U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.0 INTRODUCTION - TRANSIENT AND ACCIDENT ANALYSES

REVIEW RESPONSIBILITIES

Primary - Organizations responsible for review of transient and accident analyses

Secondary - None

The evaluation of the safety of a nuclear power plant requires analyses of the plant's responses to postulated equipment failures or malfunctions. Such analyses help to determine the limiting conditions for operation, limiting safety system settings, and design specifications for safety-related components and systems to protect public health and safety. These analyses are a focal point of the license amendment request (LAR), design certification (DC), and combined license (COL) reviews.

I. AREAS OF REVIEW

The transients and accidents reviewed under Chapter 15 are the design basis events for which the initiating event is assumed to be a single failure of a system or component to perform its intended safety functions. (Multiple failures resulting from a single failure is considered to be a single failure.) Severe accidents or initiating events with multiple failures are the subjects of probabilistic risk assessment reviewed under Chapter 19.

The specific areas of review are as follows:

A. Categorization of Transients and Accidents.

The reviewer ensures that the applicant's selection and assembly of the plant transient and accident analyses represent a sufficiently broad spectrum of transients and accidents, or initiating events.

Initiating events are categorized according to expected frequency of occurrence and by type. Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. Categorization of initiating events by type provides a basis for comparison between events, which makes it possible to identify and evaluate the limiting cases (i.e., the cases that can challenge the analysis acceptance criteria).

B. Categorization According to Frequency of Occurrence: Each initiating event is categorized as either an anticipated operational occurrence (AOO); postulated accident, which includes the infrequent event (IE) classification; or special event.

AOOs, as defined in Appendix A to Title 10, *Code of Federal Regulations* (10CFR),

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Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

The following are some examples of AOOs in pressurized-water reactor (PWR) designs:

- Inadvertent control rod or rod group withdrawal
- Loss or interruption of core coolant flow
- Inadvertent moderator cooldown
- Inadvertent chemical shim control
- Depressurization by spurious operation of an active element, such as a relief valve
- Blowdown of reactor coolant through a safety valve
- Loss of normal feedwater
- Loss of condenser cooling
- Steam generator tube leaks
- Reactor-turbine load mismatch, including loss of load and turbine trip
- Control rod drop (inadvertent addition of absorber)
- Single error of an operator
- Single failure of a control component
- Single failure in the electrical system
- Minor reactor coolant system (RCS) leak or loss of reactor coolant such as from a small ruptured pipe or from a crack in a large pipe
- Minor secondary system break
- Loss of offsite power
- Operation with a fuel assembly in an improper position
- Loss of feedwater heating
- Reactor overpressure with delayed scram

The individual event sections of the design specific review standard (DSRS) address specific AOOs and their appropriate variations.

Postulated accidents and infrequent events, which are a subset of accidents, are unanticipated occurrences that are postulated but not expected to occur during the life of the nuclear power plantThe following are some examples of postulated accidents in PWRs of current designs:

- Major rupture of a pipe containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant pressure boundary
- Ejection of a control rod assembly
- Major secondary system pipe rupture up to and including double-ended rupture

The sections of the DSRS dealing with the individual events address specific postulated accidents and appropriate variations (e.g., design-specific variations).

B. <u>Categorization According to Type</u>. AOOs, infrequent events and postulated accidents are also categorized according to type. The type of AOO or postulated accident, including IEs, is defined by its effect on the plant. For example, one type of AOO or postulated accident will cause the RCS to pressurize and possibly jeopardize RCS integrity. Another type will cause the RCS to depressurize and possibly jeopardize fuel cladding integrity. It is useful to categorize and organize analyses of AOOs and postulated accidents according to type, so that analysts can compare them on common bases, effects, and safety limits. Such comparisons can help to identify limiting events and cases for detailed examination and eliminate nonlimiting cases from further consideration.

AOOs and postulated accidents, including IEs, can be grouped into the following seven types:

i. Increase in heat removal by the secondary system
ii. Decrease in heat removal by the secondary system
iii. Decrease in RCS flow rate
iv. Reactivity and power distribution anomalies
v. Increase in reactor coolant inventory
vi. Decrease in reactor coolant inventory
vii. Radioactive release from a subsystem or component

The review of AOOs and postulated accident analyses within a type can (and should) encompass a variety of cases, each designed to produce effects or results that challenge designated safety limits. For example, one case study of the turbine trip event, an AOO that causes a decrease in heat removal by the secondary system, can be designed to yield a high peak RCS pressure, and another case study of the same AOO can be designed to yield a low, minimum thermal margin. The former case tests the safety limit for RCS pressure boundary integrity, while the latter case tests the safety limit that protects fuel cladding integrity.

The reviewer considers the possible case variations of AOOs and postulated accidents presented to verify that the licensee has identified the limiting cases. The reviewer evaluates licensees' claims that individual AOOs and postulated

accidents, including IEs, are limiting or nonlimiting, or bounded by other AOOs, IEs and postulated accidents, with particular attention to the bases used for comparison. Comparison of AOOs to other AOOs within a type, for example, is easily justified. Comparison of AOOs of one type to IEs or postulated accidents of another type requires closer scrutiny and more justification from the licensee.

There are also special events such as anticipated transient without scram (ATWS) and station blackout (SBO) which are addressed in DSRS Sections 15.8 and 8.4. Analysis acceptance criteria for special events are included in their respective DSRS sections.

Anticipated transients without scram (ATWSs) are AOOs in which a reactor scram is demanded but fails to occur because of a common-mode failure in the reactor scram system. ATWS events, therefore, are AOOs that postulate complete failure of the required (single-failure proof) protection system. As such, they are beyond the design basis, and consequently, ATWS events are addressed separately (see DSRS Section 15.8).

2. Analysis Acceptance Criteria. If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs, IEs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs, IEs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences.

   The reviewer will consider the results of licensees' analyses and evaluations of individual initiating events to ascertain whether the licensee has satisfied the applicable analysis acceptance criteria for each of the events. The licensee may propose the use of alternate acceptance criteria appropriate to the particular plant design and operation (e.g., new reactor design applications). In such cases, the reviewer will consider the alternate criteria and determine whether they are equivalent, in function and consequences, to the current criteria (see below).

   A. Analysis Acceptance Criteria for AOOs. The following are the specific criteria necessary to meet the requirements of GDC for AOOs:

      i. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

      ii. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

      The reviewer applies a third criterion, based on the ANS standards to ensure that there is no possibility of initiating a postulated accident with the frequency of occurrence of an AOO.

      iii. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

B. <u>Analysis Acceptance Criteria for IEs and Postulated Accidents</u>. Unlike an AOO, an IE or a postulated accident could result in sufficient damage to preclude resumption of plant operation. A list of the basic criteria necessary to meet the requirements of GDC for postulated accidents appears below. Individual sections of the DSRS may specify additional criteria pertaining to certain IEs or postulated accidents.

i. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

ii. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed.

iii. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Parts 52.47(a)(2)(iv) and 100. The acceptance criterion for IEs is small fraction (10%) of 10 CFR 52.47 (a) and Part 100.

iv. A postulated accident, including an IE, shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For loss-of-coolant accidents (LOCAs), the following analysis acceptance criteria of 10 CFR 50.46 also applies:

i. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

ii. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

iii. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

iv. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

v. After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature should be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

3. <u>Plant Characteristics Considered in the Safety Evaluation</u>. The reviewer ensures that the application contains the key plant parameters considered in the safety evaluation (e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics). The reviewer checks that the range of values for plant parameters is representative of fuel exposure or core reload, and that the range is sufficiently broad to cover the predicted fuel cycle ranges, to the extent practicable, based on the fuel design and acceptable analytical methodology at the time of the LAR, DC, or COL application. The reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.

4. <u>Assumed Protection and Safety Systems Actions</u>. The reviewer ensures that the application lists the settings of all the protection and safety systems functions that are used (i.e., credited) in the safety evaluation. Typical protection and safety systems functions include reactor trips, isolation valve closures, ECCS initiation and ECCS. In evaluations of AOOs and postulated accidents, the performance of each credited protection or safety system is required to include the effects of the most limiting single active failure. This verifies satisfaction of the GDC criteria that require protection and safety systems to adequately perform their intended safety functions in the presence of single active failures. The reviewer also ascertains that the application lists the expected limiting delay time for each protection or safety system function and describes the acceptable methodology for determining uncertainties (from the combined effects of calibration error, drift, instrumentation error, and other factors) to be included in the establishment of the trip setpoints and allowable values specified in the plant technical specifications.

5. <u>Evaluation of Individual Initiating Events</u>. The reviewer ensures that the application includes an evaluation of each initiating event, using the format in Subsection I.6 of this DSRS section. For initiating events that are determined to be not limiting, the reviewer may evaluate qualitative justifications and conduct comparisons with the corresponding, more limiting initiating events.

6. <u>Event Evaluation</u>

    A. <u>Identification of Causes and Frequency Classification</u>. For each initiating event evaluated, the reviewer ensures that the application includes a description of the occurrences that can lead to the event and a categorization of the event as either an AOO or postulated accident including IEs. The reviewer also checks for clear definitions of the analysis acceptance criteria appropriate to the specific nature of the initiating event, as well as the event's categorization.

    B. <u>Sequence of Events and Systems Operation</u>. The reviewer verifies that the application addresses the following considerations for each initiating event:

        i. Step-by-step sequence of events, from event initiation to the final stabilized condition (i.e., identification on a time scale of each significant occurrence, including flux monitor trips, insertion of control rods, attainment of primary coolant safety valve set points, opening and closing of safety valves, generation of containment isolation signals, and containment isolation) and

identification of all operator actions credited in the transient and accident analyses for consequence mitigation

ii. Extent to which normally operating plant instrumentation and controls are assumed to function

iii. Extent to which plant and reactor protection systems are required to function

iv. Credit taken for the functioning of normally operating plant systems

v. Credited operation of engineered safety systems

vi. Assurance of consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines or emergency operating procedures with respect to the operator response (including action time) and available instrumentation

The reviewer verifies that the applicant has specified only safety-related systems or components for use in mitigating AOO and postulated accident conditions, and has included the effects of single active failures in those systems and components. If nonsafety-related systems are operational during a response to an initiating event and negatively impact the response, then applicant would be expected to include this in analysis. The reviewer may consider the licensee's technical justifications for the operation of nonsafety-related systems or components (e.g., when they are used as backup protection and when they are not disabled, except by a detectable, random, and independent failure).

The reviewer ascertains that the applicant has evaluated the effects of single active failures and operator errors and that the licensee's application contains sufficient detail to permit independent evaluation of the adequacy of systems, as they relate to the subject events.

C. Core, System, and Barrier Performance

i. Evaluation Model. The reviewer ensures that the applicant has discussed the evaluation model used and any simplifications or approximations introduced to perform the analyses and identified digital computer codes used in the analysis. If the analysis uses more than one computer code, the applicant should describe the method used to connect the codes. The reviewer verifies that the applicant has discussed the important output of the codes (see the Results section below) with emphasis on the input data and the extent or range of variables investigated and that the applicant has included detailed descriptions of evaluation models and digital computer codes or listings by referencing documents that are available to the NRC.

The reviewer ensures that the applicant has provided a table listing the titles of topical reports (TRs) that describe models or computer codes used in transient and accident analyses and listed the associated NRC safety evaluation reports approving those TRs. The reviewer checks that implementations of NRC-approved models or codes are within the

applicable ranges and conditions and that the applicant has demonstrated compliance with each of the conditions and limitations imposed by the NRC staff in its safety evaluation reports that approve the TRs.

ii. Input Parameters and Initial Conditions. The reviewer verifies that the applicant has (1) identified the major input parameters and initial conditions used in the analyses; (2) included the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study; ; and (3) discussed the bases for the numerical values of the input parameters and initial conditions used in the analyses (including the degree of conservatism).

iii. Results. The reviewer ensures that the applicant has presented the results of the analyses, including key parameters as a function of time during the course of the transient or accident. The following are examples of parameters that should be included:

- Neutron power
- Thermal power
- Heat fluxes, average and maximum
- RCS pressure
- Departure from nucleate boiling ratio (DNBR)
- Coolant conditions, including inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam volume fractions
- Temperatures, including maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy
- Reactor coolant inventory, including total inventory and coolant level in various locations in the RCS
- Secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory
- ECCS flow rates and pressure differentials across the core, as applicable
- Containment pressure
- Relief and/or safety valve flow rate
- Reactor recirculation valves
- Flow rate from the RCS to the containment system, if applicable

- Pressurizer water volume

In addition, the discussion of the results should emphasize the margins between the predicted values of various core parameters, as well as the values of those parameters that would represent limiting acceptable conditions.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. Design basis radiological consequence analyses associated with design basis accidents are reviewed under DSRS Section 15.0.3

2. Determination of any non-safety related equipment whose reliability contributes to the event frequency in the probabilistic risk assessment is reviewed under DSRS Chapter 19.

3. The NuScale design allows for multiple reactor modules at a single plant site. The design basis analyses in Chapter 15 assumes the operation of only one reactor module that is sufficiently isolated to satisfy GDC 5. For plant sites with a common control room or a single operator operating multiple reactor modules, potential common mode failures affecting multiple modules are reviewed under SRP Section 19, and issues related to human factors are reviewed under SRP Section 18.

II. ACCEPTANCE CRITERIA

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR Part 20, "Standards for Protection Against Radiation "

2. 10 CFR Part 50,"Domestic Licensing of Production and Utilization Facilities "(especially 10 CFR 50.46 and Appendix A)

3. 10 CFR Part 100,"Reactor Site Criteria "

4. 10 CFR Part 52,"Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants "

The following GDCs from Appendix A to 10 CFR Part 50 are relevant to DSRS Section 15:

1. GDC 2 Design Bases for Protection Against Natural Phenomena.,

2. GDC 4 Environmental and Dynamic Effects Design Bases.,

3. GDC 5 Sharing of Structures, Systems, and Components.,

4. GDC 10 Reactor Design.,

5. GDC 13 Instrumentation and Control

6. GDC 15 Reactor Coolant System Design

7. GDC 17 Electric Power Systems.

8. GDC 19 Control Room.

9. GDC 20 Protection System Functions

10. GDC 25 Protection System Requirements for Reactivity Control Malfunctions

11. GDC 26 Reactivity Control System Redundancy and Capability.

12. GDC 27 Combined Reactivity Control Systems Capability. and 28 Reactivity Limits.

13. GDC 29 Protection Against Anticipated Operational Occurrences.

14. GDC 31 Fracture Prevention of Reactor Coolant Pressure Boundary.

15. GDC 34 Residual Heat Removal.

16. GDC 35 Emergency Core Cooling.

17. GDC 55 Reactor Coolant Pressure Boundary Penetrating Containment.

18. GDC 60 Control of Releases of Radioactive Materials to the Environment.

19. GDC 61 Fuel Storage and Handling and Radioactivity Control.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

III. REVIEW PROCEDURES

The review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection

II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework. Using the graded approach described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

- 10 CFR Part 50, Appendix A, General Design Criteria (GDC), Overall Requirements, Criteria 1 through 5
- 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
- 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
- 10 CFR 50.55a, Code Design, Inservice Inspection and Inservice Testing (ISI/IST) Programs
- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, Technical Specifications
- Availability Controls for SSCs Subject to Regulatory Treatment of Non-Safety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all-inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37), for design certification or combined license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v) for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

To evaluate the LAR, DC, or COL application, the reviewer verifies that the applicant has performed the applicable transient and accident analyses needed to demonstrate conformance to the regulations.

DSRS Chapter 15 subsections discuss specific review procedures for transients or accidents.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

DSRS Chapter 15 subsections discuss the statements and conclusions of evaluation findings for transients or accidents.

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the Standard Review Plan (SRP) revision in effect six months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed small modular reactor (SMR) designs, however, differ significantly from large light-water nuclear reactor power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405) (SRM). In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated pre-application activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for the evaluation of a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale

application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design or siting assumptions.

VI. DEFINITIONS

Term	Definition
anticipated operational occurrences (AOOs)	Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.
anticipated transient without scram (ATWS)	AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20, because of common-mode failure.
common-mode failure	The result of an event which, because of dependencies, causes a coincidence of failure states of components in two or more separate channels of a redundancy system, leading to the failure of the defined system to perform its intended function.
departure from nucleate boiling (DNB)	The DNB acceptance criterion for an AOO is met when there is a 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that DNB will not occur, and the fuel centerline temperature stays below the melting temperature.
departure from nucleate boiling ratio (DNBR)	The ratio of the heat flux needed to cause departure from nucleate boiling to the actual local heat flux of a fuel rod.
design basis	Information that identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted state of the art practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.
design-basis accidents	Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components.

design-basis events (DBEs)	Conditions of normal operation, including AOOs, infrequent events, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures
general design criteria (GDC)	Reference 1 lists the GDC. The GDC that mention AOOs are 10, 13, 15, 17, 20, 26, 29, 60, and 64. The GDC that mention postulated accidents are 4, 16, 17, 22, 27, 28, 31, 41, 51, 61, and 64.
Infrequent Event (IE)	A subcategory of postulated accidents with limited fuel failure and not expected to occur in the lifetime of the plant.
loss-of-coolant accident (LOCA)	A postulated accident that results in the loss of reactor coolant at a rate in excess of the replacement capability of the reactor coolant makeup system.
overpressurization	The condition that occurs when pressure exceeds the design pressure of the component of interest by more than 10 percent, in accordance with the ASME Code.
postulated accidents	Unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit).
protection system	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. (GDC 20)
single failure	An occurrence that results in a component's loss of capability to perform its intended safety functions.

VII. REFERENCES

1. U.S. Nuclear Regulatory Commission, “Final Safety Evaltuation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design,” NUREG-1966, April 2014, ADAMS Assession No. ML14100A304

2. U.S. Nuclear Regulatory Commission, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” Final Policy Statements, Published and Effective August 16, 1995, 60FR42622.

3. Appendix A to 10 CFR Part 50, “General Design Criteria for Nuclear Plants.”

4. Regulatory Guide 1.206 “Combined License Applications for Nuclear Power Plants (LWR Edition).”

5. American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code*, Section III, “Nuclear Power Plant Components,” Article NB-7000, “Protection Against Overpressure,”.

6. U.S. Nuclear Regulatory Commision, “Single-Failure Criterion,” SECY-77-439, August 1977, ADAMS Accession No. ML060260236.

7. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

8. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”