of radioactive or other hazardous materials in a hazard or accident scenario. [DOE-STD-3009-2014]

Nuclear Facility. A reactor or a nonreactor nuclear facility where an activity is conducted for or on behalf of DOE and includes any related area, structure, facility, or activity to the extent necessary to ensure proper implementation of the requirements established by 10 C.F.R. Part 830. [10 C.F.R. Part 830.3]

Preventive Control. Any structure, system, component, or administrative control that eliminates the hazard; terminates the hazard scenario or accident; or reduces the likelihood of a release of radioactive and/or hazardous materials. [DOE-STD-3009-2014]

Public. All individuals outside the DOE site boundary. [DOE-STD-3009-2014]

Pulsed Reactor. Nuclear devices designed and operated for purposes of research and as sources of sharp, intense pulses of fission-produced radiation. In the usual operation, prompt super-criticality is established in a mass of unmoderated or under moderated fissile metal, radiation is produced, and the nuclear reaction is immediately terminated by characteristics inherent in the fissile material itself or the physical configuration. These devices are also known as fast pulse reactors.

Reactor. Any apparatus that is designed or used to sustain nuclear chain reactions in a controlled manner such as research, test, and power reactors, and critical and pulsed assemblies and any assembly that is designed to perform subcritical experiments that could potentially reach criticality; and, unless modified by words such as containment, vessel, or core, refers to the entire facility, including the housing, equipment and associated areas devoted to the operation and maintenance of one or more reactor cores. [10 C.F.R. Part 830.3]

Risk. The quantitative or qualitative expression of possible loss that considers both the likelihood that an event will occur and the consequences of that event. [DOE-STD-3009-2014]

Safety Basis. The documented safety analysis and hazard controls that provide reasonable assurance that a DOE nuclear facility can be operated safely in a manner that adequately protects workers, the public, and the environment. [10 C.F.R. Part 830.3]

Safety Class Structures, Systems, and Components (SC SSCs). Structures, systems, or components, including portions of process systems, whose preventive or mitigative function is necessary to limit radioactive hazardous material exposure to the public, as determined from safety analyses. [10 C.F.R. Part 830.3]

Safety Management Program. A program designed to ensure that a facility is operated in a safe manner that adequately protects workers, the public, and the environment by covering a topic such as quality assurance; maintenance of safety systems; personnel training; conduct of operations; inadvertent criticality protection; emergency preparedness; fire protection; waste management; or radiological protection of workers, the public, and the environment. [10 C.F.R. Part 830.3]

Safety Significant Structures, Systems, and Components (SS SSCs). Structures, systems, and components which are not designated as safety class SSCs, but whose preventive or mitigative function is a major contributor to defense-in-depth and/or worker safety as determined from safety analyses. [10 C.F.R. Part 830.3]

Safety Structures, Systems, and Components (Safety SSCs). Both safety class structures, systems, and components, and safety significant structures, systems, and components. [10 C.F.R. Part 830.3]

Site Boundary. For the purpose of implementing this Standard, the DOE site boundary is a geographic boundary within which public access is controlled and activities are governed by DOE and its contractors, and not by local authorities. A public road or waterway traversing a DOE site is considered to be within the DOE site boundary if DOE or the site contractor has the capability to control, when necessary, the road or waterway during accident or emergency conditions. [DOE-STD-3009-2014]

Specific Administrative Control. An administrative control that is identified to prevent or mitigate a hazard or accident scenario and has a safety function that would be safety significant or safety class if the function were provided by a structure, system or component. Note: DOE-STD-1186-2016, Specific Administrative Controls, or successor document, provides additional information about SACs. [DOE-STD-3009-2014]

Subcritical Nuclear Assembly. Any nuclear assembly that is designed to perform subcritical experiments that could potentially reach criticality (i.e., a self-sustaining nuclear chain reaction)[3].

Technical Safety Requirements (TSRs). The limits, controls, and related actions that establish the specific parameters and requisite actions for the safe operation of a nuclear facility and include, as appropriate for the work and the hazards identified in the DSA for the facility: safety limits, operating limits, surveillance requirements, administrative and management controls, use and application provisions, and design features, as well as a bases appendix. [10 C.F.R. Part 830.3]

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[3] If a subcritical nuclear assembly is an adjunct or an experimental package related to a reactor, and not a stand-alone reactor/facility, its safety basis would be evaluated in the context of the reactor and not uniquely. If a subcritical experiment or facility has been evaluated to be inherently sub-critical, with no credible potential to reach criticality, then this subcritical assembly or facility does not fit within the reactor definition and this Standard does not apply.

