3 PRINCIPAL DESIGN CRITERIA EVALUATION

3.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) review of the principal design criteria and bases related to structures, systems, and components (SSCs) and safety protection systems is to ensure that the principal design criteria comply with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." The result of this review will determine whether the applicant adequately defined (1) the classification of SSCs according to their importance to safety and (2) the design criteria and bases for SSCs important to safety, safety protection systems, and other SSCs. These design criteria and bases include the limiting characteristics of the spent nuclear fuel (SNF), reactor-related greater-than-Class-C (GTCC) waste, or other high-level radioactive waste (HLW) materials to be stored and external conditions during normal and off-normal operations, accident conditions, and natural phenomena events.

3.2 Applicability

This chapter applies to the review of applications for specific licenses for an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage installation (MRS), categorized as a dry storage facility (DSF). It also applies to the review of applications for a Certificate of Compliance (CoC) of a dry storage system (DSS). Sections of this chapter that apply only to a DSF specific license application are identified with "**(SL)**" in the heading. Sections or tables that apply only to DSS CoC applications have "**(CoC)**" in the heading. A subsection without an identifier applies to both types of applications.

3.3 Areas of Review

This chapter addresses the following areas of review:

- classification of SSCs
- design bases for SSCs important to safety
  - SNF specifications
  - reactor-related GTCC waste specifications **(SL)**
  - HLW specifications **(SL–MRS only)**
  - external conditions
- design criteria for safety protection systems
  - general
  - structural
  - thermal
  - shielding, confinement, and radiation protection
  - criticality
  - material selection
  - decommissioning **(SL)**
  - retrievability
- design criteria for other SSCs

3.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 10 CFR Part 72 that are relevant to the review areas addressed by this chapter. The reviewer should refer to the exact language in the regulations. Table 3-1a matches the relevant regulatory requirements to the areas of review for a specific license **(SL)** review. Table 3-1b matches the relevant regulatory requirements to the areas of review for a CoC review.

Table 3-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations			
	72.24 (a)(b)(c)(e)(f)(l)	72.40 (a)(1)(2)(3)	72.90–94	72.98
Classification of SSCs	●			
Design Bases for SSCs Important to Safety	●		●	●
Design Criteria for Safety Protection Systems	●	●		
Design Criteria for Other SSCs	●			

Areas of Review	10 CFR Part 72 Regulations (cont.)					
	72.102–103	72.104	72.106	72.120	72.122–126	72.128–130
Classification of SSCs						
Design Bases for SSCs Important to Safety	●	●	●	●	●	●
Design Criteria for Safety Protection Systems		●	●	●	●	●
Design Criteria for Other SSCs				●	●	●

Table 3-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations		
	72.104[A]	72.106[A]	72.122 (a), (b)(1),(2)(i),(3) (c), (f), (h)(1)(4) (i), (l)[B]
Classification of SSCs			
Design Bases for SSCs Important to Safety	●	●	●
Design Criteria for Safety Protection Systems	●	●	●

Areas of Review	10 CFR Part 72 Regulations (cont.)			
	72.124 (a)(b)	72.126 (a)(1)(2)(3) (4)(5)(6)[B]	72.236 (a)(b)(c)(d)	72.236 (e)(f)(g) (h)(i)(l)(m)
Classification of SSCs			●	
Design Bases for SSCs Important to Safety			●	●
Design Criteria for Safety Protection Systems	●	●	●	●

A This requirement applies to CoCs and CoC applications through the requirement in 10 CFR 72.236(d).
B Note that while 10 CFR 72.122, “Overall requirements,” and 126, “Criteria for radiological protection,” are not applicable to an application for a CoC, the CoC applicant should describe how the DSS design facilitates the ability of the licensee to meet these requirements.

The reviewer should verify that the applicant has provided sufficient general or summary discussions of the SSC design features for both operational (including normal operation conditions and anticipated occurrences (that is, off-normal conditions)) and accident conditions, including natural phenomena. This demonstrates a clear and defensible case that the applicants have met the design criteria. For specific license applications, refer to Chapter 2, “Site Characteristics Evaluation,” of this Standard Review Plan (SRP) for the specific methods and guidance reviewers should use to identify site characteristics to ensure the DSF design criteria are adequate for the DSF to be built and operated at that site and will meet the 10 CFR Part 72 requirements. For CoC applications, the safety analysis report (SAR) defines a bounding envelop of conditions for normal, off-normal, and accident conditions for which the DSS is designed to fulfill its design functions. A general licensee wishing to use the DSS at its site will need to show, in a 10 CFR 72.212, “Conditions of general license Issued under § 72.210,” evaluation report, that its site is bounded by the conditions for which the DSS was analyzed. In evaluating the principal design criteria and bases related to SSCs and safety protection systems, reviewers should seek to ensure that the DSS or DSF design fulfills the design bases and design criteria described below.

3.4.1 Classification of Structures, Systems, and Components

The applicant must identify all SSCs important to safety and provide a rationale for the identification. Acceptance criteria for classification of SSCs important to safety are based on 10 CFR 72.24, "Contents of application: Technical information," for a specific license review and 10 CFR 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," for a CoC review.

The structural, thermal, shielding, confinement, radiation protection, criticality, materials, and decommissioning evaluation chapters of this SRP discuss areas of review that also include SSCs important to safety that are identified as safety protection systems. The following sections discuss design bases for SSCs important to safety and safety protection systems.

3.4.2 Design Bases for Structures, Systems, and Components Important to Safety

3.4.2.1 *Spent Nuclear Fuel Specifications*

The applicant should provide information on the SNF to be stored including a complete list of SNF parameters and characteristics. This information includes, but is not limited to, the reactor type (e.g., boiling-water reactor (BWR), pressurized-water reactor (PWR)); fuel manufacturer and model designation and number; fuel physical characteristics; fuel cladding material; thermal and radiological characteristics; and history and census, including burnup, initial enrichment, and cooling time and, for specific licenses, the total amount of SNF to be stored at the DSF. The applicant should also identify if components associated with or integral to fuel assemblies (e.g., rod cluster control assemblies, thimble plug assemblies) would be stored and provide adequate information to characterize these components. These components are also referred to as nonfuel hardware. This information includes component types, quantities, material specifications, and any other properties, including operational specifications (e.g., 10-percent insertion into the reactor core, number of cycles or duration of use in the reactor), that are important to evaluate the components' effects on or contribution to criticality safety, heat generation, radiological source terms, and structural and confinement performance of the DSS or DSF SSCs and SNF. The applicant must also provide information on the ranges of parameters of the SNF to be stored.

The application should specify the range and types of SNF that the DSS or DSF is designed to store. These specifications should also include, but are not limited to, the following:

- type of SNF (i.e., BWR, PWR, or both)
- cladding material
- maximum assembly uranium mass loading
- bounding composition specifications for mixed-oxide SNF and SNF with thoria (includes masses of uranium, plutonium, thorium; initial enrichments of uranium and plutonium isotopes)
- assembly weights
- dimensions and configurations of the fuel

- identification and limits on amount and position of damaged fuel, if damaged fuel is to be stored, and the dimensions of the “damaged-fuel can”

- maximum allowable enrichment of the fuel before any irradiation for criticality safety and minimum enrichment for the shielding evaluation

- assigned burnup loading value (i.e., in megawatt days per metric ton of uranium or per metric ton heavy metal)

- loading curves for each set of licensing conditions if burnup credit is used (required minimum burnup versus enrichment curve)

- operational history parameters (e.g., in-core soluble boron concentration, moderator temperature) if burnup credit is used

- minimum acceptable cooling time of the SNF before storage in the DSS or DSF

- maximum heat to be dissipated

- maximum number of SNF elements

- condition of the SNF (i.e., intact assembly, damaged fuel, consolidated fuel rods)

- inerting atmosphere requirements and the maximum amount of fuel permitted for storage in the DSS or DSF

For DSSs or DSFs that will be used to store components that are associated with or integral to fuel assemblies (e.g., control rods and BWR fuel channels), the reviewer should ensure that the applicant specifies, along with the already noted parameters, the types and amounts of radionuclides, heat generation, and the relevant source strengths and radiation energy spectra permitted for storage in the DSS or DSF. For these components, the SAR should also specify and evaluate the following:

- the design-basis radiation source term

- the effects of gas generation on the cask internal pressure

- the effects of the additional weight and length of the proposed material on structural and stability analyses

- the impact of the added heat from these components, including the impact on heat transfer characteristics

- credit for any negative reactivity from residual neutron-absorbing material remaining in the control components

3.4.2.2 *Reactor-Related Greater that Class C Waste Specifications* ***(SL)***

Only solid reactor-related GTCC waste may be stored under 10 CFR Part 72, provision for which is made only for specific-license DSFs. Licensees under 10 CFR Part 50 “Domestic Licensing of Production and Utilization Facilities,” are already authorized to possess and store reactor-related

GTCC waste under provisions of 10 CFR Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"; therefore, general licensees store reactor-related GTCC waste under their 10 CFR Part 50 license and not as part of their 10 CFR Part 72 general license. Solid reactor-related GTCC waste is typically activated metals, such as reactor vessel internals, and in-core instrumentation.

There are two general categories of reactor-related GTCC waste:

- Activated metals—These wastes are not integral components of a fuel assembly and include control rod blades, local power range monitor strings, intermediate-range monitor strings, short-range monitor dry tubes, in-core instrument strings, top fuel guide, BWR core shroud, upper core support plate, PWR core shroud (baffle), lower core barrel, lower core support plate, and primary and secondary neutron sources that are not contained within the fuel assembly.

- Process wastes—These are wastes generated from the operation and decommissioning of reactors. These wastes are generated from mechanical filtration operations and can consist of paper, metals and plastics. Process wastes include control rod drive strainers, fuel pool and vacuum filters, PWR miscellaneous cartridge filters, crud tank filters, and ion exchange resins.

For reactor-related GTCC waste, the application should include the following information: waste form (e.g., activated metal, process waste), the maximum quantity of waste to be stored at the ISFSI or MRS, the radionuclide inventory, and the location and configuration of reactor-related GTCC waste containers with respect to the SNF storage casks. Applicants may choose to store reactor-related GTCC waste in containers designed to store SNF or in containers specifically designed to store GTCC waste. In either case, the application should describe the characteristics of the reactor-related GTCC waste containers necessary to demonstrate DSF compliance with the regulations when storing GTCC waste. Liquid GTCC waste may not be stored under 10 CFR Part 72.

3.4.2.3 *High-Level Radioactive Waste Specifications* ***(SL–MRS only)***

The regulations in 10 CFR 72.3, "Definitions," define HLW. The regulations in 10 CFR 72.2(a)(2) identify that only HLW in solid form is acceptable for storage and may only be stored under 10 CFR Part 72 at a DOE-owned MRS. Further conditions regarding the form of this waste are discussed in 10 CFR 72.120(c). Liquid HLW is not acceptable for storage. The applicant should provide information on the waste form, proposed storage package, characteristics of any encapsulation material, radionuclide characteristics, heat generation rate, and history. The SAR should include bounding ranges of parameters of the material to be stored. This information includes quantities, material specifications, and any other properties that are important to evaluate the criticality safety, heat generation, radiological source terms, and structural and confinement performance of the DSF SSCs associated with storage of HLW.

3.4.2.4 *External Conditions*

The SAR should define the bounding conditions under which the DSS or DSF is expected to operate and perform its design functions. The principal design bases should include the following items:

- normal design conditions, including external conditions such as ambient temperature, humidity, and insolation; operational parameters such as maximum load capacity of cranes and handling equipment; and maximum dimensions of the casks or other critical equipment to be handled

- off-normal design conditions, including external conditions such as ambient temperatures and insolation, and operational parameters that do not approach accident conditions

- accident conditions, including external conditions such as tornado wind velocities, tornado missiles, tornado pressure drop, maximum wind velocities, design-basis earthquake, peak explosive overpressure, peak flood elevation, and hypothetical accidents including storage container drop and tipover.

For specific license applications, the SAR only needs to address those conditions that are credible for, applicable to, or both, the DSF site. For CoC applications, the SARs should define the enveloping conditions for normal, off-normal, and accident (including natural phenomena) conditions for which the DSS is designed. The DSS SAR analyses should show that the DSS performs its design functions for these conditions. A general licensee wishing to use the DSS will need to show in a 10 CFR 72.212 evaluation report (which is subject to NRC inspection) that its site is bounded by the conditions evaluated in the DSS SAR.

For unique designs where operations may involve multiple configurations, including temporary configurations, for normal, off-normal and accident conditions, the SAR should include analyses of these conditions for the different possible configurations. For example, in cases were storage array expansion involves removal of material (or exposure of nonstructural material) relied on for shielding, the SAR should include analyses of normal, off-normal, and accident conditions for configurations where shielding material is removed as well as configurations where the shielding material is in place. Unique aspects of storage operations or site characteristics may necessitate evaluation of normal, off-normal, and accident conditions that are not usually considered in most DSS or DSF applications.

3.4.3 Design Criteria for Safety Protection Systems

3.4.3.1 *General*

The maximum certificate term for a DSS is not to exceed 40 years (see 10 CFR 72.230(b)). The maximum license term for a DSF is 40 years from the date of issuance (see 10 CFR 72.42(a)). The applicant should demonstrate that the design will last for the proposed effective certificate or license term, as applicable. The reviewer should verify that the applicant has provided a brief description of the proposed quality assurance program and of applicable industry codes and standards that will be applied to the design, fabrication, construction, and operation of the DSS or DSF. The applicant should also describe how the design considers compatibility with removal from a reactor site, transportation, and ultimate disposition of the stored SNF.

In establishing normal and off-normal conditions applicable to the design criteria for DSS or DSF designs, applicants should account for actual facility operating conditions and configurations. Therefore, design considerations should reflect normal operational ranges, including any seasonal variations or effects and any temporary configuration changes that may occur as part of normal operations.

An aspect of the DSF design criteria and design basis is fire protection. Regulatory Guide (RG) 1.189, “Fire Protection for Nuclear Power Plants,” RG 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” and RG 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,” provide guidance related to fire protection. Chapter 11, “Operation Procedures and Systems Evaluation,” of this SRP provides details on the fire protection review of the proposed DSS or DSF design.

3.4.3.2 *Structural*

The SAR should define how the DSS or DSF structural components are designed to accommodate combined normal, off-normal, and accident loads while preserving recoverability and protecting the DSS or DSF contents, including site-generated wastes for DSFs, from significant structural degradation, criticality, loss of shielding, and loss of confinement. This discussion is generally a summary of the analytical techniques and calculation results from the detailed analysis given in the SAR chapter addressing the structural evaluation, and it should be presented in a nonproprietary form. Chapter 4, “Structural Evaluation,” of this SRP details the acceptance criteria to be considered in the structural design of the proposed DSS or DSF.

RG 1.13, “Spent Fuel Storage Facility Design Basis,” provides general design guidance for SNF storage facilities and specific design guidance for pools at those facilities. RG 1.13 refers to American National Standards Institute/American Nuclear Society (ANSI)/(ANS) standard ANSI N210-1976/ANS-57.2-1983, “Design Objectives for Light Water Reactor Spent Fuel Pool Storage Facilities at Nuclear Power Stations.” RG 1.13 specifically provides guidance for licensees under 10 CFR Part 50, but can be used for those licensed under 10 CFR Part 72. Additional guidance includes the following:

- design bases guidance for tornado protection in RGs 1.76, “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,” and 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants”

- guidance for flood protection in RG 1.59, “Design Basis Floods for Nuclear Power Plants,” and 1.102, “Flood Protection for Nuclear Power Plants,” guidance for protection against seismic events in RGs 1.29, “Seismic Design Classification,” 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” 1.92, “Combing Modal Responses and Spatial Components in Seismic Response Analysis,” 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” and 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion”

In addition, consider the guidance in ANSI/ANS 57.9, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

3.4.3.3 *Thermal*

The SAR should contain a general discussion of the proposed heat-removal systems, including the reliability and testing of such systems, and any associated limitations. All heat-removal systems should be passive and independent of intervening actions under normal and off-normal conditions. Chapter 5, “Thermal Evaluation,” of this SRP details the acceptance criteria to be considered in the thermal design of the proposed DSS or DSF.

3.4.3.4 *Shielding, Confinement, Radiation Protection*

The applicant should describe those features of the storage facility that protect occupational workers and members of the public against direct radiation doses and releases of radioactive material and minimize the dose from normal operations and from any off-normal or accident conditions.

The applicant should also identify the design criteria and design bases for the storage facility’s shielding, confinement, and radiation protection design, including discussion of any appropriate regulatory guides used for those criteria and bases.

Chapters 6, 9, 10A and 10B, and 13 (“Shielding Evaluation,” “Confinement Evaluation,” “Radiation Protection Evaluation,” “Waste Management Evaluation,” respectively) of this SRP detail the acceptance criteria to be considered in the shielding, confinement, radiation protection, and waste management design, respectively, of the proposed DSS or DSF.

3.4.3.5 *Criticality*

The SAR should address the mechanisms and design features that enable the storage facility to maintain SNF, and, as applicable for a specific license DSF, the reactor-related GTCC waste and HLW in a subcritical condition under normal, off-normal, and accident conditions. Chapter 7, “Criticality Evaluation,” of this SRP details the acceptance criteria to be considered in the criticality design of the proposed DSS or DSF.

3.4.3.6 *Material Selection*

The materials selected for the DSS or DSF must demonstrate adequate corrosion performance during normal operation, off-normal, and accident conditions in the environmental conditions of the storage facility for the duration of the license for DSFs and the environmental conditions to which the DSS may be exposed (or for which it was intended to be designed) for the duration of the certified period of storage.

The SNF cladding must be protected during storage against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational problems with respect to its removal from storage.

Chapter 8, “Materials Evaluation,” of this SRP details the acceptance criteria to be considered in the materials selection design of the proposed DSS or DSF.

3.4.3.7 *Decommissioning* ***(SL)***

The NRC outlines the regulatory requirements for decommissioning considerations for specific licenses in 10 CFR 72.130, “Criteria for decommissioning.”

DSF SSCs should be designed for ease of decontamination and eventual decommissioning. The SAR should describe the features of the design that support these two activities.

Chapter 14, “Decommissioning Evaluation,” of this SRP details the acceptance criteria to be considered in the review of decommissioning proposed for the DSF design.

3.4.3.8 *Retrievability*

The regulation in 10 CFR 72.122(l) states that "storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal." The NRC interprets this regulation to require that a storage system be designed to allow for ready retrieval in the initial design, amendments to the design, and in license renewal, through the aging management of the design. Retrievability is applicable only during normal and off-normal conditions; it does not apply to accident conditions. The retrievability requirement applies to all general licensed and specific licensed ISFSIs. The requirements in 10 CFR 72.236(m) state that CoC holders should design for retrievability "[t]o the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy."

Acceptable means for removing the spent fuel from storage include the ability to perform any of the three options below. These options may be utilized individually or in any combination or sequence, as appropriate.

- Remove individual or canned SNF assemblies from wet or dry storage.
- Remove a canister loaded with SNF assemblies from a storage cask or overpack.
- Remove a cask loaded with SNF assemblies from the storage location.

Applicants for an initial ISFSI license or an ISFSI license amendment must meet the retrievability requirement in 10 CFR 72.122(l). In order to do so, the storage system design should allow for ready retrieval by the use of option A, B, or C or a combination of the options. A dry-storage system may demonstrate retrievability by the use of a known and controlled fuel selection, limits on the loading temperature, a known atmospheric environment, and transfer cask or canister temperature control. The reviewer should also verify that applications for all storage systems identify the SSCs important to safety and the SSC subcomponents that are relied upon for ready retrieval. The reviewer should further verify that the technical specifications included in the application provide for the maintenance of SSCs relied upon for ready retrieval.

When an applicant for an initial ISFSI license or license amendment relies on Option A to demonstrate ready retrieval, the reviewer should confirm that the applicant demonstrates the fuel assemblies will not exhibit gross degradation, and will be removable. Additional review will be needed in the case where there is an assembly with gross degradation or an assembly contains rods with breaches greater than a pinhole leak or a hairline crack (i.e., gross ruptures that could lead to release of fuel particulates). The reviewer should confirm that the applicant demonstrates that the fuel assembly can be placed inside a secondary container. The secondary container must confine the fuel particulate to a known volume and be capable of removal.

When an applicant for an initial dry storage ISFSI license or license amendment relies upon Option A to demonstrate ready retrieval, it is likely the storage cask or canister will, at some point, need to be moved from the storage location to a location where the SNF assemblies can be removed from the cask or canister. When the reviewer anticipates that the cask or canister will have to be moved, the reviewer should confirm the applicant relying upon Option A to demonstrate ready retrieval also demonstrates ready retrieval under Option B or Option C.

When an applicant for an initial ISFSI license or license amendment demonstrates ready retrieval with Option B or Option C, the continued ready retrieval of the storage system should be addressed in its technical specification. However, in addition to the technical specification, an applicant may also propose to implement a program to identify, monitor, and mitigate possible degradation that could impact the intended function of the dry storage system's SSCs and subcomponents of the dry storage system that are relied upon to comply with the retrievability requirements.

When the application is for renewal of an ISFSI license, verify that the 10 CFR 72.122(l) retrievability requirement is met by ensuring that the approved design bases for the item being relied upon in the option(s) chosen (e.g., fuel assembly, cask, or canister) to demonstrate ready retrieval, including any programs implemented, has not been altered. Additionally, the reviewer should verify that aging management programs and time-limited aging analysis associated with renewed licenses provide reasonable assurance that the approved design bases will be maintained during the period of extended operation. This will include reviewing operating experience, including inspections and analyses performed during the initial storage period for ensuring SSCs relied upon for ready retrieval were maintained. The reviewer should refer to NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," issued June 2016, Agencywide Documents Access and Management System (ADAMS) Accession No. ML16179A148) for additional guidance.

CoC holders and applicants for a CoC are not required by regulation to demonstrate retrievability under 10 CFR 72.122(l); however, 10 CFR 72.236(m), which applies to CoC holders, states that retrievability should be considered to the extent practicable in the design to consider removal of the SNF from storage, transportation, and ultimate disposition. When a CoC applicant for an initial certificate, amendment, or revision chooses to incorporate retrievability aspects, the reviewer should confirm that the retrievability aspects are technically justified and verify that 10 CFR Part 72 requirements affected by retrievability are evaluated and met. This may include the NRC reviewer confirming that the design for the dry storage system includes an evaluation for potential degradation mechanisms for both the storage cask or canister and the SNF to assure that the design of the system has considered removal of the SNF from storage during the storage term. Note that the general licensee must comply with the retrievability requirement in 10 CFR 72.122(l) and should demonstrate that the canister or casks meet the amendment loading requirements.

The SAR does not need to describe specific retrieval facilities, equipment, and procedures for post-accident conditions because of the wide variety of possible post-accident conditions that may occur. The design and procedures for retrieval or recovery (following design basis accident) must be such that the operations can be conducted in compliance with the requirements of 10 CFR Part 20, "Standards for Protection against Radiation."

General regulatory requirements for retrieval capability are given in 10 CFR 72.122(a), (b)(1), (b)(2), (b)(3), (c)(f)(h). Retrievability is specifically outlined in 10 CFR 72.122(l). The applicant must include design criteria and design bases for retrieval.

3.4.4 Design Criteria for Other Structures, Systems, and Components (SL)

Design criteria and bases for other SSCs (i.e., those determined as being not important to safety) should meet the general regulatory requirements in 10 CFR 72.24(a)–(h) and (l) and the appropriate requirements in 10 CFR 72.120, "General Considerations." The applicant must identify design criteria and bases for SSCs determined not important to safety. The design criteria and bases for SSCs that are not important to safety may be adequately defined by statements in the SAR identifying the design codes and standards to be met in design and construction. More extensive definition is typically appropriate for SSCs that interface with, or that could adversely affect, SSCs important to safety. Section 4.3.2 of this SRP includes some examples of the types of SSCs which may fall into this category and are within the scope of NRC review. The application should include a description of the other SSCs which are relevant to the evaluations described in the radiation protection evaluation (see Chapter 10A), including for meeting requirements such as 10 CFR 72.24(e). The descriptions should be sufficiently detailed to support those evaluations and address the relevant regulatory requirements.

3.5 Review Procedures

Figure 3-1 shows the interrelationship between the principal design criteria evaluation and the other areas of review described in this SRP.

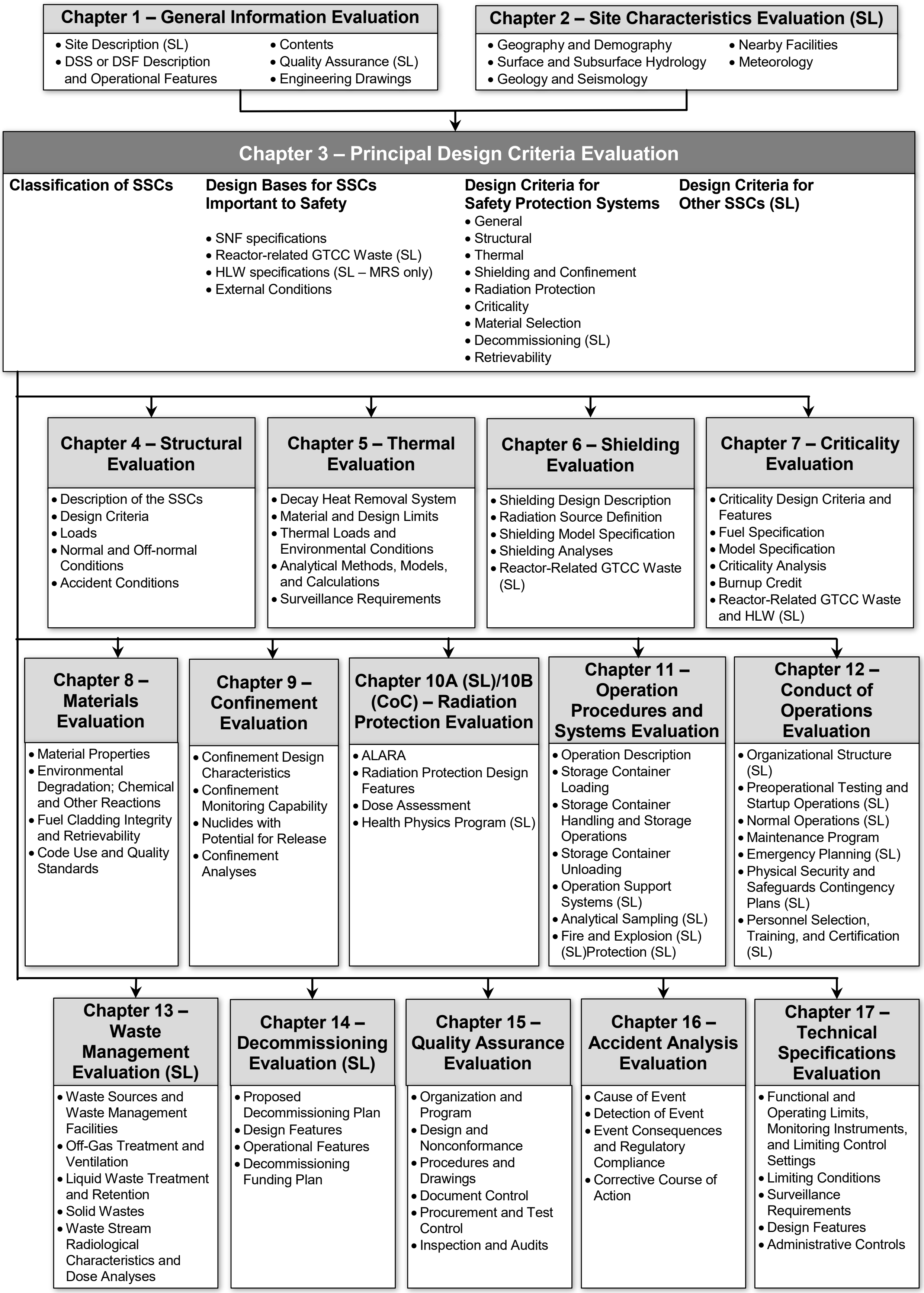

Figure 3-1 Overview of Principal Design Criteria Evaluation

Reviewers of each chapter of the SAR should consider the chapter on SSCs and principal design criteria in combination with additional details presented in their respective chapters. Evaluations of design criteria applicable to each of the relevant chapters of the SAR are discussed in detail in the respective SRP chapters. Reviewers should coordinate the review of each chapter with the applicable disciplines to ensure that multidisciplinary issues that impact more than one chapter have been addressed.

A DSF application may involve use of one or more DSSs certified under 10 CFR Part 72, Subpart L, “Approval of Spent Fuel Storage Casks,” including the SARs for the certified DSSs by reference. The application should provide additional information relating to the DSSs, including the applicant’s evaluations that establish that the site parameter limits are within the bounds of those established as limiting conditions as set forth in the referenced CoCs.

3.5.1 Classification of Structures, Systems, and Components

Although not an exhaustive list, determine if the application includes any of the SSCs and functions listed below that typically are considered important to safety. Determine if the application includes (or should include) other SSCs or functions that may be considered important to safety, based on the design of the DSF/DSS:

- components of the confinement boundary and integral components and structures used within the confinement boundary of the storage containers
- SSCs providing criticality control (e.g., SNF basket, neutron absorbers)
- radiation shielding
- SSCs providing capabilities for lifting, handling, and transferring SNF, reactor-related GTCC waste, or HLW, as applicable
- instrumentation and controls SSCs if they are used as the primary means for real-time recognition of off-normal conditions or accident conditions
- SSCs providing either active or passive decay heat removal
- the confinement systems to prevent the release of radioactive liquid wastes generated from site operations **(SL)**
- SSCs for retaining radioactive material within the pool building, if applicable **(SL)**

The radiation shielding includes any engineered features, such as berms or shield walls, that are used to ensure compliance with 10 CFR 72.104(a) or 10 CFR 72.106(b).

3.5.2 Design Bases for Structures, Systems, and Components Important to Safety

Verify that the types of materials to be stored comply with 10 CFR 72.120(b) or 10 CFR 72.120(c). Confirm that the SAR gives SNF, reactor-related GTCC waste, or HLW acceptance specifications, as applicable, including upper- or lower-bound limits, as appropriate, of acceptable variability. Verify that appropriate specifications are incorporated into the technical specifications for the DSS or DSF. For DSF applications, confirm that the SAR gives the criteria for procedures for testing, inspecting, and verifying wastes received for storage at the facility. Verify that the SAR defines criteria for procedures for handling; repackaging (if needed); and shipping out-of-specification wastes. For DSS applications, confirm that the SAR describes procedures for identifying and verifying that SNF and any nonfuel hardware to be loaded and stored in the DSS meet the specifications for allowable DSS contents.

3.5.2.1 *Spent Nuclear Fuel*

Review the detailed specifications for the SNF to be stored in the DSS as presented in the chapter of the SAR on principal design criteria, and ensure that the specifications are consistent with those discussed in the chapter of the SAR on general information and other locations. The descriptions of the SNF and components associated with the fuel assemblies (that is, nonfuel hardware) to be stored should include the information described in Section 3.4.2.1 of this SRP.

Examine any limitations regarding the condition of the SNF. If damaged fuel is allowed, the effects of such damage should be assessed in later sections of the SAR. Section 8.5.15.1 of this SRP provides specific conditions that define damaged fuel and identifies methods for classifying fuel. If damaged rods have been removed from a fuel assembly and they have or have not been replaced with solid dummy rods, the criticality reviewer should determine whether the intended loading configuration has been adequately analyzed to show subcriticality. The presence of additional moderating material will need to be addressed in the criticality analysis in the SAR. Coordinate the review with the structural reviewer if there are structural defects in the assembly hardware.

The release of fill and fission product gases from failed fuel rods increases the pressure in the cask cavity and the potential source term in the event of confinement failure. Verify that the application provides information regarding the fill or fission product gas present in the fuel as well as the free volume in the cask cavity to enable an evaluation of the pressure in the cask cavity resulting from cladding failure during storage. For the purpose of calculating internal cask pressures, the NRC staff has accepted the bounding assumptions presented in Section 5.5.4.6, Pressure Analysis," of this SRP on pressure analysis, as regards the minimum percentages of fuel rods that have failed (and released their gases).

Pay particular attention to the specification of burnup, cooling time, and decay-heat generation rate. These parameters are generally not independent, and the manner in which they are specified and combined can significantly affect the maximum allowed cladding temperature as discussed in Chapter 5 of this SRP.

The SARs typically list various fuel assemblies that can be stored in the DSS or at the DSF. It is not expected that one type of fuel assembly will bound all analyses. Ensure that the application justifies which specifications are bounding for each of the evaluations presented in subsequent sections of the SAR. Ensure that the SAR chapter on technical specifications and operational controls and limits clearly identifies or references the specifications used in the analyses.

If the applicant requests permission for the storage of components that are associated with or integral to the fuel assembly in the DSS or DSF storage container, examine the relevant detailed specifications, conditions, and constraints presented in the SAR. These specifications should be as detailed as the applicable information presented for the fuel designs to provide the reviewer with a basis for determining that the relevant safety functions of the DSS or DSF SSCs will be maintained. Ensure that the applicant also considers the storage of these components in the analyses.

If the applicant requests burnup credit, examine the relevant detailed specifications of the contents to which burnup credit is being applied. These specifications include those that are already considered in criticality analyses for fresh fuel (e.g., maximum initial enrichment). Additional specifications that must be reviewed include the cooling time, the burnup, the requested amount of credit (i.e., the specific actinides), and operational history parameters (e.g., core average boron concentration and assembly average moderator temperature).

3.5.2.2 *Reactor-Related GTCC Waste* ***(SL)***

Ensure that the reactor-related GTCC waste is appropriately characterized so that the reviewer has reasonable assurance that storage is in compliance with the regulations. For reactor-related GTCC waste, the applicant should provide the waste form (e.g., activated metal, process waste), the maximum quantity of waste to be stored at the ISFSI or MRS, the radionuclide inventory, and the location and configuration of reactor-related GTCC waste containers with respect to the SNF storage casks. Verify that the reactor-related GTCC waste form is solidified and that there are no liquids present in the container. The applicant should describe the means by which this verification will be done and justify that the means are sufficient to ensure that received materials meet the license requirements for storage at the facility.

Applicants may choose to store reactor-related GTCC waste in containers designed to store SNF or in containers specifically designed to store reactor-related GTCC waste. In either case, ensure that the application describes the characteristics of the reactor-related GTCC waste containers necessary to demonstrate DSF compliance with the regulations when storing reactor-related GTCC waste. Verify that the applicant has evaluated the impact(s) associated with the safe colocation of reactor-related GTCC waste and SNF at an ISFSI or MRS under normal, off-normal, and accident conditions.

3.5.2.3 *High-Level Radioactive Waste* ***(SL–MRS only)***

Determine that the HLW is appropriately characterized so that the necessary design and analytical calculations and acceptance tests may be carried out. For HLW, such characteristics include waste form, decay heat, inventory of radionuclides, and the characteristics described in Section 3.4.2.3 of this SRP.

Ensure that the waste form is solid and not liquid. If the waste form contains liquid, as in undried filter residues, the NRC staff must establish waste acceptance specifications and bounding limits of acceptability.

3.5.2.4 *External Conditions*

Verify that the SAR identifies those external conditions that significantly affect, or could potentially affect, the performance of the DSS or DSF. For a DSS, these design-basis conditions will generally restrict either the sites at which the DSS can be used for SNF storage or the manner in which the DSS can be handled. For example, by selecting the design earthquake, the SAR limits the use of the DSS being reviewed to sites that are bounded by this seismic limit. For a DSF, these design-basis conditions should be based on, or include conditions that are based on, the characteristics of the site at which the DSF will be built and operated. By establishing a design-basis drop, the SAR defines the maximum height to which a DSS or DSF storage container can be lifted without additional safety analysis or design changes (e.g., addition of impact limiters) by the applicant.

Note that movement of DSS or storage container components within a reactor building may not meet the NRC's criteria described in the NRC Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety Related Equipment," dated April 11, 1996, for movement of heavy loads within the reactor building. As such, if a potential DSS user (licensee) has been identified or the DSF is co-located with a 10 CFR Part 50 or 10 CFR Part 52 licensee and involves (storage container handling) operations in a building or with SSCs licensed as part of the 10 CFR Part 50 or Part 52 facility, the reviewer should coordinate with the appropriate project manager or technical lead from the NRC's Office of Nuclear Reactor Regulation (NRR) during the early stages of the review.

At a minimum, the NRC staff has generally addressed the conditions discussed below; however, other conditions may be relevant depending on specific details of the DSS or DSF design. Pay particular attention to special design features and how these might be affected by other external conditions and other components of the DSS or DSF. Ensure that the SAR provides all required information for the design earthquake accident analysis.

"Normal" conditions (including conditions involving handling and transfer) and the extreme ranges of normal conditions are presumed to exist during design-basis accidents or design-basis natural phenomena, with the exception of irrational or readily avoidable combinations. For example, an earthquake or tornado may occur at any time and in combination with any "normal" condition. By contrast, it can be presumed that transfer, loading, and unloading operations would not be conducted during a flood.

"Off-normal" conditions and events are presumed to occur in combination with normal conditions that are not mutually exclusive. Nonetheless, the SAR is not required to analyze nor must the DSS or DSF be designed for the simultaneous occurrence of independent off-normal conditions or events, design-basis accidents, or design-basis natural phenomena.

Conditions involving a "latent" equipment or instrument failure or malfunction (that is, one that occurs and remains undetected) should be presumed to exist concurrently with other off-normal or design-basis accident conditions and events. Typical latent malfunctions include a misreading instrument that is not detected as part of routine procedures, an undetected ventilation blockage, or undetected damage from an earlier design-basis off-normal or accident event or condition if no provisions exist for detection, recovery, or remediation of such conditions.

For normal, off-normal, and accident conditions, verify that the application defines appropriate operating and accident scenarios. For these scenarios, verify that the SAR includes a comprehensive evaluation of the effects of such scenarios on the SSCs important to safety. The individual chapters of this SRP address the analyses of such events. For example, Chapter 4 addresses the analyses of an earthquake on the structural components of the DSS or DSF. Verify that the applicant's evaluations demonstrate that the requirements in 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," 10 CFR 72.106, "Controlled area of an ISFSI or MRS," and 10 CFR Part 20 are or will be met for DSFs and can be met for DSSs. While the requirements in 10 CFR Part 20 do not apply to DSSs, they may be useful in informing the reviews of DSS applications.

Verify that the scenarios and evaluations address all relevant configurations of the SSCs of the DSS or DSF. For example, a storage design may rely on nonstructural materials that, for that design, may be removed, exposed, or otherwise disturbed during normal, though temporary, operations such as activities to expand the existing array of storage containers. For such designs, evaluate impacts of normal, off-normal, and accident conditions for these temporary configurations as well as the long-term design configurations. Ensure that evaluations of external conditions address any conditions or events that may be unique to the design at the different stages of DSS or DSF operations. Also, for DSFs, ensure that the evaluations address SSCs in addition to the storage containers (e.g., SSCs for waste management), as applicable, and unique site characteristics and features.

If appropriate, verify that the SAR chapter on technical specifications and operational controls and limits evaluation includes the following design bases as operating controls and limits:

3.5.2.4.1 Normal Conditions

For a given SNF specification, the primary external conditions that affect DSS or DSF performance are the ambient temperatures, insolation, and the operational environment experienced by the DSS or DSF.

Ensure that the maximum and minimum “normal” temperatures are the highest and lowest ambient temperatures recorded in each year, averaged over the years of record. For a CoC SAR, the applicant may select any design-basis temperatures as long as any operational restrictions imposed are acceptable to both the applicant and the NRC. If the storage container is also designed for transportation, the temperature requirements in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” could determine the design-basis temperatures for storage. For a specific license SAR, the NRC accepts as the maximum and minimum normal temperature the highest and lowest recorded for the hottest or coldest month of each year, averaged over the years of record.

For storage containers, the NRC staff accepts a treatment of insolation similar to that prescribed in 10 CFR 71.71, “Normal conditions of transport,” for transportation packages. If the applicant selects another design approach, it should justify the alternative approach in the SAR.

The operational environment experienced by the DSS or DSF under normal conditions includes the manner in which the DSS or DSF storage container is loaded, unloaded, and lifted. Occupational dose rates will, in part, depend on whether the DSS or DSF storage container is sealed in a wet or a dry environment. Fuel cladding temperatures may also be affected by these conditions. The manner in which the DSS or DSF storage container is lifted will determine the load on the trunnions, the lifting yoke, or both. The orientation of the DSS or DSF storage container (vertical or horizontal) and its height above ground during transport to the storage pad will establish initial conditions for the drop accidents discussed below.

NUREG-2174, “Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Casks, Final Report,” issued March 2016, provides further guidance for reviewing the thermal impact and environmental conditions (e.g., ambient temperature, wind) on a DSS or DSF storage container.

3.5.2.4.2 Off-Normal Conditions

Ensure the SAR addresses several off-normal conditions, such as variations in temperatures beyond normal, failure of 10 percent of the fuel rods combined with off-normal temperatures, partial blockage of air vents, human error, out-of-tolerance equipment performance, equipment failure, and instrumentation failure or faulty calibration. Ensure that the SAR addresses retrievability of the stored SNF, reactor-related GTCC, and HLW, as applicable for the application, for these conditions.

3.5.2.4.3 Accident Conditions

The staff has generally considered that the SAR evaluates the accidents listed in this section. These do not constitute the only accidents that should be addressed if the SAR is to serve as a reference for accidents for a specific application. Other credible accidents that may be derived from a hazard analysis could include accidents resulting from operational error, instrument failure, lightning, and other occurrences. The regulations in 10 CFR 72.122 and 10 CFR 72.236 require that the storage container be designed to withstand the effects of accident conditions and natural phenomena events without impairing its capability to perform safety functions. Consequently, in the analyses for conditions resulting from design-basis accidents and natural phenomena, the NRC has asserted and the applicant should assume a release of 100 percent of the initial rod fill gases and a release of 30 percent of the fission product gases from the fuel rods into the storage container interior. The remaining 70 percent of the fission product gases is presumed to be retained within the fuel pellet. In coordination with the confinement reviewer, verify that the storage container is designed to provide the confinement safety function under all credible conditions.

Postaccident recovery of damaged fuel may require such systems as overpacks or dry-transfer systems since ready retrieval of the fuel is required only for normal and off-normal conditions. Ensure that the SAR identifies and justifies accident situations that are not credible because of design features or other reasons. Chapter 16, “Accident Analysis Evaluation,” and the technical chapters of this SRP provide more detail regarding accidents.

Storage Container Drop

Verify that the SAR identifies the operating environment experienced by the storage container as well as the drop events (i.e., end, side, corner) that could result. Generally, the design basis is established either in terms of the maximum height to which the storage container may be lifted when handled by equipment not meeting the single-failure proof criteria or in terms of the maximum acceleration that the storage container could experience in a drop.

Cask Tipover

Although cask system supporting structures may be identified and constructed as important to safety (i.e., designed to prevent cask tipovers), ensure that the applicant analyzes cask tipover events. In some cases, cask tipover may be determined to be a credible hazard, and the associated analysis should reflect the conditions (e.g., heights and accelerations) associated with that hazard.

Fire

Ensure that the fire conditions postulated in the SAR provide an “envelope” for subsequent comparison with site-specific conditions for DSS applications. For DSF applications, ensure that the postulated fire conditions in the SAR are based on the site characteristics, including facility design and layout, that are described in the DSF application that may affect the fire conditions that are credible at the DSF. The NRC accepts the methods discussed in 10 CFR 71.73(c)(4). In addition, the NRC staff accepts that the availability of flammable material at a DSF may be limited such that the applicant may consider only materials such as those that are associated with vehicles transporting or lifting the storage containers or sources of nearby combustible materials. Regardless of which approach the applicant takes, the SAR should specify and justify the bounding conditions for a “design-basis” fire.

Explosive Overpressure

The conditions under which the SCCs for a DSS or DSF may be exposed to the effects of an explosion vary greatly among individual sites. Generally, explosive overpressure is postulated to originate from an industrial accident. Consequently, this SRP does not consider explosive overpressures from sabotage events.

For DSS applications, the extent to which the SAR addresses explosive overpressure directly affects the degree of site-specific review required of a general licensee to meet the requirements in 10 CFR 72.212. For DSF applications, the extent to which the SAR addresses these events should be commensurate with the site characteristics and facility design features of the DSF. The principal concern in the SAR should be the effects of explosive overpressure on the storage system and containers and, for DSFs, other important SSCs rather than descriptions of hypothesized causes. Though, for DSF applications, facility design and site characteristic information will enable the identification of possible sources of these events and the bases for estimates of the events' design parameters. Verify that the design parameters for blast or explosive overpressures identify pressure levels as reflected ("side-on") overpressure and provide an appropriate (assumed, for DSSs) pulse length and shape. For DSS applications, ensure this discussion provides sufficient information for general licensees to determine in their 10 CFR 72.212 evaluations if the effects of their site-specific hazards are bounded by the DSS design bases.

Air Flow Blockage

For storage designs with internal air flow passages, verify that the application considers blockage of air inlets and outlets in an accident condition. The NRC staff considers that the effects of such an assumption should be used in determining the appropriate inspection intervals or monitoring systems, or both, for the DSS or DSF storage containers.

3.5.2.4.4 Natural Phenomena Events

The NRC staff has generally considered that the SAR should evaluate the following events as design-basis accidents:

Flood

Ensure that the SAR establishes a design-basis flood condition. For a specific license application, verify that the design-basis flood condition is based on the site flood parameters. For a CoC application, this condition may be determined on the basis of the presumption that the DSS cannot tip over and the yield strength of the DSS will not be exceeded. Alternatively, the SAR can show that credible flooding conditions have negligible impact on the DSS design.

If the SAR establishes parameters for a design-basis flood, ensure that it recognizes all of the potential effects of flood water and ravine flood byproducts. Serious flood consequences can involve effects such as blockage of ventilation ports by water and silting of air passages. Other potential effects include scouring below foundations and severe temperature gradients resulting from rapid cooling from immersion.

Tornado

The NRC staff accepts design-basis tornado wind loading as defined by RG 1.76 and RG 1.117. Ensure that the application includes design criteria for the DSS or DSF on the basis of these wind-loading and missile-impact definitions. The DSS or DSF storage container should not tip over, and the capability to perform the confinement safety function should not be impaired. The NRC staff considers that tornados and tornado missiles may occur without warning.

Earthquake

Ensure that the SAR states the parameters of the design earthquake. For use of a DSS at reactor sites, this is equivalent to the safe-shutdown earthquake used for analysis of nuclear facilities under 10 CFR Part 50. An analysis for an operating-basis earthquake is not required for a DSS SAR prepared in accordance with 10 CFR Part 72, Subpart L. While the SAR analyzes tipover accidents, tipover caused by an earthquake may not be a credible event. Verify that the SSCs meet appropriate guidance in RG 1.29, RG 1.61, and RG 1.92.

Burial Under Debris

Debris resulting from natural phenomena or accidents that may affect storage container performance may be addressed in the SAR or left to the general licensee’s site-specific 10 CFR 72.212 evaluation for DSS applications. Ensure the SAR for a DSF specific license application addresses this scenario. Such debris can result from floods, wind storms, or landslides. The principal effect typically is on thermal performance.

Lightning

Lightning typically has a negligible effect on DSFs or DSSs; however, the design of the DSF or DSS structures should adhere to the requirements of National Fire Protection Association 780, “Lightning Protection Code,” and National Fire Protection Association 70, “National Electrical Code.” Ensure that the applicant cites these codes as part of the general design criteria for the DSF or DSS (see Section 3.4.3.1 of this SRP). In addition, verify that the SAR addresses lightning as a natural phenomenon if DSF or DSS performance may be impacted by the effect of lightning on an SSC.

Other

The regulations in 10 CFR Part 72 identify several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for SNF storage. The DSS SAR may include these natural phenomena as design-basis events or show that their effects are bounded by other events. If these events are not addressed in the SAR and they prove to be applicable to a specific site, a safety analysis is required before approval for use of the DSS under a general license. Ensure that the DSF SAR addresses these other natural phenomena and their effects on the DSF’s SSCs or justify why they are not applicable for the DSF site.

3.5.3 Design Bases for Safety Protection Systems

SCCs for the DSS or DSF that are to be used in facility areas subject to review under 10 CFR Part 50 should satisfy the requirements in 10 CFR Part 72 (with review guided by this SRP) and 10 CFR Part 50 (with review guided by NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LRW Edition,” issued March 2007). If the application states that the DSS or DSF will be located at a specific reactor site, then the DSS or DSF project manager should inform the appropriate NRR project manager. Note that heavy loads are likely a matter of interest to NRR.

Use Table 3-2 during the initial stages of the review for both DSS and DSF applications to ensure the SAR identifies the listed design criteria (and design bases). The table also includes or identifies, as applicable, additional information that is only relevant to a DSF specific license review. The applicability of Table 3-2 may vary depending on the details of the DSS or DSF.

Table 3-2 Outline of Design Criterial and Bases

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Design Life	Limited to the requested term in the application, not to exceed the applicable limit in either 10 CFR 72.42(a) or 10 CFR 72.230(b)	
Design Bases	SNF Specifications: (1) Assembly type(s) (2) Configuration and vendor (3) Enrichment (maximum and minimum) (4) Weight or range of weights of assemblies (5) Burnup (6) Type of cladding (7) Assemblies or cask (8) Dimensions (9) Uranium or heavy metal mass loading per assembly (10) Thoria amount or plutonium isotopic compositions for SNF with thoria and mixed-oxide (MOX) SNF, along with amount and enrichment of uranium Decay Heat Assembly: (1) Minimum decay or cooling time (e.g., 5 years, 10 years) (2) Maximum kilowatts per assembly (3) Heat-load pattern Gas Volume (at temperature) Fuel Condition or Damage Allowed Burnup Credit: (1) Credit amount (burnup and specific nuclides) (2) Operational history parameters Non-Fuel Hardware	Specifications of radioactive material to be stored (including HLW and reactor-related GTCC waste, as applicable) as described in the appropriate section of this SRP chapter. Maximum total quantities of SNF, reactor-related GTCC waste, and HLW to be stored at the facility

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Normal Design Event Conditions	Ambient Temperature: (1) Maximum (2) Minimum Loading: (1) Wet or dry Storage and Handling (e.g., loading, transfer) Orientation: (1) Vertical or Horizontal Maximum Lift Height Maximum Cladding Temperature 1% Fuel Rod Rupture Solar Insolation Other Relevant Operational Environment Conditions (see SRP Section 3.5.2.4.1)	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)
Off-Normal Design Event Conditions	• Temperature Variation Beyond Normal • 10% Fuel Rod Failure Combined with Off-Normal Temperatures • Failure of One of the Confinement Boundaries • Partial Air Flow Blockage • Human Error • Out-of-Tolerance Equipment Performance • Equipment Failure • Instrumentation Failure • Faulty Instrumentation Calibration Other Events Relevant to the Design and Operations Summarize Events Considered under External Conditions (see SRP Section 3.5.2.4.2)	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Design-Basis Accident Design Events and Conditions	End Drop: (1) Lift height (or maximum acceleration) Side Drop: (1) Lift Height (or Maximum Acceleration) Tipover: (1) Acceleration (if applicable) Fire: (1) Duration (2) Temperature Complete Air Flow Blockage Explosive Overpressure Other Events Relevant to the Design and Operations (see SRP Section 3.5.2.4.3), as applicable	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)
Design-Basis Natural Phenomena Design Events and Conditions	Flood Earthquake Tornado Burial Under Debris Lightning Other potentially relevant events identified in 10 CFR Part 72 (see SRP Section 3.5.2.5), as applicable	Other conditions or events relevant to operations of the DSF facility as described in the appropriate section of this SRP chapter. Storage containers and other DSF SSCs (e.g., waste management SSCs)
Structural	Design Code (e.g., ASME, AISC): (1) Containment (2) Noncontainment (3) Basket (4) Trunnions (5) Storage radiation and protective shielding and enclosure (6) Transfer radiation and protective shielding and enclosure (7) Cooling structure or system Design Weight Design Cavity Pressure: (1) Normal, off-normal, accident Response and Degradation Limits: (1) Normal, off-normal, or accident	Design Code: • Other SSCs important to safety and SSCs that affect SSCs important to safety • Radiation and protective shielding • Waste management facility SSCs important to safety • Reinforced concrete

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Thermal	Maximum Design Temperatures: (1) Cladding (2) Other components Insolation (side, top, or bottom) Fill Gas: (1) Type (e.g., helium) (2) Initial fill pressure (at temperature) Modes of Heat Transfer Used in the Design	Maximum Design Temperatures: • Reinforced concrete • Maximum temperature gradients for structures subject to thermal stress Maximum stored materials decay heat load
Confinement	Description of Confinement Boundary Redundant Seals for Closure Maximum Leak Rate for Confinement Boundary: (1) Normal, off-normal, or accident (2) Justification of leakage rate (if not leaktight) Monitoring System Specifications	
Waste Management (SL)		Description of confinement of site-generated wastes and ventilation and treatment systems
Radiation Protection and Shielding	Storage Container: (1) Surface position (normal, off-normal, or accident) Exterior of Shielding: (1) Transfer configuration position (2) Storage configuration position (normal, off-normal, or accident) Controlled Area Boundary: (1) Dose rate (2) Annual dose (normal or off-normal) (3) Accident Dose Occupational Dose Estimates ALARA Considerations (public and occupational) in Design and Operations	DSF SSCs in addition to the storage containers ALARA policies and programs Health Physics and Radiation Protection programs Radiological Environmental Monitoring Program
Criticality	Method of Control: Geometry, Fixed Poison, Soluble Poison Minimum Boron Concentration: Fixed and Soluble Poison Maximum K_{eff} Burnable Neutron Absorber Credit Burnup Credit Analysis	

Applicability	Scope for Certificate of Compliance and Specific License Reviews	Additional Scope for Specific License Review
Decommissioning		Design for decontamination and decommissioning
Materials	Cladding Hoop Stress Corrosion	
Operating Procedures	Normal and Off-Normal Post-Accident and Natural Phenomenon Event	
Acceptance Tests and Maintenance	Industry Codes and Standards	
Technical Specifications	Operational Controls and Limits	

3.5.3.1 *General*

Check the SAR chapter on design criteria and ensure that the descriptions are consistent with the descriptions in the sections of the SAR that address confinement, cooling, subcriticality, radiation protection, decommissioning, retrieval capability, and ALARA considerations. Verify that the SAR identifies and evaluates the design criteria and bases for the system as a whole.

Determine that the criteria derived from the site characteristics **(SL)** and generic site characteristics **(CoC)** and accident analyses (accident and off-normal conditions) are consistent with the analyses used in the qualification of the SSCs. For DSFs, verify that these criteria are equivalent to those proposed in site characteristics chapter of the SAR.

Confirm that the applicant's general design criteria reflect consideration of ALARA as applicable and appropriate. For specific license applications, the criteria should reflect any stated applicant ALARA goals and policies.

Verify that criteria defining the response of SSCs to normal, off-normal, and accident conditions are satisfactory.

Determine that the application presents design criteria for normal conditions and operations that do not result in or allow any degradation of the capabilities of the DSS or DSF. Ensure that the SAR sufficiently describes routine maintenance that would correct any "wear and tear" from normal conditions and operations that would degrade the capabilities of the DSS or DSF.

(SL) Determine that the application presents design criteria for off-normal conditions that do not permit any degradation of the capabilities of the DSF, assuming contingency operations during and following off-normal conditions. The NRC does not require that radioactive material handling or waste processing functions or capabilities at a storage facility continue during an off-normal condition or that such operations resume immediately. The licensee may impose inspections and system checkouts following any event or condition.

Determine that the application presents design criteria for accident conditions that do not permit the degradation of SSCs important to safety, including, but not limited to, (1) reduced radioactive material handling and waste processing capability **(SL)**, (2) reduced capability to withstand further accident conditions without excess response and without remedial action, and (3) reduced ability to provide functions for the full system or facility life time without remedial action. Determine that design criteria for accident conditions prevent (1) criticality, (2) unacceptable releases of radioactive material, (3) unacceptable radiation doses for the public and workers, and (4) loss of retrieval capability.

The NRC does not require the assumption of multiple failure scenarios of SSCs important to safety unless these multiple failure scenarios are credible consequences of the initiating event.

The NRC requires analysis or testing of SSCs for some events (e.g., cask drop or tipover) even though the events may be determined as noncredible in the accident analysis. Verify that the application presents criteria for the survival of SSCs important to safety for these “nonmechanistic” events as the same as the criteria for the survival of credible accidents.

3.5.3.2 *Other Safety Protection Systems*

Review procedures for the evaluations of design criteria for other safety protection systems applicable to each of the relevant chapters of the SAR are discussed in detail in the respective chapters of this SRP. Coordinate the review of each chapter with the applicable disciplines to ensure that multidisciplinary issues that impact more than one chapter are addressed.

Regardless of where the descriptions and associated criteria are located in the SAR, include a description and evaluation of the safety protection systems in the chapter of the safety evaluation report on principal design criteria. The system descriptions should address the functions of the various system components in providing confinement, cooling, subcriticality, radiation protection of the public and workers, and SNF retrievability. Also, ensure the SAR describes summary criteria for the performance of the system as a whole in providing for these capabilities or functions. Verify that the design-basis assumptions presented in the SAR are consistent with and reasonable for actual site and facility conditions. Include a description and evaluation of the DSS or DSF storage container(s) design’s compatibility with removal from a reactor site or from the DSF, transportation, and ultimate disposition of the stored SNF.

Verify that the SAR describes and evaluates criteria relating to redundancy and allowable levels of response by the DSS or DSF SSCs under normal, off-normal, and accident conditions and events. In general, no unacceptable degradation in physical condition or functional performance should result from normal or off-normal conditions. Verify that the design criteria regarding limits of permissible response and degradation resulting from an accident condition are evaluated against SSC capabilities to perform the principal safety functions. Considerations of permissible responses should include detectability and corrective actions that may be proposed as conditions of system use.

The NRC staff accepts that both routine surveillance programs and active instrumentation meet the intent of “continuous monitoring” as required in 10 CFR 72.122(h)(4).

Note that some DSS or DSF designs may contain a component or feature for which continued performance over the license or certified storage period has not been demonstrated to the staff with a sufficient level of confidence (e.g., rubber “O” rings). Therefore, the NRC may require the use of active instrumentation if the failure of that system or component causes an immediate threat to the public health and safety and if that failure would not be detected by any other means. In some cases, to demonstrate compliance with 10 CFR 72.122(h)(4), the applicant or the NRC may propose a technical specification requiring such instrumentation as part of the first use of a DSS for a CoC application, or as part of operations of the DSF for a specific license application. For DSSs, after first use, and if warranted and approved by the NRC, such instrumentation may be discontinued or modified.

Verify that the applicant has met the intent of continuous monitoring so that the applicant can determine when corrective action needs to be taken to maintain safe storage conditions.

3.5.4 Design Criteria for Other Structures, Systems, and Components (SL)

Verify that the design bases and criteria for other SSCs not important to safety meet the general regulatory requirements in 10 CFR 72.24(a)–(h) and (l) and 10 CFR 72.120.

Typical concerns for general design criteria reviews of other SSCs not important to safety include, but are not limited to, adequate functional performance, interfacing with other SSCs, potentially adverse interactive effects, and recognition of appropriate site characteristics. Confirm with the other reviewers that the application includes descriptions of the other SSCs that are relevant to their review of the facility and that the descriptions are sufficient to enable evaluation of facility compliance (design and operations) with the relevant regulatory requirements. In determining the SSCs and level of detail needed for the review, consider the descriptions provided in the final safety analysis reports for facilities that the NRC has previously licensed, as appropriate. For example, for a facility that is not co-located with a 10 CFR Part 50 or 52 facility, a review of the information in the final safety analysis reports for similar previously licensed facilities would provide useful insights.

3.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 3.4 of this SRP. If the documentation submitted with the application fully supports positive findings for each of the regulatory requirements, the statements of findings should be similar to the following:

F3.1 The SSCs have been classified as important to safety or not important to safety and meet the requirements given in 10 CFR 72.24(b) for specific licenses and 10 CFR 72.236 for CoCs.

F3.2 The SAR and docketed materials adequately identify and characterize the SNF to be stored in a DSS or DSF, reactor-related GTCC waste to be stored at a specific license DSF, high-level radioactive waste to be stored at a specific license MRS, as applicable. The acceptable form of the reactor-related GTCC waste and HLW is only solid and meets the requirements given in 10 CFR 72.120(b) and (c).

F3.3 The SAR and docketed materials adequately define the bounding conditions under which the DSF or DSS is expected to operate in accordance with the requirements of 10 CFR 72.24(a), 10 CFR 72.92, 10 CFR 72.94, and 10 CFR 72.122(b)(c) for specific license applications, and 10 CFR 72.236 for CoC applications.

F3.4 The SAR and docketed materials relating to the design bases and criteria for structures categorized as important to safety meet the requirements given in 10 CFR 72.24(c); 10 CFR 72.102; 10 CFR 72.103; 10 CFR 72.104(a); 10 CFR 72.106(b); 10 CFR 72.120(a)(b)(c)(d); 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c)(d)(f); 10 CFR 72.126(a)(d) for specific license applications; and 10 CFR 72.236 for CoC applications.

F3.5 The SAR and docketed materials meet the regulatory requirements for design bases and criteria for thermal consideration as given in 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c)(d)(f)(g)(h)(i);

10 CFR 72.128(a)(4) for specific license applications; and
10 CFR 72.236(f) for CoC applications.

F3.6 The SAR and docketed materials relating to the design bases and criteria for shielding, confinement, radiation protection, and ALARA considerations meet the regulatory requirements as given in 10 CFR 72.24(c), 10 CFR 72.104, 10 CFR 72.106, 10 CFR 72.122(a–i), 10 CFR 72.126, 10 CFR 72.128 for specific license applications, and 10 CFR 72.236(b)(d) for CoC applications.

F3.7 The SAR and docketed materials relating to the design bases and criteria for criticality safety meet the regulatory requirements as given in 10 CFR 72.124 and, for CoC applications, 10 CFR 72.236(c).

F3.8 The SAR and docketed materials relating to materials selection meet the regulatory requirements as given in 10 CFR 72.24(c)(3), 10 CFR 72.120(d), 10 CFR 72.122(a)(b)(c), 10 CFR 72.124(a)(b), 10 CFR 72.128(a)(2) for special license applications, and 10 CFR 72.124(a)(b) and 10 CFR 72.236(b)(c)(d)(g)(m) for CoC applications.

F3.9 **(SL)** The SAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.24(c)(1), (c)(2), (c)(4); 10 CFR 72.104; 10 CFR 72.106; 10 CFR 72.120(a)(b)(c)(d); 10 CFR 72.122; 10 CFR 72.124; and 10 CFR 72.126(a)(d).

F.3.10 **(SL)** The SAR and docketed materials relating to design criteria for decommissioning of the facility comply with the regulatory requirements in 10 CFR 72.130 and the guidance in applicable portions of RGs 1.184 and 1.191.

F3.11 **(SL)** The SAR and docketed materials relating to the design bases and criteria for retrieval capability meet the regulatory requirements in 10 CFR 72.122(a), (b)(1), (b)(2), and (b)(3), (c)(f)(h)(l).

F3.12 **(SL)** The SAR and docketed materials relating to the design bases and criteria for other SSCs not important to safety, but subject to NRC approval, meet the general regulatory requirements in 10 CFR 72.24(a–h) and (l) and the appropriate requirements in 10 CFR 72.120 and 10 CFR 72.122.

F3.13 **(CoC)** The SAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.236(b).

The reviewer should provide a summary statement similar to the following:

The staff finds that the descriptions of the DSF or DSS characteristics are such that appropriate design criteria and bases for the DSF or DSS could be defined and evaluated. The staff concludes that the principal design criteria for the DSF or DSS are acceptable with regard to meeting the regulatory requirements in 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation, itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. Chapters 3 through 16 of the safety evaluation report present a more detailed evaluation of the design criteria and an assessment of compliance with those criteria.

3.7 References

10 CFR Part 20, “Standards for Protection against Radiation.”

10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

10 CFR Part 71, “Packaging and Transportation of Radioactive Material.”

10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.”

10 CFR Part 73, “Physical Protection of Plants and Materials.”

American National Standards Institute (ANSI) N210-1976/American Nuclear Society (ANS) 57.2-1983, “Design Objectives for Light Water Reactor Spent Fuel Pool Storage Facilities at Nuclear Power Stations.”

ANSI/ANS 57.9-1992, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

U.S. Nuclear Regulatory Commission Bulletin 96-02, “Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety Related Equipment,” dated April 11, 1996 (ADAMS Accession No. ML082590698).

National Fire Protection Association (NFPA) 780, “Standard for the Installation of Lightning Protection Systems.”

NFPA 70, “National Electrical Code.”

NUREG-0800, U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” March 2007 (ADAMS Accession No. ML070660036 (package)).

NUREG-1927, U.S. Nuclear Regulatory Commission, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” Revision 1, June 2016, (ADAMS Accession No. ML16179A148).

NUREG-2174, U.S. Nuclear Regulatory Commission, “Impact of Variation in Environmental Conditions on the Thermal Performance of Dry Storage Casks, Final Report,” issued March 2016 (ADAMS Accession No. ML16081A181).

Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” (ADAMS Accession No. ML070310035).

Regulatory Guide 1.29, “Seismic Design Classification,” (ADAMS Accession No. ML16118A148).

Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants,” (ADAMS Accession No. ML003740388).

Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” (ADAMS Accession No. ML13210A432).

Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” (ADAMS Accession No. ML070260029).

Regulatory Guide 1.76, “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,” (ADAMS Accession No. ML070360253.pdf).

Regulatory Guide 1.86, “Termination of Operating Licenses for Nuclear Reactors,” (ADAMS Accession No. ML16099A267).

Regulatory Guide 1.92, “Combing Modal Responses and Spatial Components in Seismic Response Analysis,” (ADAMS Accession No. ML12220A043).

Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants,” (ADAMS Accession No. ML003740308).

Regulatory Guide 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants,” (ADAMS Accession No. ML15356A213).

Regulatory Guide 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” (ADAMS Accession No. ML003739367).

Regulatory Guide 1.184, “Decommissioning of Nuclear Power Reactors,” (ADAMS Accession No. ML13144A840).

Regulatory Guide 1.189, “Fire Protection for Nuclear Power Plants,” (ADAMS Accession No. ML092580550).

Regulatory Guide 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” (ADAMS Accession No. ML011500010)

Regulatory Guide 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” (ADAMS Accession No. ML033280143).

Regulatory Guide 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,” (ADAMS Accession No. ML092730314).

Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” (ADAMS Accession No. ML070310619).