16 ACCIDENT ANALYSIS EVALUATION

16.1 Review Objective

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) accident analysis review is to conduct a systematic evaluation of the applicant's identification and analysis of hazards for both off-normal and accident conditions involving structures, systems, and components (SSCs) important to safety, and other SSCs that may affect SSCs important to safety. This chapter provides guidance for a minimum set of events the applicant should consider in its safety analysis report (SAR). Depending on the design of the dry storage system (DSS) or dry storage facility (DSF) or the DSF location, the applicant may need to consider additional events or additional DSS or DSF SSC configurations against which the events should be evaluated.

The accident analysis review ensures that the applicant has conducted thorough accident analyses as reflected by completing the following:

- identified all relevant off-normal conditions for the DSF
- identified all credible accidents for the DSF
- identified the envelop or bounding set of off-normal conditions and accident conditions that are relevant to the DSS design and operations and for which the DSS is analyzed to ensure performance of its design functions
- provided complete information in the SAR
- analyzed the safety performance of the DSF or DSS in each review area
- fulfilled all applicable regulatory requirements

16.2 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 DSF. It also applies to the review of applications for a certificate of compliance of a DSS for use at a general license facility. Sections or paragraphs of this chapter that apply only to specific license applications are identified with "**(SL)**."

16.3 Areas of Review

The accident analysis evaluation covers the applicant's identification and analysis of hazards, as well as the summary analysis of system responses. It places particular emphasis on the safety performance of the storage container under off-normal events and conditions and accident or design-basis events.

This chapter addresses the following areas of review that may encompass a comprehensive accident analysis evaluation:

- cause of the event
- definition of operating environment and physical parameters
- detection of the event
- summary of event consequences and regulatory compliance
- corrective course of action

The review for each off-normal and each accident condition, as presented in the SAR, should address each of these five areas.

16.4 Regulatory Requirements and Acceptance Criteria

This section summarizes those parts of 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," that are relevant to the review areas this chapter addresses. The reviewer should refer to the exact language in the regulations. Tables 16-1a and 16-1b match the relevant regulatory requirements to the areas of review this chapter covers. Note that regulatory requirements in 10 CFR Part 20, "Standards for Protection Against Radiation," (see SRP Chapters 10A and 10B, "Radiation Protection Evaluation," for a DSF and a DSS, respectively) also apply for off-normal events, and the reviewer must consider those regulations in evaluations of these events.

Accidents and natural phenomena events may share common regulatory and design limits. Consequently, this chapter sometimes refer to these scenarios collectively as accident conditions.

By contrast, off-normal conditions (anticipated occurrences) are distinguished, in part, from accidents or natural phenomena by the appropriate regulatory guidance and design criteria. For example, the radiation dose from an off-normal event, in combination with doses from normal operations, must not exceed the limits specified in 10 CFR Part 20 and 10 CFR 72.104(a), whereas the radiation dose from an accident or natural phenomenon must not exceed the limits specified in 10 CFR 72.106(b). Accident conditions may also have different allowable structural and thermal criteria compared to off-normal conditions.

Table 16-1a Relationship of Regulations and Areas of Review for a DSF (SL)

Areas of Review	10 CFR Part 72 Regulations					
	72.24(a)(d) (1)(2), (m)	72.90 (c)	72.92	72.94	72.100 (a)	72.104 (a)(c)
Cause of the Event	●	●	●	●		
Detection of the Event						
Summary of Event Consequences and Regulatory Compliance	●			●	●	●
Corrective Course of Action	●				●	●

Areas of Review	10 CFR Part 72 Regulations (cont.)				
	72.106 (b)	72.122(b)(c)(d)(e)(g)(h)(i)(j) (k)(1)(3)(4), (l)	72.124	72.126 (b)(c)(d)	72.128 (a)(1)(2) (3)(4)
Cause of the Event		●			
Detection of the Event		●	●	●	●
Summary of Event Consequences and Regulatory Compliance	●	●	●	●	●
Corrective Course of Action	●	●			●

Table 16-1b Relationship of Regulations and Areas of Review for a DSS (CoC)

Areas of Review	10 CFR Part 72 Regulations					
	72.104 (a)(c)	72.106 (b)	72.122(b)(c)(d) (g)(h)(i)(j) (k)(4), (l)[A]	72.124	72.128 (a)(1)(2) (3)(4)[a]	72.236 (c)(d)(l)
Cause of the Event			●			
Detection of the Event			●	●	●	
Summary of Event Consequences and Regulatory Compliance	●	●	●	●	●	●
Corrective Course of Action	●	●	●		●	

A Note that while 10 CFR 72.122, "Overall Requirements," and 10 CFR 72.128, "Criteria for Spent Fuel, High-Level Radioactive Waste, and Other Radioactive Waste Storage and Handling," 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.

In general, the accident analysis evaluation seeks to ensure that the design and the applicant's hazard identification and analyses of related DSS or DSF responses fulfill the relevant design and regulatory criteria (including the criteria discussed in Sections 16.4.1–6 below) for the following types of events or conditions. The hazard identification and analyses should include appropriate consideration of the different operation configurations that may occur, or are likely to occur, for the DSS or DSF design, including temporary configurations. Such configurations may include construction activities to expand an operating array of storage containers that removes or exposes shielding materials.

- Off-Normal Events and Conditions—The following is a minimum list of the off-normal events that the applicant must consider in the SAR:

  – partial vent blockage (if applicable)
  – operational events resulting in radioactive release
  – off-normal ambient temperatures
  – off-normal events associated with the facility **(SL)**

- Accident Events and Conditions—The following is a minimum list of the accident conditions that the applicant must consider in the SAR:

  – storage container tipover
  – storage container drop
  – flood
  – fire and explosion
  – lightning
  – earthquake
  – loss of shielding
  – adiabatic heatup
  – tornadoes and missiles generated by natural phenomena
  – accidents at nearby sites **(SL)**
  – building structural failure onto SSCs **(SL)**

- Other Off-Normal and Accident Events and Conditions—In addition to all of the accidents and off-normal events listed above, the applicant must list and evaluate other off-normal and accident events that are specific to the applicant's design. These events include those that might have negligible consequences for most designs, but characteristics of the proposed design may result in nonnegligible consequences for the same events (e.g., crane malfunction). If these other off-normal and accident events have results that are enveloped by the events previously considered, the applicant must provide the basis for this evaluation, and no further consideration is required. It is expected that the required off-normal events and accidents listed in this section may envelope events such as human errors, operational errors, and material aging.

16.4.1 Dose Limits for Off-Normal Events

During normal operations and off-normal conditions (that is, anticipated occurrences), the DSF must meet the annual dose limits in 10 CFR 72.104(a). The DSF applicant must also demonstrate that the DSF will meet the requirements specified in 10 CFR Part 20.

16.4.2 Dose Limit for Accidents

The dose from any accident to any individual located on or beyond the nearest boundary of the controlled area may not exceed the limits specified in 10 CFR 72.106(b).

16.4.3 Criticality

In accordance with 10 CFR 72.124(a) and, for DSSs, 10 CFR 72.236(c), the licensee must maintain the SNF in a subcritical condition under credible conditions (i.e., effective neutron multiplication factor (k_{eff}), including all biases and uncertainties, equal to or less than 0.95). DSS or DSF SSCs must be designed so that at least two unlikely, independent, and concurrent or sequential changes in the conditions essential to nuclear criticality safety must occur before a nuclear criticality accident is possible (double contingency). Similar criteria should be applied, as appropriate, to other radioactive materials to be stored at a DSF (e.g., HLW at a MRS).