3.0 DSA DEVELOPMENT REQUIREMENTS

This Standard provides a safe harbor methodology to develop DSAs in compliance with 10 C.F.R. Part 830, Subpart B. The approach described below may be used based on the type of reactor. No specific Safety Basis Strategy (SBS) is required to be developed and approved for this approach, however a SBS may be developed and can be useful to summarize the approach. It is anticipated that most reactors using this Standard will use this approach.

3.1 Types of Reactors

This approach first categorizes a current or planned reactor facility into a general reactor type to define a pre-approved safety basis strategy. The most relevant reactor type shall be identified for the DSA:

(1) Subcritical Nuclear Assemblies;

(2) Critical Assemblies;

(3) Other Category B Reactors; and

(4) Category A Reactors.

3.2 Subcritical Nuclear Assemblies

The following requirements provide the pre-approved approach for developing DSAs for subcritical nuclear assemblies.

<u>DSA Format and Content</u>: DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,* (Section 4) shall be applied for the DSA Format. This DSA format requires discussion of criticality safety controls; any exemptions to the requirements in the ANSI/ANS 8 series of standards that need to be taken in order to allow mission work need to be addressed in DSA Chapter 6*, Prevention of Inadvertent Criticality.* In addition to description of criticality safety, SSCs and processes that are necessary for reactor safety shall be clearly identified. DSA discussion shall be provided to reflect the implementation of requirements and guidance from ANSI/ANS-1-2000 including a discussion of provisions for addressing the ability to shut down in the event of an inadvertent criticality and to confirm that no significant decay heat will be produced.

<u>Supporting Industry Standards</u>: ANSI/ANS-1-2000 and ANSI/ANS-14.1-2004 shall be used, as applicable.

<u>Design Criteria</u>: DOE O 420.1C applies. Attachment 3 requires identification of specific reactor design criteria.

<u>Hazard Identification and Evaluation</u>: DOE-STD-3009-2014 (Section 3.1) shall be applied.

Design Basis Accident Identification: DOE-STD-3009-2014 (Section 3.2.1) shall be applied.

Accident Consequence Analysis: DOE-STD-3009-2014 (Sections 3.2.2 – 3.2.4) shall be applied.

Hazard Control Selection: DOE-STD-3009-2014 (Section 3.3) shall be applied.

SSC Classification Hierarchy: DOE-STD-3009-2014 (Section 3.3) shall be applied.

Probabilistic Risk Assessment: Not expected.

Defense in Depth: DOE-STD-3009-2014 (Section 3.3.2) shall be applied. See also DOE-STD-3009-2014 Appendix A.9 and DOE G 420.1-1 for additional information.

3.3 Critical Assemblies

The following requirements provide the pre-approved approach for developing DSAs for critical assemblies.

DSA Format and Content: DOE-STD-3009-2014 (Section 4) shall be applied for the general DSA Content. ANSI/ANS-15.21-2012, "Format and Content for Safety Analysis Reports for Research Reactors," may be used to supplement the DSA format and content as appropriate.

Supporting Industry Standards: ANSI/ANS 1-2000 and ANSI/ANS-14.1-2004 shall be used, as applicable.

Design Criteria: DOE O 420.1C applies. Attachment 3 requires identification of specific reactor design criteria. ANSI/ANS-1-2000, and ANSI/ANS-14.1-2004, should be considered for application.

Hazard Identification and Evaluation: DOE-STD-3009-2014 (Section 3.1) shall be applied. The following standards may be used, as applicable, to supplement hazard evaluation methodologies: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.

Design Basis Accident Identification: DOE-STD-3009-2014 (Section 3.2.1) shall be applied. The following standards may be used, as applicable, for guidance in identifying design basis accidents: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.

Accident Consequence Analysis: DOE-STD-3009-2014 (Sections 3.2.2 – 3.2.4) shall be applied. The following standards may be used, as applicable, for accident analysis guidance: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.

Hazard Control Selection: DOE-STD-3009-2014 (Section 3.3) shall be applied.

SSC Classification Hierarchy: DOE-STD-3009-2014 (Section 3.3) shall be applied.

Probabilistic Risk Assessment: Not expected.

Defense in Depth: DOE-STD-3009-2014 (Section 3.3.2) shall be applied. See also DOE-STD-3009-2014 Appendix A.9 and DOE G 420.1-1 for additional information.

3.4 Other Category B Reactors

The following requirements provide the pre-approved approach for developing DSAs for other category B reactors (not including subcritical nuclear assemblies and critical nuclear assemblies that meet the 10 C.F.R. 830 definition of reactors).

DSA Format and Content: DOE-STD-3009-2014 (Section 4) shall be applied for the general DSA Content. ANSI/ANS-15.21-2012 "Format and Content for Safety Analysis Reports for Research Reactors," ANSI/ANS-14.1-2004, and NUREG-1537 may be used to supplement the DSA format and content as appropriate.

Supporting Industry Standards: ANSI/ANS-1-2000, ANSI/ANS-15.21, and ANS 14.1-2004 shall be used, as applicable.

Design Criteria: DOE O 420.1C applies; Attachment 3 requires identification of specific reactor design criteria. ANSI/ANS-1-2000, and ANSI/ANS-14.1-2004, should be considered for application.

Hazard Identification and Evaluation: DOE-STD-3009-2014 (Section 3.1) shall be applied. The following standards may be used, as applicable, to supplement hazard evaluation methodologies: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.

Design Basis Accident Identification: DOE-STD-3009-2014 (Section 3.2.1) shall be applied. The following standards may be used, as applicable, for guidance in identifying design basis accidents: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.

Accident Consequence Analysis: DOE-STD-3009-2014 (Section 3.2.2 – 3.2.4) shall be applied. The following standards may be used, as applicable, for accident analysis guidance: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.

Hazard Control Selection: DOE-STD-3009-2014 (Section 3.3) shall be applied.

SSC Classification Hierarchy: DOE-STD-3009-2014 (Section 3.3) shall be applied. SSCs that protect the reactor core shall be designated at least safety-significant.

Probabilistic Risk Assessment: Not expected.

Defense in Depth: DOE-STD-3009-2014 (Section 3.3.2) shall be applied. See also DOE-STD-3009-2014 Appendix A.9 and DOE G 420.1-1 for additional information.

3.5 Category A Reactors

The following requirements provide the pre-approved approach for developing DSAs for category A reactors.

DSA Format and Content: NUREG-1537 shall be applied for the general DSA Content. ANS-15.21-2012 and DOE-STD-3009-2014 (Section 4) may be used to supplement the DSA format and content.

Supporting Industry Standards: DOE-STD-3009-2014, NUREG-1537, ANSI/ANS-15.21, and ANS 1-2000 shall be used, as applicable.

Design Criteria: DOE O 420.1C applies; Attachment 3 requires identification of specific reactor design criteria. NRC Regulatory Guide 1.232, Rev 0, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, shall be evaluated for applicability and use.

Hazard Identification and Evaluation: DOE-STD-3009-2014 (Section 3.1) shall be applied. The following standards shall be used, as applicable, to supplement hazard identification and evaluation methodologies: NRC DG-1353, NUREG-1537, NRC Reg Guide 1.70, 10 C.F.R. 50.34, and NRC Reg Guide 1.203.

Design Basis Accident Identification: The following standards shall be used, as applicable, to identify credible design basis accidents: DOE-STD-3009-2014 (Section 3.2.1), NRC DG-1353, NUREG-1537, NRC Reg Guide 1.70, 10 C.F.R. 50.34, and NRC Reg Guide 1.203.

Accident Consequence Analysis: DOE-STD-3009-2014 (Section 3.2.2 – 3.2.4) shall be applied. The following standards shall be used, as applicable, to identify acceptable accident analysis methodologies: NUREG-1537, NRC Reg Guide 1.70, 10 C.F.R. 50.34, and NRC Reg Guide 1.203.

Hazard Control Selection: DOE-STD-3009-2014 (Section 3.3) shall be applied.

SSC Classification Hierarchy: DOE-STD-3009-2014 (Section 3.3) shall be applied. SSCs that protect the reactor core shall be designated at least safety-significant.

Probabilistic Risk Assessment: A probabilistic risk assessment shall be prepared in accordance with DOE-STD-1628-2013 if sufficient data exists. NEI Technical Report 18-04, *Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,* 2018, may be used as guidance.

Defense in Depth: DOE-STD-3009-2014 (Section 3.3.2) shall be applied. See also DOE-STD-3009-2014 Appendix A.9 and DOE G 420.1-1 for additional information.

3.6 Successor Versions of Standards

If successor versions of the DOE technical standards or industry codes and standards that are cited above have been approved for use, they may be used in the development of the DSA if their use is approved by the DOE Safety Basis Approval Authority as providing an equivalent level of safety.

4.0 SAFETY BASIS STRATEGY APPROACH

In rare cases, the approach described in Section 3 will not be appropriate for a particular reactor design. In such cases, Section 3 is not required, and Section 4 shall be used. The reactor type shall be determined, and the requirements and guidance for this reactor type in Section 3 shall be evaluated for use and applicability in the course of completing Section 4 requirements. If Section 3 is applied, Section 4 is not required, although a SBS may be developed and approved as a summary strategy to ensure clear expectations.

The Section 4 approach is a structured approach to develop a SBS that identifies the essential requirements for developing the reactor's DSA format and contents. The SBS is reviewed and approved by DOE to ensure that common expectations are established regarding the implementation of this approach. Prior to initiating development of a new or upgraded DSA, a SBS shall be prepared and submitted to the DOE Safety Basis Approval Authority for approval with the concurrence by the Chief of Nuclear Safety (CNS) or with written advice from the Chief of Defense Nuclear Safety (CDNS) as appropriate. The SBS shall describe the development approach for each of the following topics, as applicable:

- Identification and Use of Industry Codes and Standards;
- Identification and Use of DOE Technical Standards;
- Summary of Design Criteria and Design Requirements
- Summary of Process for Hazards Identification and Evaluation;
- Summary of Approach to Identify and Analyze Design Basis Accidents;
- Summary of Approach to Classification of Hazard Controls;
- Summary of Approach to Use Probabilistic Risk Assessment insights;
- Summary of Approach to Defense in Depth; and
- Format and Content of DSA.

These topics and the required contents of the SBS for each topic are described in more detail below, in Sections 4.1-4.8 of this Standard.

The form and packaging of the SBS may vary based on the needs of the project. The SBS may be a stand-alone document. The SBS may also be part of or an appendix to the Safety Design Strategy (SDS) developed to meet the required method for nuclear project design and integration that is described in DOE-STD-1189-2016, *Integration of Safety into the Design Process*.

4.1 Use of Industry Codes and Standards

4.1.1 The SBS shall identify applicable industry codes and standards, and the sections of these standards that will be used. DOE reactors shall make use of industry codes and standards to the maximum extent possible, when they are available and applicable.

4.1.2 The following industry codes and standards should be evaluated for applicability and use, depending on the type of reactor facility:

- NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,* Rev. 3, 1978;
- ANSI/ANS-1-2000, *Conduct of Critical Experiments* (R2012);
- ANSI/ANS-14.1-2004, *Operation of Fast Pulse Reactors* (R2014);
- ANSI/ANS-15.1-2007, *The Development of Technical Specifications for Research Reactors* (R2018);
- ANSI/ANS-15.4-2016, *Selection and Training of Personnel for Research Reactors;*
- ANSI/ANS-15.8-1995, *Quality Assurance Program Requirements for Research Reactors* (R2013);
- ANSI/ANS-15.11-2016, *Radiation Protection at Research Reactors;*
- ANSI/ANS-15.15-1978, *Criteria for the Reactor Safety Systems of Research Reactors* (withdrawn);
- ANSI/ANS-15.16-2015, *Emergency Planning for Research Reactors;*
- ANSI/ANS-15.17-1981, *Fire Protection Program Criteria for Research Reactors* (R1987);
- ANSI/ANS-15.21-2012, *Format and Content for Safety Analysis Reports for Research Reactors* (R2018);
- ANSI/ANS-53.1-2011, *Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants*, 2011 (R2016);
- NRC NUREG-1537*, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, 1996;
- NRC Regulatory Guide 1.201, Rev. 1, *Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power plants according to their Safety Significance,* May 2006 (R2015);
- NRC Regulatory Guide 1.203, *Transient and Accident Analysis Methods,* December 2005;

- NRC Regulatory Guide 1.232, Rev 0, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, April 2018; and
- IAEA SSG-20, *Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report for Protecting People and the Environment*, 2012.

4.2 Use of DOE Technical Standards

4.2.1 The SBS shall identify applicable DOE technical standards, and the sections of these standards that will be used. These technical standards may be modified or added to as the design develops with a revised, approved SBS. Some DOE-specific safety analysis topics are not addressed fully in industry codes and standards. These topics may be in the areas of worker safety, co-located worker safety, chemical safety, specific administrative controls, and safety management programs.

4.2.2 The following DOE technical standards should be considered for use, depending on the type of reactor facility:

- DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis*, November 2014;
- DOE-STD-1186-2016, *Specific Administrative Controls*, December 2016;
- DOE-STD-1189-2016, *Integration of Safety into the Design Process*, December 2016;
- DOE-STD-1020-2016, *Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities*, December 2016;
- DOE-STD-1066-2016, *Fire Protection*, December 2016; and
- DOE-STD-1628-2013, *Development of Probabilistic Risk Assessments for Nuclear Safety Applications*, November 2013.

4.3 Summary of Design Criteria and Design Requirements

Identification of appropriate design criteria and requirements is a critical element in the safety of nuclear reactors. By its design, certain systems and design decisions affect how the reactor will behave and respond to potential hazard and accident conditions. Title 10 C.F.R. Part 830 requires that the nuclear design criteria in DOE O 420.1, *Facility Safety,* be used for new nuclear facilities or major modifications, otherwise DOE approval is required. DOE O 420.1C design requirements include use of DOE-STD-1189-2016 and development of a SDS which documents the approach to identifying and applying design criteria. DOE O 420.1C, Change 3, also requires identification of reactor-specific design criteria that may be applicable in addition to the general nuclear design criteria. The SBS shall provide a summary of the design criteria and requirements that will be applied, consistent with the SDS. The reactor design shall address the following topics to the extent relevant.

4.3.1 Acceptable Fuel Design Limits.

The DSA shall identify acceptable fuel design limits. These fuel design limits mark the point at which reactor fuel and/or cladding damage is expected to occur. The identification of fuel design limits provides the clear criteria by which the transient response of the reactor to initiating events may be compared to determine if fuel/cladding damage has either occurred or has been averted (by inherent reactor design features or an automatic reactor shutdown system).

4.3.2 Automatic Reactor Shutdown System

An automatic reactor shutdown system shall be employed to prevent or minimize reactor fuel and cladding damage in response to evaluated hazardous event scenarios, assuming a single credible failure of the shutdown system mechanism (e.g., most reactive absorber rod remains stuck out of the core).

4.3.3 Residual Heat Removal System

A residual heat removal system shall be employed, as needed, to ensure the continued cooling of the reactor fuel after reactor shutdown. The need for such a system depends upon the reactor's operating power and duty cycle and is determined by the cooling needed to preclude post-shutdown fuel and/or cladding damage. A technical evaluation may be provided for some lower power reactors and critical assemblies to demonstrate that a residual heat removal system is not needed to ensure continued cooling.

4.4 Summary of Process for Hazards Identification and Evaluation

4.4.1 The SBS shall provide a summary of the process that will be used for hazards identification and evaluation.

4.4.2 DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,* provides an acceptable approach for hazards identification and evaluation, which may or may not need to be augmented, depending on the reactor type and hazards. This approach is further described in DOE-HDBK-1224-2018, *Hazard and Accident Analysis Handbook.* Other approaches that provide an equivalent level of safety may be acceptable if approved by DOE in the SBS.

4.5 Summary of Approach to Identify and Analyze Design Basis Accidents

4.5.1 The SBS shall provide a summary of the approach that will be used to identify and analyze design basis accidents.

4.5.2 DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,* provides an acceptable approach for design basis accident identification and analysis, which may or may not need to be augmented, depending on the reactor type and hazards. This approach is further described in DOE-HDBK-1224-2018, *Hazard and Accident Analysis Handbook.* Other approaches that provide an equivalent level of safety may be acceptable if approved by DOE in the SBS.

4.6 Summary of Approach to Classification of Hazard Controls

4.6.1 The SBS shall provide a summary of the approach that will be used to classify hazard controls.

4.6.2 DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,* provides an acceptable approach for classification of hazard controls, which may or may not need to be augmented, depending on the reactor type and hazards. This approach is further described in DOE-HDBK-1224-2018, *Hazard and Accident Analysis Handbook.* Other approaches that provide an equivalent level of safety may be acceptable if approved by DOE in the SBS.

4.7 Summary of Approach to Use Probabilistic Risk Assessment Insights

4.7.1 The SBS shall provide a summary of the approach that will be used to identify and apply probabilistic risk assessment insights, if used.

4.7.2 DOE-STD-1628-2013, *Development of Probabilistic Risk Assessments for Nuclear Safety Applications*, provides an acceptable approach for using probabilistic risk assessments. Other approaches that provide an equivalent level of safety may be acceptable if approved by DOE in the SBS.

4.7.3 NEI Technical Report 18-04, *Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,* 2018, may be used as guidance.

4.8 Summary of Approach to Defense in Depth

4.8.1 The SBS shall provide a summary of the approach that will be used to provide defense-in-depth and meet the defense-in-depth requirements of DOE O 420.1C.

4.8.2 DOE G 420.1-1A*, Nonreactor Nuclear Safety Design Criteria for Use with DOE O 420.1C, Facility Safety*, provides an acceptable approach to defense-in-depth, which may or may not need to be augmented, depending on the reactor type and hazards.

4.9 Format and Content of the DSA

4.9.1 The SBS shall provide the DSA Format and Content that will be used and how it satisfies 10 C.F.R. 830 Section 830.204(b) DSA contents requirements.

4.9.2 NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,* provides an acceptable format and content for power reactors, which may need to be augmented to address DOE-specific safety analysis topics. DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis,* provides an acceptable DSA format and content for small reactors (less than Category A reactors) and critical and sub-critical experimental facilities. A hybrid approach or another approach that provides an equivalent level of safety may also be acceptable if approved by DOE in the SBS. NUREG-1537 may be used to augment the DSA format and content.

4.10 Successor Versions of Standards

If successor versions of the DOE technical standards or industry codes and standards that are cited above have been approved for use, they may be used in the development of the DSA if they are identified and included in the SBS and approved for use by the DOE Safety Basis Approval Authority as providing an equivalent level of safety.

4.11 DOE Review and Approval

The DOE Safety Basis Approval Authority is responsible for providing review and approval of the Safety Basis Strategy, with concurrence by the CNS or with written advice from the CDNS as appropriate. DOE-STD-1104-2016 provides additional guidance and requirements for DOE review and approval of safety basis documents and safety design basis documents.

5.0 DEVELOPMENT OF DSA

5.1 The DSA shall be developed in accordance with the requirements described in Section 3 or Section 4. If using Section 4, decisions to make significant departures from the approved SBS would require development or revision of a SBS along with DOE approval.

5.2 For new reactors and planned changes to existing reactors, the mitigated off-site dose consequence for credible design basis accidents shall not exceed the Evaluation Guideline of 25 rem[4]. Note: DOE O 420.1C, Change 3, Section 4.h requires DSAs to include results of the analysis prescribed in Section 3.3.1 of DOE-STD-3009-2014 pertaining to "Existing Facilities with Mitigated Offsite Consequence Estimates over the [Evaluation Guideline]" whenever the safety analysis concludes that the mitigated off-site dose consequences for one or more accident scenarios exceed 25 rem. This could potentially occur due to the discovery of a Potential Inadequacy of the Documented Safety Analysis if it were not subsequently mitigated.

5.3 10 C.F.R. Section 830.206 provides requirements for a Preliminary DSA for new DOE reactors and for major modifications to existing DOE reactors. DOE O 413.3B provides process-related requirements including use of DOE-STD-1189-2016*, Integration of Safety into the Design Process,* which describes the relationship between the Preliminary DSA and the DSA.

[4] See DOE-STD-3009-2014 for more discussion of the Evaluation Guideline.

6.0 REFERENCES

DOE REGULATIONS

10 C.F.R. Part 830, *Nuclear Safety Management*

DOE DIRECTIVES

DOE O 413.3B, *Program and Project Management for the Acquisition of Capital Assets,* Change 5, August 12, 2018

DOE G 420.1-1A*, Nonreactor Nuclear Safety Design Criteria for Use with DOE O 420.1C, Facility Safety*, December 2012

DOE G 421.1-2A, *Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 C.F.R. 830*, December 2011

DOE O 420.1C, *Facility Safety,* Change 3, November 2019

DOE TECHNICAL STANDARDS AND HANDBOOKS

DOE-STD-1027-92, Change Notice No. 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23*, *Nuclear Safety Analysis Reports*, September 1997

DOE-STD-1083-2009 (R2015), *Processing Exemptions to Nuclear Safety Rules and Approval of Alternative Methods for Documented Safety Analyses*, June 2009

DOE-STD-1186-2016, *Specific Administrative Controls*, December 2016

DOE-STD-1189-2016, *Integration of Safety into the Design Process*, December 2016

DOE-STD-1628-2013, *Development of Probabilistic Risk Assessments for Nuclear Safety Applications*, November 2013

DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis*, November 2014

DOE-HDBK-1224-2018, *Hazard and Accident Analysis Handbook*, July 2018

INDUSTRY CODES AND STANDARDS

ANSI/ANS-1-2000, *Conduct of Critical Experiments* (R2012)

ANSI/ANS-14.1-2004, *Operation of Fast Pulse Reactors* (R2014)

ANSI/ANS-15.1-2007, *The Development of Technical Specifications for Research Reactors* (R2018)

ANSI/ANS-15.4-2016, *Selection and Training of Personnel for Research Reactors*

ANSI/ANS-15.15-1978, *Criteria for the Reactor Safety Systems of Research Reactors* (withdrawn)

ANSI/ANS-15.21-2012, *Format and Content for Safety Analysis Reports for Research Reactors* (R2018)

ANSI/ANS-53.1-2011, *Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants*, 2011 (R2016)

NRC Regulatory Guide 1.201, Rev. 1, *Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power plants according to their Safety Significance,* May 2006 (R2015)

NRC Regulatory Guide 1.203, *Transient and Accident Analysis Methods,* December 2005

NRC Regulatory Guide 1.232, Rev 0, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, April 2018

NRC Draft Regulatory Guide DG-1353, *Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors*, April 2019

NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,* Rev. 3, 1978

NRC NUREG-1537*, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, 1996

IAEA SSG-20, *Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report for Protecting People and the Environment*, 2012

NEI Technical Report 18-04, *Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,* 2018

APPENDIX A: DOE REACTORS

The following list identifies current DOE operating reactors. Each of these DOE reactors currently use NRC Regulatory Guide 1.70 as their safe-harbor DSA standards, augmented to some degree by other DOE and industry standards. Each of these reactors may, but are not required to, update their existing DSAs to meet this Standard.

Advanced Test Reactor (ATR). The ATR is a nuclear research reactor located at the Idaho National Laboratory. Constructed in 1967, the reactor was designed and is used to test nuclear fuels and materials to be used in power plants, naval propulsion, research, and advanced reactors. The ATR also produces rare isotopes for use in medicine and industry. It can operate at a maximum power level of 250 MW thermal power and has a "Four Leaf Clover" core design that allows for a variety of testing locations. The ATR is a pressurized light water reactor, using water as both coolant and moderator. The core is surrounded by a beryllium neutron reflector to concentrate neutrons on experiments and houses multiple experiment positions as well. ATR is a Category A reactor and a Hazard Category 1 DOE nuclear facility.

High Flux Isotope Reactor (HFIR). The HFIR is a nuclear research reactor located at the Oak Ridge National Laboratory. Constructed in 1965, the reactor produces a high flux of neutrons that are used to study physics, chemistry, materials science, engineering, and biology. The HFIR also produces rare isotopes for use in medicine and industry. It can operate at a maximum power level of 85 MW thermal power. HFIR is a beryllium-reflected, light-water-cooled and moderated, flux-trap type reactor that uses highly enriched uranium-235 as the fuel. The preliminary conceptual design of the reactor was based on the "flux trap" principle, in which the reactor core consists of an annular region of fuel surrounding an unfueled moderating region or "island." Such a configuration permits fast neutrons leaking from the fuel to be moderated in the island and thus produces a region of very high thermal-neutron flux at the center of the island. HFIR is a Category A reactor and a Hazard Category 1 DOE nuclear facility.

Annular Core Research Reactor (ACRR). The ACRR is a nuclear research reactor located at the Sandia National Laboratories. The ACRR, which was re-designed and re-named from its predecessor reactor and became operational in 1978, subjects electronics to high-intensity neutron irradiation and conducts reactor safety research. The primary mission of the ACRR is to subject components or systems, known as experiments, to pulse and steady-state irradiation environments in either a central experiment cavity in the center of the ACRR, the Fuel-Ringed External Cavity II (FREC-II) experiment cavity, or the Neutron Radiography experiment cavity in the ACRR pool. Sandia's ACRR is a water-moderated, pool-type research reactor capable of steady-state, pulsed and tailored transient operations and, in the past, has been configured for medical isotope production. Other duties for ACRR include: reactor-driven laser experiments; space reactor fuels development; pulse reactor kinetics; reactor heat transfer and fluid flow; electronic component hardening; and explosive component testing. At peak power in its steady state mode, the ACRR produces about 2.5 MW thermal power, but it has been operated at up to 4 MW in the past. But during a maximum pulse, it generates up to 35,000 megawatts of power for seven milliseconds. The reactor underwent extensive upgrades in 2002 and again in 2019, including upgrades to reactivity control circuitry. ACRR is a small Category B reactor (rated at less than 4 MWth) and a Hazard Category 2 DOE nuclear facility. As of the date of publication of this Standard, ACRR follows an Alternate Methodology approved by National Nuclear Security Administration with Central Technical Authority concurrence on March 26, 2020.

Transient Reactor Test Facility (TREAT). The TREAT is a nuclear research reactor located at the Idaho National Laboratory. The TREAT is an air-cooled, graphite moderated, thermal spectrum test nuclear reactor designed to test reactor fuels and structural materials. Constructed in 1958, and operated from 1959 until 1994, TREAT was built to conduct transient reactor tests where the test material is subjected to neutron pulses that can simulate conditions ranging from mild transients to reactor accidents. The TREAT was restarted in 2017 to test new accident-tolerant fuel for nuclear reactors. TREAT is capable of a wide range of operations and test conditions. TREAT can operate at a steady state power of 100 kW, produce short transients of up to 19 GW, or produce shaped transients controlled by the TREAT automatic reactor control system and the Control Rods. Test assemblies can be inserted into the reactor core and can simulate the conditions of a light water reactor, heavy water reactor, liquid metal fast breeder reactor, or a gas cooled reactor. TREAT is a small Category B reactor (rated at less than 4 MWth) and a Hazard Category 2 DOE nuclear facility.

Neutron Radiography Reactor (NRAD). The NRAD is a nuclear research reactor located at the Idaho National Laboratory. The NRAD primarily supports neutron radiography of highly irradiated materials with beam tubes that are coupled to the main hot cell of the Hot Fuel Examination Facility. Additionally, NRAD also supports limited in-core irradiations of samples. NRAD is a 300 kW Training, Research, Isotopes, General Atomics (TRIGA) pool-type reactor, installed in 1977. NRAD is a small Category B reactor (rated at less than 4 MWth) and a Hazard Category 2 DOE nuclear facility.

Advanced Test Reactor Critical Facility (ATR-C). The ATR-C is a low power pool-type reactor that operates at 5kW or less. The ATR-C was constructed in 1964 and is located at the Idaho National Laboratory. The ATR-C provides an identical core mockup as the ATR, however it operates at lower power and pressures. The primary purpose of ATR-C is reactivity measurements of experiments, benchmarking of new models and computer codes, and a test bed for new ATR applications. ATR-C is a small Category B reactor (rated at less than 4 MWth) and a Hazard Category 2 DOE nuclear facility.

Sandia Pulsed Reactor Facility Critical Experiments (SPRF/CX). The SPRF/CX is a critical experiment facility located at Sandia National Laboratories. The SPRF/CX provides a test bed for critical assemblies which provide benchmark data for reactor physics design methods and computer codes. SPR/CX was originally constructed in 1960 as a pulsed reactor and converted to its current critical experiment function in the early 2000's.It has been used since 2006 for conducting the Seven Percent Critical Experiments containing 7% enriched UO2 fuel and the Burn-Up Credit Critical Experiment (BUCCX) containing 4.3% enriched UO2 fuel. The SPRF/CX is also used for hands-on training as part of the DOE Nuclear Criticality Safety Program. SPRF/CX is a zero power critical assembly and a Hazard Category 2 DOE nuclear facility. As of the date of publication of this Standard, SPRF/CX follows an Alternate Methodology approved by National Nuclear Security Administration with Central Technical Authority concurrence on March 26, 2020.

APPENDIX B: DOE TECHNICAL STANDARDS AND INDUSTRY STANDARDS

The following list identifies and describes DOE technical standards and industry standards that are potentially relevant to preparing reactor DSAs. The selection of the exact standards to be used is a main topic of this Standard.

DOE-STD-3009-2014. DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis*, November 2014. This Standard is a safe-harbor standard used by DOE nonreactor nuclear facilities to meet the requirements of 10 C.F.R. 830, Subpart B, regarding development of a Documented Safety Analysis. This Standard provides chemical safety and worker safety requirements. DOE-STD-3009 is the most commonly used DOE safe harbor standard for DSA development. The methodology for hazards and accident analysis provided in this Standard is largely applicable to DOE reactors, however, this Standard does not provide detailed identification and analysis methods for reactor accidents.

DOE-STD-1027-92. DOE-STD-1027-92, Change Notice No. 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23*, *Nuclear Safety Analysis Reports*, September 1997. This Standard, or methods consistent with it, are required to be used by all DOE nuclear facilities per 10 C.F.R. 830, Subpart B, to determine nuclear facility hazard categorization. DOE-STD-1027-2018 is an update of this document and is consistent with its methods and approved for use.

DOE-STD-1628-2013. DOE-STD-1628-2013, *Development of Probabilistic Risk Assessments for Nuclear Safety Applications*, November 2013. This Standard describes methods for developing and applying probabilistic risk assessments for nuclear safety applications.

NRC Regulatory Guide 1.70. NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,* Rev. 3, 1978. This is the safe harbor DSA standard identified for DOE reactor Documented Safety Analysis per 10 C.F.R. 830, Subpart B, promulgated in 2001. The approach described in this Standard is for power reactors and does not fully translate for small research and test reactors. NRC Reg. Guide 1.70 does not provide chemical safety and worker safety expectations to meet 10 C.F.R. 830, Subpart B requirements.

ANSI/ANS-1-2000. ANSI/ANS-1-2000, *Conduct of Critical Experiments* (R2012). This Standard provides for the safe conduct of critical experiments. Such experiments study neutron behavior in a fission device where the energy produced is insufficient to require auxiliary cooling, and the power history is such that the inventory of long-lived fission products is insignificant.

ANSI/ANS-14.1-2004. ANSI/ANS-14.1-2004, *Operation of Fast Pulse Reactors* (R2014). This Standard is for those involved in the design, operation, and review of fast pulse reactors. It has been formulated in general terms to be applicable to all current fast pulse reactors. This Standard does not apply to periodically pulsed reactors or booster assemblies.

ANSI/ANS-15.21-2012. ANSI/ANS-15.21-2012, *Format and Content for Safety Analysis Reports for Research Reactors* (R2018). This Standard provides the criteria for the format and content for safety analysis reports for research reactors. It also provides a detailed listing of reactor accident initiating events to be considered for analysis and evaluation.

NRC NUREG-1537. NRC NUREG-1537*, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, 1996. NUREG-1537, Part 1 gives guidance to non-power reactor licensees and applicants on the format and content of applications to the NRC for licensing actions. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination. NUREG-1537, Part 2 gives guidance on the conduct of licensing action reviews to NRC staff who review non-power reactor licensing applications. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

NRC Reg. Guide 1.201. NRC Regulatory Guide 1.201, Rev. 1, *Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power plants according to their Safety Significance,* May 2006 (R2015). This Reg. Guide describes a risk-informed process for categorizing SSCs according to their safety significance.

NRC Reg. Guide 1.232. NRC Regulatory Guide 1.232, Rev 0, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, April 2018. This Reg. Guide provides designers, applicants, and licensees of non-light water nuclear reactors guidance for developing principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety.