U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

The loss of nonemergency alternating current (ac) power is assumed to result in the loss of all power to the station auxiliaries. This situation, which could be the result of a complete loss of either the external (offsite) grid or the onsite ac distribution system, is different from the loss of load condition considered in Design-Specific Review Standard (DSRS) Section 15.2.1-15.2.5 because, in the latter case, ac power remains available to operate the station auxiliaries. A major difference between the NuScale small modular reactor (SMR) and large light pressurized-water reactors (PWRs) is that there are no reactor recirculation pumps; the primary system flow is driven by natural circulation.

The Decay Heat Removal System (DHRS) consists of two natural convection-driven heat exchanger trains. Each train comprises a loop that includes a steam generator and a decay heat removal heat exchanger (DHR HX). Primary system water circulates by natural convection through the steam generators within the reactor vessel and transfers energy to the secondary side of the steam generator. The secondary coolant circulates through the heat transfer loop that includes the steam generator and DHR HX by natural convection, and transfers energy to the reactor pool through the DHR HX.

Within a few seconds, the turbine trips, the reactor coolant system is isolated, and the pressure and temperature of the coolant increase. A reactor trip is initiated. The sensible and decay heat loads are handled by actuation of the steam relief system, steam bypass to the condenser, and passive DHRS.

The loss of ac power has the following effects: (1) an immediate load rejection with fast closure of the turbine control valves, (2) loss of power to the condensate and feedwater pumps, resulting in loss of feedwater, and (3) the reactor is isolated after loss of main condenser vacuum. Therefore, the review of the loss-of-ac-power transient includes the sequence of events, the analytical model, the values of parameters in the analytical model, and the predicted consequences of the transient. The specific areas of review are as follows:

1. The sequence of events described in the applicant's technical submittal is reviewed with concentration on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

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2. The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been reviewed previously and accepted by the staff. If a referenced analytical method has not been reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by appropriate reviewers.

3. The predicted results of the transient analysis are reviewed for whether the consequences meet the acceptance criteria of subsection II of this DSRS section. The results of the analysis are reviewed for whether the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

4. Combined Operating License (COL) Action Items and Certification Requirements and Restrictions. For a Design Certification (DC) application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. General information on transient and accident analyses is provided in DSRS Section 15.0.

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

3. Aspects of the sequence described in the applicant's technical submittal are reviewed for whether the reactor and plant protection, and safeguards controls and instrumentation systems function as assumed in the safety analysis for automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems under DSRS Chapter 7.

4. Technical specifications are reviewed under DSRS Section 16.0.

5. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the applicant's technical submittal analysis are reviewed under SRP Sections 4.2, 4.3, and DSRS Section 4.4.

6. The determination of the safety-related and risk significance of structures, systems, and components (SSCs) relied upon to meet required functions during the accidents are based on the review of the probabilistic risk analysis under SRP Chapter 19.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 10, "Reactor Design," as to reactor coolant system (RCS) design with appropriate margin so specified acceptable fuel design limits are not exceeded during normal operation including anticipated operational occurrences.

2. GDC 13, "Instrumentation and Control," as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, "Reactor Coolant System Design," as to design of the RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operation including anticipated operational occurrences.

4. GDC 26, "Reactivity Control System Redundancy and Capability," as to reliable control of reactivity changes so specified acceptable fuel design limits are not exceeded in anticipated operational occurrences. This control is accomplished by appropriate margin for malfunctions like stuck rods.

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 (for 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.

Specific criteria necessary to meet the relevant requirements of GDCs 10, 13, 15, and 26 for events of moderate frequency (see definitions of design and plant process conditions in are as follow:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity should be maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. For the requirements of GDCs 10 and 15, the positions of Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Systems," affect the plant response to the type of transient addressed in this DSRS section.

5. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

   The applicant's analysis of the loss-of-ac-power transient should be based on an acceptable and NRC-approved model. If the applicant proposes analytical methods not approved, these are evaluated by the staff for acceptability and approval. For new generic methods, the reviewer requests an appropriate evaluation.

   The parameter values in the analytical model should be suitably conservative. The following values are acceptable:

   A. The initial power level is taken as the licensed core thermal power plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level.

   B. Conservative scram characteristics are assumed (i.e., for the maximum time delay with the most reactive rod held out of the core).

   C. The core burn-up is selected to yield the most limiting combination of moderator temperature reactivity feedback, void reactivity feedback, Doppler reactivity feedback, power profile, and radial power distribution.

   D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105. Compliance with RG 1.105 is determined.

Programmatic Requirements: The NRC regulations require that each operating license contain a technical specification (TS) that defines "… the limits, operating conditions, and other requirements imposed upon facility operation for the protection of public health and safety…" The licensee's analysis of DSRS Section 15.2.6 must be consistent with the information presented in the licensee's TS.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. GDC 10 requires that the reactor core and its coolant, control, and protection systems be designed with appropriate margin so specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 10 applies to this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This anticipated operational occurrence creates a potential for specified acceptable fuel design limits to be exceeded. Within seconds after the loss of power, the turbine and the reactor both trip, and the pressure and temperature of the reactor coolant increase. RG 1.53 provides guidance for application of the single-failure criterion to the design and analysis of nuclear power plant protection systems. RG 1.105 describes a method acceptable to the staff for keeping instrument setpoints within the technical specification limits.

GDC 10 requirements provide assurance that specified acceptable fuel design limits are not exceeded and that fuel cladding integrity is maintained in loss of nonemergency ac power to the station auxiliaries.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. GDC 15 requires that the RCS and its auxiliary, control, and protection systems be designed with sufficient margin so design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. GDC 15 applies to this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This transient is an anticipated operational occurrence, and the reactor coolant pressure must be analyzed to confirm whether the pressure acceptance criterion is satisfied.

   GDC 15 requirements provide assurance that the design conditions of the reactor coolant pressure boundary are not exceeded in the loss of nonemergency ac power to the station auxiliaries.

4. GDC 26 requires that one of the reactivity control systems consist of control rods capable of reliably controlling reactivity changes with appropriate margin for malfunctions like stuck rods so that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences.

   GDC 26 applies because the transient analyzed in this section involves the movement of control rods in response to the loss of nonemergency ac power and because rod misalignment, including stuck rods, can produce thermal-hydraulic conditions more severe than otherwise. GDC 26 requires a thermal margin sufficient to accommodate these conditions. Under DSRS Section 15.2.6 these margins are examined where applicable for whether the thermal criteria remain satisfied.

GDC 26 requirements provide assurance by appropriate margin for malfunctions of the reactivity control system, including stuck rods, that specified acceptable fuel design limits are not exceeded.

III. REVIEW PROCEDURES

These 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: Light-Water Small Modular 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 small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain 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 completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

   - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
   - 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 Nonsafety 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 DC or COL applications submitted under 10 CFR 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, "Resolution of Generic Safety Issues," 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.

The procedures below are used for the review of DC application review construction permit (CP), COL, and operating license (OL) applications. During the CP review the values of system parameters and setpoints in the analysis are preliminary in nature and subject to change. At the COL or OL review stage, final values should be in the analysis and the reviewer should compare these to the limiting safety system settings in the proposed technical specifications.

1. The description of the loss-of-ac-power transient presented by the applicant in the applicant's technical submittal is reviewed by the organization responsible for review of transients and accidents analyses regarding the occurrences leading to the initiating event. The sequence of events from initiation until condition stabilization is reviewed to ascertain:

   A. the extent to which normally operating plant instrumentation and controls are assumed to function

   B. the extent to which plant and reactor protection systems are required to function

   C. the extent to which credit is taken for the functioning of normally operating plant systems

   D. the extent to which operation of engineered safety systems is required

   E. the extent to which operator actions are required

   F. the extent to which operation of standby diesel generators is required

G. whether the description accounts for appropriate margin for malfunctions like stuck rods (per subsection II.B of this DSRS section)

2. If the applicant's technical submittal states that the loss-of-ac-power transient is not as limiting as some other similar transient, the reviewer evaluates the applicant's justification. If the SAR presents a quantitative analysis of the loss-of-ac-power transient, the timing of the initiation of those protection, engineered safety, standby diesel generator, and other systems needed to limit transient consequences acceptably level is reviewed. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints. The review of applicant's technical submittal Chapter 7 confirms whether the instrumentation and control systems design is consistent with the requirements for safety system actions for these events. To the extent necessary, the reviewer evaluates the effects of single, active system and component failures which may affect the course of the transient. This aspect of the review uses the procedures described in DSRS sections for applicant's technical submittal Chapters 4, 5, 6, 7, 8, and 9.

3. The applicant's mathematical models for evaluating core performance and predicting system pressure in the RCS and main steam lines are reviewed by for whether these models have been reviewed and accepted by the staff. If not, the reviewer initiates a generic review of the applicant's proposed model.

4. System parameter values and initial core and system conditions as input to the model are reviewed. Of particular importance are the reactivity feedbacks and control rod worths in the applicant's analysis and the variations of moderator temperature, void, and Doppler reactivity feedback with core life. The applicant's justification to show that it has selected the core burn-up that yields minimum margins is evaluated. The values of the reactivity parameters in the applicant's analysis are reviewed.

5. The results of the analysis are reviewed and compared to the acceptance criteria of subsection II of this DSRS section as to the maximum pressure in the reactor coolant and main steam systems. The variations during the transient of neutron power, heat fluxes (average and maximum), RCS pressure, minimum DNBR, coolant conditions (inlet temperature, core average temperature , average exit and hot channel exit temperatures), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the RCS to the containment system (if applicable) are reviewed. Time-related variations of the following parameters are reviewed:

    A. reactor power

    B. heat fluxes (average and maximum)

    C. RCS pressure

    D. minimum DNBR

    E. core flow rates

F. coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions)

G. steam line pressure

H. containment pressure

I. pressure relief valve flow rate

J. flow rate from the RCS to the containment system (if applicable)

The more important of these parameters for the loss-of-ac-power transient are compared to those predicted for other similar plants to confirm that they are within the expected range.

6. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the applicant's technical submittal meets the acceptance criteria. DCs have referred to the applicant's technical submittal as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC applicant's technical submittal.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) 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.

The staff concludes that the plant design as to transients expected to occur with moderate frequency and to result in the loss of all power to the station auxiliaries is acceptable and meets the relevant requirements of GDCs 10, 13, 15, and 26 and the applicable Three Mile Island Action Plan items. This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10 and 26 by demonstrating that resultant fuel integrity is maintained because the specified acceptable fuel design limits were not exceeded for the event.

2. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

3. The applicant meets GDC 15 requirements by demonstrating that the reactor coolant pressure boundary limits were not exceeded by this event and that resultant leakage is within acceptable limits. This requirement is met because the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressure.

4. The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during this event with appropriate margin for stuck rods because the specified acceptable fuel design limits were not exceeded.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

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 6 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 SMR designs, however, differ significantly from large light-water nuclear 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 Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010. 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 preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating 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 to address new design or siting assumptions.

VI. REFERENCES

1. *U.S. Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy,” Appendix A, GDC 10, “Reactor Design.”

2. *U.S. Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy,” Appendix A, GDC 13, “Instrumentation and Control.”

3. *U.S. Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy,” Appendix A, GDC 15, “Reactor Coolant System Design.”

4. *U.S. Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy,” Appendix A, GDC 26, “Reactivity Control System Redundancy and Capability.”

5. U.S. Nuclear Regulatory Commission, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems,” Regulatory Guide (RG) 1.53, Revision 2, November 2003, Agencywide Documents Access and Management System (ADAMS) Accession No. ML033220006.

6. U.S. Nuclear Regulatory Commission, “Setpoints for Safety-Related Systems,” RG 1.105, Revision 3, December 1999, ADAMS Accession No. ML993560062.

7. U.S. Nuclear Regulatory Commission, “NRC Action Plan Developed as a Result of the TMI-2 Accident,” NUREG-0660, May 1980, ADAMS Accession Nos. ML072470526 and ML072470524.

8. U.S. Nuclear Regulatory Commission, “Clarification of TMI Action Plan Requirements,” NUREG-0737, November 1908, ADAMS Accession No. ML051400209.