16.4.4 Confinement

The regulation in 10 CFR 72.128(a) states that DSF systems must be designed with confinement structures and systems and 10 CFR 72.236(d) requires DSS' confinement (and shielding) systems be sufficient to meet the requirements in §§ 72.104 and 72.106. The applicant must evaluate the DSS or DSF SSCs and features important to safety using appropriate tests or by other means acceptable to the NRC to demonstrate that the SSCs will reasonably maintain confinement of radioactive material under accident conditions, consistent with 10 CFR 72.122(b), 10 CFR 72.122(c), and 10 CFR 72.122(h) for DSFs and as specified in 10 CFR 72.236(l) for DSSs. The applicant should show that a breach of a confinement barrier does not occur as a result of any off-normal or accident event. A confinement system is defined in 10 CFR 72.3, "Definitions," as a system, including ventilation, which acts as a barrier between areas containing radioactive substances and the environment.

16.4.5 Recovery and Retrievability

Recovery is the capability of returning the stored radioactive materials from an accident to a safe condition without endangering public health and safety or causing significant or unnecessary exposure to workers. Any potential release of radioactive materials during recovery operations must not exceed the radioactive exposure limits in 10 CFR Part 20.

Retrievability is applicable only during normal and off-normal conditions and does not apply to accident conditions. Retrievability is specified in 10 CFR 72.122(l), which states that "storage systems must be designed to allow ready retrieval of spent fuel, high level radioactive waste, and reactor-related greater than class C waste for further processing or disposal." Ready retrieval is defined as the ability to safely remove the SNF, HLW, or reactor-related GTCC from storage for further processing or disposal. A storage system must be designed to allow for ready retrieval in the initial design, amendments to the design, and in licenses and CoCs, as applicable, through the licensing period(s) of the design, including through renewals. The retrievability requirement applies to all ISFSIs operating under a general license or a specific license. The requirements in 10 CFR 72.236(m) for CoC holders states that "[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." Any potential release of radioactive materials during retrieval operations must not exceed the radioactive exposure limits in 10 CFR Part 20.

In order to demonstrate the ability for ready retrieval of SNF, a licensee should demonstrate it has the ability to perform any of the three options shown below. Note that these options may be utilized individually or in any combination or sequence, as appropriate:

- Remove individual or canned spent fuel assemblies from wet or dry storage.
- Remove a canister loaded with spent fuel assemblies from a storage cask or overpack, as applicable.
- Remove a cask or DSF storage container, as applicable, loaded with spent fuel assemblies from the storage location.

16.4.6 Instrumentation

For DSFs, the SAR must identify all instruments and control systems that must remain operational under normal, off-normal and accident conditions as required by 10 CFR 72.122(i). For DSSs, the SAR should show how the DSS design facilitates the general licensee's ability to meet 10 CFR 72.122(i).

16.5 Review Procedures

This section provides review guidance for each off-normal and accident event evaluation. The review guidance varies in complexity for each evaluation. In general, the staff's review includes the operating environment, the physical parameters, the methodology used, and the actual analysis the applicant performed as part of its review.

Items of unique or special safety significance should receive special emphasis. Refer to Chapter 3, "Principal Design Criteria Evaluation," of this SRP for a discussion of the SSCs important to safety.

The effects of various off-normal events and accidents may be interrelated, and some degree of overlap is expected to occur during the accident analyses review process. An example of such overlap would be a tornado missile accident leading to a loss of shielding as described in Section 16.5.2.7 of this chapter, or an accident reviewed according to Section 16.5.2.9. If two or more off-normal events and accidents are interrelated, assess the combined occurrence and effects of the interrelated off-normal and accident conditions.

Ensure that the applicant identifies and evaluates all relevant, credible off-normal and accident conditions, including any that are unique to the design and, for DFSs, the site. Ensure that the applicant's evaluations include the occurrence and effects of these events for all relevant and likely, or credible, operating configurations, including temporary normal conditions. Ensure that the applicant's evaluations address all relevant criteria. Coordinate the review with the other SAR reviewers to evaluate the design and site characteristics to determine whether all relevant off-normal and accident conditions have been identified and evaluated. The detailed evaluations of the conditions may be done in the accident chapter of the SAR or in the respective SAR chapters for each technical discipline, with the accident chapter merely summarizing and referencing the evaluations in those other chapters. In either case, coordinate with the reviewers of those chapters to ensure the evaluations are adequate and to verify that the design and regulatory criteria are met. Also identify those parameters that may need to be included in the technical specifications based on the accident evaluation analyses. For example, DSS applications are not for a particular site and so must make assumptions in its analyses regarding conditions, such as natural phenomena, that may occur at sites that may use the DSS. Some of the assumptions may need to be translated into one or more appropriate conditions in the technical specifications for the DSS.

Figure 16-1 shows the interrelationship between the accident analysis evaluation and the other areas of review described in this SRP.

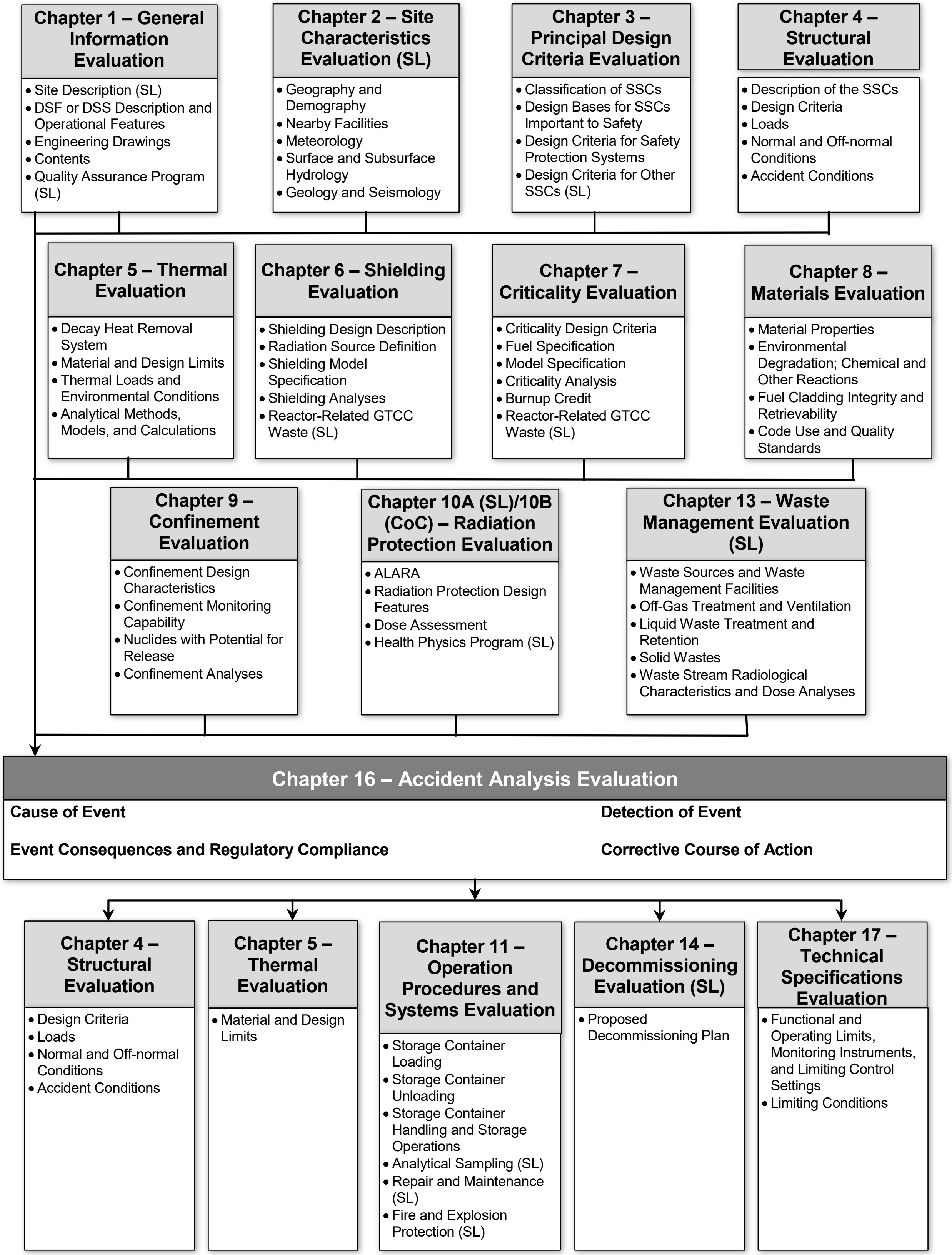

Figure 16-1 Overview of Accident Analysis Evaluation

For each off-normal and accident event described in these review procedures, verify that the applicant has addressed the following areas of review: (1) the cause of the event, (2) means of detecting the event, (3) a summary of the analysis of the event, including estimated consequences and comparison to regulatory limits, and (4) a corrective course of action.

- Cause of the Event—The applicant should describe the cause of the off-normal or accident condition. The description should include the chain of events that leads to the credible off-normal or accident condition and any bounding conditions.

- Definition of Operating Environment and Physical Parameters—The applicant should describe the conditions and environment that the DSF or DSS SSCs experience for off-normal and accident conditions. This includes parameters and information items such as the configuration and physical location (as applicable) of the DSS or DSF SSCs, ambient conditions, extent of degradation (e.g., fraction of vent blockage), surface contamination levels, properties of impact objects or surfaces, and sources of hazards (e.g., flood water source).

- Detection of the Event—The applicant may detect an event through surveillance programs or monitoring instrumentation and alarms. Surveillance programs and monitoring instrumentation and alarms should have reasonable flexibility to allow for the identification of an accident condition or noncompliance situation that has not been previously considered in the SAR. The method of detection will be intuitively obvious for some events, whereas other events (e.g., fuel rod rupture) may remain undetected for a significant period of time.

- Summary of Event Consequences and Regulatory Compliance—The applicant should address event consequences in each functional area corresponding to earlier chapters of the SAR (i.e., structural, thermal, shielding, criticality, confinement, materials, and radiation protection). This area of review includes evaluation of (1) the analysis method and (2) the event analysis. The SAR should describe the analysis method(s) the applicant used, including the tools and techniques. The SAR should present the analysis of the event, including the design criteria and design codes and standards, as applicable. This discussion should refer back to each SAR chapter in which the individual consequences are evaluated in detail. The applicant should provide a summary of the accident dose calculations and show that the consequences comply with the applicable regulatory criteria. For off-normal conditions, the applicant should include the resulting doses with the doses from normal operations in the evaluations for demonstrating the facility design and operations meet, or will meet, the requirements, including dose limits, demonstrate compliance within 10 CFR Part 20 as well as 10 CFR Part 72. As applicable and appropriate, the consequence analyses should address occupational doses as well as doses to members of the public.

- Corrective Course of Action—The applicant should identify what action(s), if any, would be necessary to recover from the event. If various courses of action are possible, the applicant should present a discussion concerning the selection of the most appropriate action. Because the SNF, HLW, or reactor-related GTCC, as applicable, must be readily retrievable after an off-normal event and after returning to storage after an accident, reloading the SNF, HLW, or reactor-related GTCC, as applicable, into a new storage container is a viable option. If corrective courses of action are to be included in operating procedures or administrative programs, then the applicable sections of the SAR that cover operating procedures and administrative programs should be referenced.

16.5.1 Off-Normal Events

This section discusses the review of off-normal conditions that may include malfunctions of systems, minor leakage, limited loss of external power, and operator error. The consequences of these events should not have a significant effect beyond the facility operation areas (e.g., handling, loading, storage areas).

Verify that the SAR also defines the analysis and design criteria and design codes and standards (as applicable) for each off-normal event as related to HLW or reactor-related GTCC waste storage and handling systems.

16.5.1.1 *Partial Vent Blockage (if applicable)*

For confinement systems, such as natural convection cooling systems that are subject to a temperature rise from a partial vent blockage, verify that the applicant has made an evaluation of the event. The purpose of the evaluation is not to establish a surveillance frequency, as in the case of the adiabatic heatup accident, but rather to establish that no critical temperature limits will be reached for an extended time period.

16.5.1.1.1 Define the Operating Environment and Physical Parameters

Verify that the SAR identifies the operating environment of the off-normal event, including the following:

- the operational configuration of the confinement system
- the fraction of vent blockage
- the ambient temperature
- the design-basis decay heat load

16.5.1.1.2 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, including assumptions and calculational models or experimental testing.

16.5.1.1.3 Off-Normal Event Analysis

Verify the identification of the vent flow area and revised vent flow loss coefficients associated with any blockage of the normal air inlet vent flow area.

Verify the air outlet temperature and the unit internal material maximum temperatures for all key DSS or DSF SSCs. Use the flow areas and flow loss coefficients assuming normal ambient air temperature (as defined in Chapter 3 of this SRP). Also use the maximum design-basis decay heat and the identical thermal models and computer codes that were used in the normal conditions thermal analysis of the DSS or DSF SSCs.

Compare the calculated maximum material temperatures with their respective off-normal temperature limits, and verify that no critical temperature limits will be reached for the time period.

Coordinate with the structural integrity reviewer to ensure that these temperatures are used to determine the appropriate allowable stress-intensity levels.

Verify that the applicant evaluated the worker dose required to clear debris that is blocking air inlet(s) using the design-basis calculated dose rate at the air inlets and an appropriate estimate of the time necessary to clear the vents. Ensure that the doses are below the worker dose limits in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

16.5.1.2 *Operational Events Resulting in Radioactive Release*

This subsection shows the process for evaluating a typical off-normal condition and any resulting radiological consequences.

16.5.1.2.1 Define the Operating Environment and Physical Parameters

Verify that the SAR describes the maximum allowable container surface contamination, based on applicable technical specifications or health physics procedures, or both. This contamination is usually expressed in terms of counts per minute, counts per unit surface area, or microcuries per square centimeter, and different values are provided for alpha contamination and beta or gamma contamination.

Verify the calculation of the total surface area of the SNF, HLW, or reactor-related GTCC waste container.

Verify the calculation of the total container surface contamination by multiplying the values of surface contamination (in terms of curies of activity per unit surface area) and surface area.

16.5.1.2.2 Review the Analysis Methodology

Verify that the SAR contains the 95-percent probability value for the atmospheric dispersion factor from the SNF storage facility container to members of the public at or beyond the controlled area boundary. The technical basis and applicability of the atmospheric dispersion value should be included. Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," provides detailed directions on acceptable methods for calculations of values of dispersion parameters. The NRC has previously accepted RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," or RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," for conservative generic values of atmospheric dispersion factors in the absence of site-specific meteorological data.

Use the guidance in Chapters 6, "Shielding Evaluation," 9, "Confinement Evaluation," and 10A and 10B to evaluate the applicant's analyses of event consequences for releases. Verify that the applicant uses an appropriate method for evaluating radiological consequences to operations personnel and members of the public on the site.

16.5.1.2.3 Off-Normal Event Analysis

Use the guidance in Chapters 6, 9, 10A, and 10B to evaluate the applicant's analyses of event consequences for releases. Verify that the applicant has determined the dose consequences to individuals on site for purposes of demonstrating compliance with 10 CFR 20.1101(d), 10 CFR 20.1201(a), and 10 CFR 20.1301(b).

16.5.1.3 *Off-Normal Ambient Temperatures*

Off-normal ambient temperatures are expected to occur during the operational life of the DSS or DSF. The numerical values of off-normal ambient temperatures are expected to be greater than the normal ambient temperature but less than the accident ambient temperature. The higher probability of occurrence of off-normal ambient temperatures, compared to the accident temperatures, requires that calculated material temperatures as a result of off-normal ambient temperatures meet the normal operational material temperature limits.

16.5.1.3.1 Define the Operating Environment and Physical Parameters

Verify that the SAR specifies appropriate maximum and minimum off-normal ambient temperatures. Examples of previously accepted conditions include maximum and minimum ambient temperature values of 52 degrees Celsius (°C) (125 degrees Fahrenheit (°F)) and -40 °C (-40 °F). For previously licensed or certified DSF or DSS, a typical annual average ambient temperature has been 24 °C (75 °F). The maximum and minimum ambient temperature values should equal the 99-percent values in Table 1, "Climatic Conditions for the United States," in the American Society of Heating, Refrigeration and Air-Conditioning Engineers' publication, "ASHRAE Handbook—Fundamentals." If the DSF or DSS does not correspond with a location cited in this reference, verify that the applicant has supplied technical justification for using the same climatic data as shown in the ASHRAE Handbook.

Similarly, verify the site-specific or generic value of solar insolation or heat flux for the DSF or DSS. This value should be used in conjunction with the normal and off-normal maximum ambient temperature, but a value of zero solar heat flux should be used with the minimum ambient air temperature scenario.

16.5.1.3.2 Review the Analysis Methodology and Off-Normal Event Analysis

Verify that the applicant calculated the steady-state temperature distribution within the DSS or DSF SSCs using the same methodology and computer codes that were used for the normal ambient air temperature scenario.

Evaluate the calculated temperature distribution in terms of material temperature limits (e.g., fuel cladding, concrete, and proprietary neutron shielding materials) and thermal stresses. The material temperature limits should be consistent with the acceptable limits identified in the thermal analysis evaluation.

16.5.1.4 *Other Off-Normal Events Associated with the Facility*

16.5.1.4.1 Define the Off-Normal Events

The following off-normal events are estimated to occur with a frequency of approximately once per year of storage operation and should be evaluated regardless of which American National

Standards Institute (ANSI) standard the SAR cites. The list is intended to be representative and not all inclusive.

- failure of any single active component to perform its intended function on demand
- spurious operation of certain active components such as a relief valve or a control valve
- loss of external power supply for a limited duration (e.g., less than 8 hours) that could cause loss of cooling
- single-operator error followed by proper corrective action
- minor leakage from component

If the SAR cites ANSI/American Nuclear Society (ANS) 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," then the applicant should consider a single failure in the electrical or control system in addition to the above events.

16.5.1.4.2 Define the Operating Environment

Verify that the SAR identifies the operating environment and conditions of the off-normal events.

16.5.1.4.3 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the off-normal events, possibly including the following:

- level or temperature of water at the time of failure or spurious operation of active components
- any protective devices designed to mitigate the consequences of the off-normal events
- alarms and response times for corrective action

16.5.1.4.4 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology for evaluating the consequences of the off-normal events, including assumptions used as a part of the off-normal event.

16.5.1.4.5 Off-Normal Event Analysis

Verify that the SAR presents the analysis, design criteria, and design codes and standards for each of the off-normal events that the SAR defines. The following codes and standards are the primary design and construction codes acceptable to the NRC; consult ANSI/ANS 57.2 or ANSI/ANS 57.7, "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)," for a more detailed listing of design codes and standards.

- SNF storage racks
  - American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components."

- SNF storage container and HLW or reactor-related GTCC handling systems
  - Crane Manufacturers Association of America, Specification No. 70, “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes”
  - ANSI N14.6, “Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More,” for special lifting devices for shipping containers weighing more than 10,000 pounds
  - ASME B30.2, “Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist),” for overhead and gantry cranes for ANSI/ANS 57.2 designs
- SNF or waste form handling systems
  - Institute of Electrical and Electronics Engineers (IEEE) C2, “National Electrical Safety Code”
  - IEEE 835, “Standard Power Cable Ampacity Tables.”
  - National Fire Protection Association (NFPA) 70, “National Electrical Code”
  - ASME B30.16, “Overhead Hoists (Underhung)”
- heating, ventilation, and air-conditioning systems
  - ASHRAE Handbook
  - Air Movement and Control Association standards and application guides
  - ASME N509, “Nuclear Power Plant Air-Cleaning Units and Components”
  - International Code Council, “International Building Code”
  - International Code Council, “International Mechanical Code”
- buildings
  - ANSI/American Concrete Institute (ACI) 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” for reinforced concrete for ANSI/ANS 57.2 designs and ANSI/ACI 318, “Building Code Requirements for Structural Concrete and Commentary,” for ANSI/ANS 57.7 designs, and as appropriate for ANSI/ANS 57.9 designs
  - NFPA 780, “Standard for the Installation of Lightning Protection Systems”
  - American Iron and Steel Institute, “Steel Products Manual”
- radioactive waste treatment
  - 10 CFR Part 71, “Packaging and Transportation of Radioactive Material”
  - 10 CFR Part 20 for radiation protection

Verify that the applicant has identified any radiological consequences related to these other off-normal events associated with the facility and has calculated dose rates and doses.

16.5.2 Accidents

Verify that the SAR includes a rigorous discussion of potential accidents, both external natural events and man-induced events. The accident analysis review focuses on the effects of the natural phenomena and man-induced events on SSCs important to safety. Ensure that the SAR presents analytical techniques, uncertainties, and assumptions.

For those SNF storage facility license applications that propose to use a certified DSS listed in 10 CFR 72.214, "List of Approved Spent Fuel Storage Casks," the SAR may reference rather than repeat the evaluation of the impacts of accidents to the DSS that have been previously evaluated as part of that DSS's certification. However, verify that the SAR shows that the analyses for the DSS (or the conditions used for those analyses) bound the relevant conditions of the proposed facility. Also, ensure that the SAR addresses any effects to the facility and facility equipment from the event (e.g., pool lining damage).

For each accident condition, the SAR discussion of each event should include (a) a discussion of the cause of the event, (b) the means of detection of the event, (c) an analysis of the consequences (particularly any radiological consequences) and the protection provided by devices or systems designed to limit the extent of the consequences, and (d) any actions required of the operator. For each accident the applicant should provide and discuss the results of dose calculations.

16.5.2.1 *Storage Container Tipover*

Confirm that the SAR evaluates the container tipover accident. For this analysis, the NRC will accept container tipover about a lower corner onto a receiving surface from a position of balance with no initial velocity. Other analyses of tipover accidents may also be accepted; for example, the NRC has also accepted analysis of container drops with the longitudinal axis horizontal that, together with the longitudinal axis vertical, could bound a nonmechanistic tipover analysis.

16.5.2.1.1 Define the Operating Environment

Verify that the SAR identifies the operating environment of the accident, including the following:

- the operational configuration of the storage container (e.g., a storage container on the pad, a canister inside a transfer cask suspended from a cable on a crane or hoist, with or without impact limiters)

- the physical location of the tipover accident

16.5.2.1.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters necessary to evaluate the accident, including the following:

- the receiving surface upon which the storage container slaps down (e., the storage pad materials, dimensions, and properties and the foundation properties; the surface must be defined to quantify the maximum deceleration levels)

- the design of the storage container and associated SSCs (i.e., material properties, dimensions, and weights)

16.5.2.1.3 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, such as the following:

- reference to horizontal and vertical analyses if the tipover can be shown to be bounded by these two accidents

- specific analysis modeling tools such as closed-form manual techniques or computer codes

16.5.2.1.4 Accident Analysis

Verify that the SAR presents the accident analysis, design criteria, and design codes and standards, such as the following:

- deceleration level

- design code for evaluation—the NRC accepts the ASME B&PV Code, Section III, Service Level D

- specification if elastic or elastic-plastic analysis is used and appropriate citation of design code

- evaluation of calculated stress-intensity level against the allowable stress-intensity level at the design temperature and pressure for each component in the storage container; the evaluation should also consider components associated with confinement O-rings or seals and relevant pressure-monitoring systems for bolted lids

- evaluation of buckling stability for each component member of the storage container subject to compressive loading

- evaluation of deformation of container internal members that contribute to the spacing geometry for criticality safety

- evaluation of deformation of, or damage to, the SNF or HLW (MRS only) contents of the storage container

- evaluation of damage or deformation of the reactor-related GTCC waste storage container **(SL)**

- evaluation of impacts to other facility systems or features **(SL)**

- calculation of dose consequences

16.5.2.2 *Storage Container Drop*

The drop of the storage container is one of the hypothetical accident scenarios that the applicant must evaluate. The following steps provide an outline of the methodology that the applicant should provide in the SAR. The steps are representative of a typical SNF storage container but are not intended to cover every aspect of every possible container design.

16.5.2.2.1 Define the Operating Environment

Verify that the SAR identifies the operating environment of the accident, including the following:

- the operational configuration of the storage container (e.g., a storage container with no other SSCs, a canister inside the transfer cask or transportation package, with or without impact limiters)

- the storage container orientation at the moment of impact (i.e., end drop on top or bottom, side drop at various azimuths, and corner drop at various azimuths and inclinations)

- the physical location of the drop accident (i.e., outside the SNF pool building or inside the SNF pool building or other DSF structures or buildings where the materials stored at the site or the storage containers are handled)

16.5.2.2.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the accident, including the following:

- the receiving surface upon which the storage container impacts (i.e., the storage pad materials, dimensions, and properties and the foundation properties, or dimensions and properties of the SNF pool or building floor materials or floor materials of other DSF buildings where operations occur); the surface should be sufficiently characterized to quantify the maximum deceleration levels

- the design of the storage container and associated SSCs (i.e., material properties, dimensions, and weights)

- the drop height of the storage container onto the receiving surface for each orientation; the analysis should use the maximum height above the impact surface to which the container could be lifted

16.5.2.2.3 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, such as the following:

- static equivalent deceleration with appropriate dynamic load factors
- dynamic modeling with appropriate test data to benchmark deceleration
- specific analysis modeling tools such as manual techniques or computer codes

16.5.2.2.4 Accident Analysis

Verify that the SAR presents the accident analysis, design criteria, and design codes and standards, such as the following:

- deceleration level for each case considered
- design code for evaluation—the NRC accepts the ASME B&PV Code, Section III, Service Level D
- specification if elastic or elastic-plastic analysis is used and appropriate citation of the design code
- evaluation of calculated stress-intensity level against the allowable stress-intensity level at the design temperature and pressure for each component member of the storage container; the evaluation should also consider components associated with confinement O-rings or seals and relevant pressure-monitoring systems for bolted lids
- evaluation of the buckling stability for each component member of the storage container subjected to compressive loading
- evaluation of the deformation of container internal members that contribute to spacing geometry of the SNF assemblies or HLW materials that are subject to criticality safety as given in Chapter 7, "Criticality Evaluation," of this SRP
- evaluation of deformation of, or damage to, the SNF or HLW (MRS only) contents of the storage container
- evaluation of damage or deformation of the reactor-related GTCC waste storage container **(SL)**
- evaluation of impacts to other facility systems or features **(SL)**
- calculation of accidental dose consequences

16.5.2.3 *Flood*

The flood accident is one of the accidents that the applicant must evaluate, in accordance with 10 CFR 72.122(b)(2)(i). Coordinate the review of the flood evaluation in the SAR with that of the site characteristics for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of flood parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters, including the evaluation of any entrained debris. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.3.1 Define the Operating Environment

Verify that the SAR identifies the operating environment of the accident, including the following:

- the operational configuration of the storage container or other SSCs important to safety (e.g., a storage container on a storage pad, a storage container in a shielding structure)

- the physical location of the SSCs important to safety at the time of the hypothetical flood **(SL)**

- the source of the flood water based on historical data for the site as well as current and projected site characteristics (e.g., nearby dams and reservoirs) **(SL)**

- objects that may pose a flood-borne hazard

16.5.2.3.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the flood condition, including the following:

- the quantity of flood water (i.e., the static head of water and the maximum flow velocity)

- any protection devices placed at the site to prevent containers from tipping over or sliding

- any protections against flood-borne objects **(SL)**

16.5.2.3.3 Review the Analysis Methodology

Verify that the SAR defines the analysis methodology used in the evaluation, such as the following:

- sliding and overturning
- evaluation of external pressure stress intensity

16.5.2.3.4 Accident Analysis

Verify that the SAR presents the accident analysis, design criteria, and design codes and standards, such as the following:

- the design-basis flood conditions

- the determination of the maximum drag force acting on the confinement container or other SSCs important to safety

- the determination of the pressure loading acting on the SSCs

- the determination of the external pressure stress intensity and comparison with the allowable stress as found in the ASME B&PV Code, Section III, Service Level C

- determination that there is no sliding and overturning of the SSCs, or other damage to SSCs

- determination of the consequences of impacts from flood-borne objects and hazards

- compliance with RG 1.59, “Design Basis Floods for Nuclear Power Plants,” and RG 1.102, “Flood Protection for Nuclear Power Plants,” where applicable

- calculation of dose consequences

16.5.2.4 *Fire and Explosions*

The applicant must evaluate fire and explosion accidents, in accordance with 10 CFR 72.122(c). Coordinate the evaluation of these accidents with that for the site characteristics, as defined in the SAR and reviewed using Chapter 2, “Site Characteristics Evaluation for Dry Storage Facilities,” of this SRP for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of fire and explosion parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters. The following steps provide with an outline of the methodology for evaluating the fire and explosion accidents.

16.5.2.4.1 Define the Operating Environment

Verify that site characteristics chapter (for SLs), thermal chapter, and the materials chapter of the SAR identify the operating environment for a fire or explosion accident, including the following:

- the presence of materials that could accidentally burn or explode in the vicinity of the SNF storage facility or along the route of transfer at the site for DSFs; for DSSs, the presence of materials close to the DSS that could burn or explode (e.g., fuel tank of transporter moving the DSS to the storage pad) and other conditions that are reasonable to anticipate for sites that may use the DSS

- operational conditions that could accidentally initiate combustion or explosion

16.5.2.4.2 Define the Physical Parameters

Verify that the SAR defines the physical parameters associated with the accidents, including the following:

- the quantity of combustible fuel and materials present at the site for DSFs; for DSSs, the quantity of such materials assumed present and the basis for the assumptions

- the barriers in place to protect the SSCs from damage by heat or explosive overpressure

- the presence of a fire protection program **(SL)**

16.5.2.4.3 Review the Analysis Methodology

Verify that the SAR defines the methodology by which the fire or explosion hazards are to be evaluated, including the following:

- modeling techniques for calculating the temperature rise of SSCs

- assumptions and modeling techniques for predicting the structural response of SSCs to external or internal pressure

16.5.2.4.4 Accident Analysis

Verify that the SAR presents the accident analysis and design criteria and standards to do the following:

- Establish design criteria for temperature limits for temperature-sensitive materials and SSCs such as concrete, fuel cladding, shielding materials, and confinement boundary components subject to internal pressure rise or external pressure rise.

- Show that the maximum temperature resulting from the accidental fire does not reach the design limit and that the effect on the SSCs has been evaluated in the structural evaluation chapter.

- Show that the maximum internal pressure for a storage container is properly evaluated and verify that the maximum internal pressure of the storage container remains within its design pressures for accident conditions (assuming 100-percent fuel rod rupture with 100 percent of the initial fill gas and 30 percent of the fission product gas generated within the fuel rods during operation).

- Show that the maximum external pressure does not cause a breach of the confinement boundary and that the stress-intensity level is below the stress limit (i.e., ASME B&PV Code, Section III, Service Level D). Also consider the effect of confinement O-rings or seals and relevant pressure monitoring systems of bolted lid designs.

- Verify that a fire protection program provides assurance that a fire will not significantly increase the risk of radioactive releases to the environment; ensure that the fire protection program consists of fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel. **(SL)**

- Confirm that control room or control area ventilation system piping and instrumentation drawings show monitors located in the system intakes that can detect radiation, smoke, and toxic chemicals, if applicable. **(SL)**

- Confirm that monitors actuate alarms in the control room or other suitable locations, if applicable; consult RG 1.189, “Fire Protection for Nuclear Power Plants,” for detailed guidance. **(SL)**

- Verify that areas storing flammable, combustible, and hazardous materials are located and protected so that a fire, explosion, or release of hazardous materials will not adversely affect any SSCs important to safety. **(SL)**

- Verify that materials that collect and contain radioactive materials, such as spent ion exchange resins, charcoal filters, and high-efficiency particulate air filters, are stored in closed metal tanks located away from ignition sources and combustible material.

- Confirm that any accidental release together with direct radiation results in doses that do not exceed the limits in 10 CFR 72.106(b).

16.5.2.5 *Lightning*

Lightning is an event that the applicant must evaluate, in compliance with 10 CFR 72.122(b)(2)(i) and 10 CFR 72.236(b). The following steps provide an outline of the methodology for evaluating the lightning accident.

16.5.2.5.1 Define the Operating Environment and Physical Parameters

Verify that the SAR identifies the operating environment condition for a lightning strike, including the following:

- storage container SSCs that are exposed to possible lightning strikes
- other storage facility SSCs that are exposed to lightning strikes **(SL)**
- lightning protective devices included as a part of the design

16.5.2.5.2 Review the Analysis Methodology and Accident Analysis

Verify that the SAR presents an analysis or discussion of the effects of lightning strikes on all SSCs important to safety and, for DSFs, facility buildings, including the following:

- a discussion of structural materials or components, including monitoring or surveillance instrumentation and equipment, that might be damaged by heat or mechanical forces generated by passing current to ground

- any radiological consequences associated with the lightning strike

16.5.2.6 *Earthquake*

The earthquake accident is one of the accidents that the applicant must evaluate, in accordance with 10 CFR 72.122(b)(2)(i) and 10 CFR 72.236(b). Coordinate the evaluation of this topic in the SAR with the site characteristics evaluation under Chapter 2 of this SRP for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of earthquake or ground-motion parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.6.1 Define the Operating Environment

Determine the design ground motion according to the SAR. For SLs, refer to Chapter 2 of this SRP, which discusses this parameter, including the evaluation of the rationale for its selection.

Verify that the SAR has defined the configuration of the SSCs at the time of the seismic event (e.g., the container on the storage pad, the loaded transfer cask during transfer operations, the loaded transfer cask or the container suspended from a crane); the applicant should consider multiple configurations in the evaluation of seismic events and their impacts, including temporary expected configurations (e.g., construction activities to expand an operating array of storage containers that removes or exposes materials relied on for shielding by the operating storage containers).

16.5.2.6.2 Define the Physical Parameters

Determine which components of the DSS or DSF must be designed to withstand the effects of the design earthquake. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that nuclear power plant SSCs be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. RG 1.29, "Seismic Design Classification," describes a method for identifying those features of a light-water reactor that should be designed to withstand the effects of the safe-shutdown earthquake. The staff has interpreted this regulatory guide to mean that those SSCs identified as important to safety, and other SSCs that could affect SSCs important to safety, should be designed for the design-basis earthquake. Refer to Chapter 3 of this SRP for an evaluation of the identification of these components. Confirm that the applicant has identified protection devices to mitigate effects of the event, such as a seismic sensor to trip power to overhead cranes or extra seismic supports to be installed during transfer operations.

16.5.2.6.3 Review the Analysis Methodology

If the applicant uses an equivalent static load method, verify that the method produces conservative results and that the SSCs can be realistically represented by a simple model.

If the applicant uses a response spectrum analysis technique, verify that the response spectra meet the requirements in RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and that damping ratios are in accordance with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

If the applicant has performed a time-history analysis, verify that the time-history of acceleration is in compliance with American Society of Civil Engineers (ASCE) 4-98, "Seismic Analysis of Safety-Related Nuclear Structures."

16.5.2.6.4 Accident Analysis

Verify that the analysis has used the three components of earthquake motion and has combined them in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.7.2, "Seismic System Analysis," Subsection 6 and RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

In accordance with NUREG-0800, Section 3.7.2, Subsection 14, verify that the applicant has considered a determination of Category I structure overturning moments. To be acceptable, the determination of the design overturning moment should incorporate three components of input motion and conservative consideration of vertical and lateral seismic forces. Verify that the structure neither overturns nor slides because of the design earthquake.

Verify that the applicant has provided a summary of natural frequencies of the SSCs important to safety. If the applicant has used the direct integration method of analysis, verify that total responses of the SSCs have been calculated.

Verify that the applicant has identified any radiological consequences associated with the seismic event and calculated dose rates and doses. Although SSCs are not required to survive accident condition earthquakes without permanent deformation, verify that the stress intensities are less than the stress allowables (i.e., ASME B&PV Code, Section III, Service Level D).

16.5.2.7 *Loss of Shielding*

The applicant must evaluate the loss of shielding of any SSCs identified as important to safety to determine the dose to workers and the public. Loss of shielding can occur because of a variety events, such as a penetration of the concrete shielding induced by the impact of a tornado missile, the reduction in hydrogen content of neutron shielding by high-temperature exposure, loss of water or lowering of the water level by leakage from shields that are composed of water, or structural failure or melting of shielding by fire or explosion, and others. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.7.1 Define the Operating Environment and Physical Parameters

Verify that the SAR identifies the operating environment and the physical parameters of the accident, including the following:

- the operational configuration of the SCCs such as a container design that uses a liquid shielding material

- the design threshold for safety pressure-relief valves or rupture discs for liquid shield tanks

- relevant material specifications for shield materials (e.g., melting temperature, temperature of decomposition, mechanical strength)

16.5.2.7.2 Review the Analysis Methodology and Accident Analysis

Verify that the applicant has appropriately determined the maximum reduction of the radiation shielding thickness, material shielding effectiveness, or loss of temporary shielding in DSS or DSF SSCs and features as a result of postulated accidents such as tornado missiles, explosions, fires, liquid shield tank leaks, and container drop. Confirm that the applicant evaluated all possible shielding areas.

Verify that the applicant has performed a revised neutron and gamma dose rate shielding analysis with the accident-induced reduction or loss of shielding. The analysis should use computer codes and methodologies, as applicable, identical to those of the design shielding calculations for the DSS or DSF SSCs and features.

16.5.2.8 *Adiabatic Heatup*

Adiabatic heatup is a key assumption for an evaluated accident because it ensures that the applicant has evaluated the most conservative thermal transient response of the DSS or DSF SSCs. The transient temperature response of internal container components, including the contents, is solely a result of the decay heat of the contents and the individual container material heat capacity (i.e., mass and specific heat). The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.8.1 Define the Operating Environment

Verify that the SAR defines the ambient temperature, including insolation, used in the accident analysis. Verify that the applicant has defined the configuration of the SSCs (e.g., all inlets and outlets blocked for casks). Evaluate the highest design-basis decay heat load of the design, which should be stated in the principal design criteria chapter of the SAR.

16.5.2.8.2 Define the Physical Parameters

Verify the minimum mass of each material that constitutes a component of the DSS or DSF SSCs and features and the stored radioactive materials. Such materials are typically uranium dioxide, zircaloy, stainless steel, inconel, carbon steel, neutron absorber plates (e.g., boral, borated aluminum), (borated) resin, (borated) polyethylene, and concrete. In general, the mass can be calculated by determining the volume of the material and using a value for density of the material that is obtained from an established reference of material properties. The density should be appropriate for the anticipated temperature range for this calculation.

Determine the specific heat of each material from established references for the expected range of temperatures.

Determine the maximum short-term accident temperature limit of each material comprising DSS or DSF SSCs and features from established references.

16.5.2.8.3 Review the Analysis Methodology and Accident Analysis

Ensure that all containers that rely on natural air convection through internal labyrinthine passages assume that all air inlet and outlet passages are completely blocked. The thermal response must be calculated by assuming that no heat loss to the environment occurs. For example, for SNF casks having multiple air inlets and outlets; the staff has previously found it unacceptable to assume that one air outlet would become an air inlet while the other air outlets would continue to function as outlets. The staff has rejected this assumption because it has not been verified by experimental test data.

Calculate the sum of the product of mass and specific heat for each material. This is denoted as the heat capacity of the DSS or DSF SSCs.

Calculate the adiabatic heatup rate of the SSCs by dividing the total DSS or DSF storage container maximum decay heat load by the total SSC heat capacity.

Using the highest calculated temperature for each material at normal operating ambient temperatures, the maximum short-term accident temperature limit for each material, and the DSS or DSF SSC adiabatic heatup rate that was calculated in accordance with the above paragraph, determine the earliest time that a specific material temperature limit will be exceeded after the onset of an adiabatic heatup scenario.

Report, as the key result, the minimum time to reach the first material temperature limit during an adiabatic heatup event. The technical specifications must include a surveillance frequency. Ensure that the applicant provided a technical specification for any material that might exceed its temperature limit during an adiabatic heatup. See Chapter 5, “Thermal Evaluation,” of this SRP for more details.

Verify that the applicant has identified any radiological consequences associated with the adiabatic heatup and has calculated dose rates and doses.

16.5.2.9 *Tornadoes and Missiles Generated by Natural Phenomena*

The applicant must evaluate tornado and tornado-generated missile accidents, in accordance with 10 CFR 72.122(b)(2). Coordinate the evaluation of this material in the SAR with the site characteristics review based on Chapter 2 of this SRP for DSF specific license applications. For DSS applications, ensure that the SAR defines a set of tornado and missile parameters, the effects of which the DSS must withstand, and the basis for the selection of those parameters. The following steps provide an outline of the methodology that the applicant should provide in the SAR.

16.5.2.9.1 Define the Operating Environment and Physical Parameters

Review the SAR to determine the design wind and tornado wind velocities. Verify that the applicant analyzed design-basis tornado characteristics given in RG 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants."

Verify that the applicant used design-basis tornado missile spectra and maximum horizontal speeds from RG 1.76 in the analysis of missile impacts.

The NRC considers the missiles described in RG 1.76 capable of striking in all directions with horizontal velocities of V_{Mh}^{max} and vertical velocities equal to 67 percent of V_{Mh}^{max}. Barrier design should be evaluated assuming a normal impact to the surface for the Schedule 40 pipe and automobile missiles. The automobile missile is considered to impact at all altitudes less than 9.14 meters (30 feet) above all grade levels within 0.8 kilometer (0.5 mile) of the plant structures. Table 2 of RG 1.76 includes a different size and weight automobile for Region III than for Regions I and II (as defined in defined in RG 1.76). The heavier automobile used in the calculations for Regions I and II will have a lower kinetic energy in Region III. This effect is a consequence of the low maximum horizontal speed V_{Mh}^{max} of the heavier automobile in the Region III tornado wind field.

16.5.2.9.2 Review the Analysis Methodology and Accident Analysis

Verify the transformation of wind velocity into pressure. The NRC staff accepts the procedures used to transform the wind velocity into an effective pressure to be applied to structures and parts and portions of structures found in ASCE/Structural Engineering Institute (SEI) 7, "Minimum Design Loads for Buildings and Other Structures." These procedures specify that the maximum velocity pressure, *p* (in pounds per square foot), should be obtained from the formula, $p = 0.00256 V^2$, where *V* is in miles per hour; the velocity pressure should be assumed constant with height; and the maximum pressure applies at the radius of the tornado funnel at which the maximum velocity occurs. ASCE Paper No. 3269, "Wind Forces on Structures," issued in 1961, may be used to obtain the effective wind pressures for cases that ASCE/SEI 7 does not cover.

Verify that the applicant has analyzed all SSCs important to safety for damage from missiles that the design-basis tornado might generate (note: the design-basis tornado can vary depending on the location of the DSS or DSF). Also review the applicant's analysis of missile impact on SSCs important to safety. In previous submittals, the NRC has accepted the use of "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects"

(Kennedy 1975); “Design of Structures for Missile Impact” (Linderman et al. 1974); and “U.S. Reactor Containment Technology” (Cottrell and Savolainen 1965).

Verify that the applicant has calculated the most adverse combination of tornado wind, differential pressure, and missile load and used it in combination with other loads. To obtain the most adverse combination, the combinations should include wind alone, differential pressure alone, missile alone, wind plus half of the differential pressure, wind plus missile, and wind plus missile plus half of the differential pressure.

Verify that the applicant has identified any radiological consequences associated with the tornado and tornado-generated missiles and calculated the dose rates and doses.

16.5.2.10 *Accidents at Nearby Sites* ***(SL)***

Verify that the applicant has considered potential accidents at nearby sites and transportation routes. Reviews conducted under other sections of this SRP will have covered the procedures for reviewing these accidents (e.g., a natural gas explosion at a nearby site may result in an explosive overpressure and the effects of a fire at a nearby site). Verify that the effects of nearby site accidents have been encompassed by the effects of other accidents identified and evaluated in the SAR and reviewed as part of this SRP chapter.

Confirm that the SAR defines the analysis, design criteria, and design codes and standards (as applicable) for each off-normal and accident event as related to SNF, HLW, or reactor-related GTCC waste storage and handling systems.

16.5.2.11 *Structural Failures Resulting from Fire and Their Potential Impacts* ***(SL)***

Buildings must be designed to withstand collapse from the effects of flood, fire and explosion, lightning, earthquake, tornado and tornado-generated missiles, and accidents at nearby sites in accordance to their importance to safety or the potential impacts of their failures on SSCs important to safety. Other parts of Section 16.5 of this chapter present the review procedures for these events for SSCs important to safety. Verify that the applicant has analyzed the building structure to meet the applicable portions of these procedures. The applicant’s analysis should provide evidence that, although equipment or structures may be damaged, the surviving equipment and structures will continue to protect the SNF, HLW, and reactor-related GTCC waste and that the radiological consequences are within acceptable levels.

16.5.3 Other Non-Specified Off-Normal Events and Accidents

Evaluate other off-normal and accident scenarios included in the SAR but not identified in the previous subsections of this SRP. Coordinate the accident analysis review with the reviewers of all technical chapters of this SRP to verify that design and operations characteristics of the DSS or DSF do not pose potential off-normal events or accidents that the applicants have not identified or evaluated.

16.6 Evaluation Findings

The NRC reviewer should prepare evaluation findings on satisfaction of the regulatory requirements in Section 16.4 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:

F16.1 The SAR includes acceptable analyses of the design and performance of confinement and SSCs important to safety, and other SSCs that affect SSCs important to safety, under off-normal and accident scenarios to meet the requirements in 10 CFR 72.24 for a DSF or 10 CFR 72.236(c), (d), and (l) for a DSS. Applicable off-normal events analyzed in the SAR include [reviewer to select from the following:] partial vent blockage, operational events resulting in radioactive release [reviewer to list], off-normal ambient temperature scenarios, and [other off-normal events identified by the applicant or as part of the review]. Applicable accident events analyzed in the SAR include [reviewer to select from the following:] container tipover, container drop, flood, fire and explosion, lightning, earthquake, loss of shielding [if applicable], adiabatic heatup of the container, tornadoes and missiles generated by natural phenomena, accidents at nearby sites, building structural failure onto SSCs, and [other scenarios identified by the applicant or as part of the review].

F16.2 **(SL)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSF will acceptably meet the applicable regulatory requirements without endangering the public health and safety, in compliance with the overall requirements in 10 CFR 72.122.

F16.3 **(CoC)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS will facilitate meeting the applicable regulatory requirements without endangering the public health and safety, in compliance with the overall requirements in 10 CFR 72.122.

F16.4 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS or DSF will acceptably meet the requirements of 10 CFR 72.124, “Criteria for Nuclear Criticality Safety,” and, for DSSs, 10 CFR 72.236(c) regarding the maintenance of the SNF or HLW, or both, in a subcritical condition.

F16.5 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS or DSF will acceptably meet the requirements in (10 CFR 72.126, “Criteria for Radiological Protection,” (for DSFs) or 10 CFR 72.236(d) (for DSSs)) regarding criteria for radiological protection.

F16.6 **(SL)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSF will acceptably meet the requirements of 10 CFR 72.128 regarding handling and storage of the SNF and other radioactive material and confinement.

F16.7 **(CoC)** The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the DSS will facilitate meeting the requirements of

10 CFR 72.128 regarding handling and storage of the SNF and other radioactive material and confinement.

F16.8 No instruments or control systems are required to remain operational under accident conditions [as applicable] under 10 CFR 72.122(i).

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

> The staff concludes that the accident design criteria for the [DSS or DSF designation] are in compliance with 10 CFR Part 72, and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the DSS or DSF adequately demonstrates that it will provide for the safe storage of the stored radioactive materials during off-normal and accident conditions (for DSFs) or the accident conditions for which the DSS was designed (for DSSs) and during off-normal conditions (for which it was designed (for DSSs)). This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

16.7 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."

Air Movement and Control Association, Standards and Application Guides.

American Concrete Institute (ACI) 318, "Building Code Requirements for Structural Concrete and Commentary."

ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary."

American Iron and Steel Institute, "Steel Products Manual"

American National Standards Institute (ANSI) N14.6, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," Institute for Nuclear Materials Management.

ANSI/American Nuclear Society (ANS) 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants."

ANSI/ANS 57.7, "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)."

ANSI/ANS 57.9-1992, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type),” 1992 (reaffirmed 2000).

American Society of Civil Engineers (ASCE) 4-98, “Seismic Analysis of Safety-Related Nuclear Structures.”

ASCE/Structural Engineering Institute 7, “Minimum Design Loads for Buildings and Other Structures.”

ASCE Paper No. 3269, “Wind Forces on Structures,” *Transactions*, 126(Part II), pp. 1124–1198, 1961.

American Society of Heating, Refrigeration and Air-Conditioning Engineers, “ASHRAE Handbook—Fundamentals.”

American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code.
Section III, “Rules for Construction of Nuclear Facility Components.”

ASME B30.2, “Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist).”

ASME B30.16, “Overhead Hoists (Underhung).”

ASME N509, “Nuclear Power Plant Air-Cleaning Units and Components.”

Cottrell, W.B. and A.W. Savolainen, “U.S. Reactor Containment Technology,” in ORNL-NSIC-5, Volume 1, Chapter 6, Oak Ridge National Laboratory, August 1965.

Crane Manufacturers Association of America Specification No. 70, “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes.”

Institute of Electrical and Electronics Engineers (IEEE) C2, “National Electrical Safety Code.”

IEEE 835, “Standard Power Cable Ampacity Tables.”

International Code Council (ICC), “International Building Code.”

ICC, “International Mechanical Code.”

Kennedy, R.P., “A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects,” in ORNL-NSIC-5, Volume 1, Chapter 6, Holmes and Narver, Inc., September 1975.

Linderman, R.B., J.V. Rotz, and G.C.K. Yeh, “Design of Structures for Missile Impact,” Topical Report BC-TOP-9-A, Revision 2, Bechtel Power Corporation, September 1974.

National Fire Protection Association (NFPA) 70, “National Electrical Code.”

NFPA 780, “Standard for the Installation of Lightning Protection Systems.”

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

Regulatory Guide 1.29, "Seismic Design Classification."

Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants."

Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

Regulatory Guide 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007.

Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants."