U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

15.6.6 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM (ECCS)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

An accidental reactor vessel depressurization and decrease of reactor vessel coolant inventory may occur due to the inadvertent operation of the emergency core cooling system (ECCS), which could be caused by a spurious electrical signal, hardware malfunction, or operator error. As this event can occur one or more times during the plant's lifetime, it is an anticipated operational occurrence (AOO), as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 10 CFR Part 50, Appendix A. The event covered in this design-specific review standard (DSRS) section should be addressed in an individual section of the applicant's technical submittal as specified in Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

1. The review of these transients should consider 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:

   The staff reviews the sequence of events described in the applicant's technical submittal. The reviewer focuses on the need for the reactor protection system (RPS), the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

   The analytical methods are reviewed to determine whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced method has not been reviewed, the reviewer initiates a generic evaluation. The values of all parameters in the new analytical model, including the initial conditions of the core and system, are reviewed. The predicted results of the transient are reviewed to determine whether the consequences meet the acceptance criteria of DSRS Section 15.0 and Subsection II of this DSRS section. The analysis results are reviewed to determine whether pertinent system parameter values are within ranges expected for the type and class of reactor under review.

   Reactor protection system functions (e.g., automatic reactor trips) that are identified as available protection for this event, other than the credited function, are evaluated to

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ascertain whether the specified functions would be effective (i.e., setpoints and response times would lead to timely action, to satisfy the acceptance criteria). For example, the credited protection function might be the low pressurizer pressure reactor trips.

The reviewer verifies whether the applicant's core physics data are appropriate.

2. Combined 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 and Standard Review Plan (SRP) 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 in DSRS Section 15.0.3.

3. Verification of the safety analysis sequence described by the applicant for automatic actuation, remote sensing, indication, control, interlocks with auxiliary or shared systems, potential bypass modes, and the possibility of manual control by the operator is performed under DSRS Chapter 7.

4. Verification of whether the equipment necessary to mitigate the event is qualified for the transient and posttransient environments and identification, if requested, of equipment that the failure of which as a result of the initiating event could have adverse consequences are performed under applicable DSRS sections.

5. The reviewer verifies whether the control systems power sources needed to mitigate the event are available, as required by the applicant's description of the event, under DSRS Chapter 8.

6. The determination of the risk significance of SSCs relied upon to meet required functions during the accidents is based on the review of the probabilistic risk analysis in SRP Chapter 19.

7. Containment response and associated acceptance criteria are reviewed under DSRS Section 6.2.1.

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

2. GDC 13, Instrumentation and Control

3. GDC 15, Reactor Coolant System Design

4. GDC 20, Protection System Functions

5. GDC 26, Reactivity Control System Redundancy and Capability

6. GDC 29, Protection against Anticipated Operational Occurrences

7. GDC 35, Emergency Core Cooling

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.

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

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

3. An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.

To meet the requirements of GDC 10, 13, 15, 26, and 35, the positions of RG 1.105, "Instrument Setpoints for Safety-Related Systems," are useful as to their impact on the plant response to the type of transient addressed in this DSRS section.

The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, should be assumed in the analysis and should satisfy the positions of RG 1.53.

The applicant's analysis of this transient should use an acceptable analytical model. If the applicant proposes to use analytical methods not previously reviewed and approved by the staff, the staff evaluates them for acceptability. For new generic methods, the reviewer initiates an evaluation of the new analytical model.

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

1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to operate plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The operating condition at the initiation of the event should correspond to the operating condition that maximizes the consequences of the event.

2. Applicant should conservatively assume the maximum time delay and the most reactive rod held out of the core.

3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

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

<u>Technical Rationale</u>

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 design of the reactor core and its coolant, control, and protection systems with appropriate margin so specified acceptable fuel design limits (SAFDLs) are not exceeded during any conditions of normal operation, including the effects of AOOs.

   GDC 10 applies to this section because the reviewer evaluates the consequences of an inadvertent operation of the ECCS. This AOO could exceed allowable thermal design criteria for fuel cladding integrity. RG 1.53 provides guidance for applying the single-failure criterion to the design and analysis of nuclear power plant protection systems. RG 1.105 provides guidance for ensuring that instrument setpoints remain within the TS limits.

   GDC 10 requirements provide assurance that SAFDLs are not exceeded during an inadvertent operation of the ECCS.

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 design of the RCS and its auxiliary, control, and protection systems with sufficient margin so reactor coolant pressure boundary design conditions are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 applies to this section because the reviewer evaluates the consequences of the inadvertent operation of the ECCS.

4. GDC 20 requires that each RPS be designed (1) to initiate the automatic operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

   GDC 20 is applicable to this section because the reviewer evaluates the effects and consequences of an inadvertent operation of the ECCS due to system malfunction or operator error to ensure that acceptable fuel design limits are not exceeded as a result of AOOs. The reactor protection system automatically initiates the operation of appropriate systems, including the reactivity control system, to terminate the AOOs analyzed in this DSRS section. AOOs such as those caused by an inadvertent operation of the ECCS are terminated in a timely manner so that acceptable specified fuel design limits are not exceeded. This DSRS section and SRP Sections 4.2, 4.3, and DSRS Section 4.4, and Chapter 7 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of AOOs.

5. GDC 26 requires that reactivity control systems at nuclear power plants include control rods that can control reactivity changes so SAFDLs are not exceeded under conditions of normal operation, including AOOs. This system design must have an appropriate margin to accommodate malfunctions (e.g., stuck rods).

   GDC 26 applies to this section because the transient analyzed by the reviewer may require the responsive movement of control rods. In such instances, rod misalignment, including stuck rods, can result in more-severe thermal-hydraulic conditions. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. DSRS Section 15.6.6 examines this margin for whether thermal criteria are satisfied. Compliance with GDC 26 is best demonstrated by showing that adequate thermal margin is maintained during the event as the result of automatic protective action (e.g., a reactor trip) actuated by the monitoring of parameters related directly to thermal margin. The review should encompass claims that automatic reactor protection is available from specified trip signals in addition to the trip signal credited in the licensing-basis analysis. The review should verify whether such signals can provide adequate, timely protection.

GDC 26 requirements provide assurance of appropriate margins to accommodate malfunctions of the reactivity control system, including stuck rods, minimizing the possibility that SAFDLs would be exceeded.

6. GDC 29 requires the protection and reactivity control systems to be designed in a way to assure an extremely high probability of accomplishing their safety functions in the event of an AOO.

   GDC 29 applies to this section because the reviewer evaluates the consequences of an inadvertent operation of the ECCS. The protection and reactivity control systems must be able to accomplish their safety function in the event of this AOO.

GDC 35 requires providing abundant emergency core cooling. The system safety function must be satisfied to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible.

1. 10 CFR 50.34(f), Additional TMI-Related Requirements, applies to this section because the TMI incident involved a stuck-open power-operated relief valve. For plants licensed under 10 CFR Part 52, the requirements of 10 CFR 50.34 are incorporated under 10 CFR 52.47 and 10 CFR 52.79.

III. REVIEW PROCEDURES

The review procedures described below 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.

3. The applicant's description of the inadvertent operation of the ECCS is reviewed for the occurrences leading to the initiating event. The sequence of events from initiation until 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 RPSs are required to function.

C. The credit taken for the functioning of normally operating plant systems.

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

E. The extent to which operator actions are required.

F. If the applicant's technical submittal states that the inadvertent operation of the ECCS is not as limiting as some other similar transient, the reviewer evaluates the applicant's justification. If the applicant's technical submittal presents a quantitative analysis of the transient, the timing of the initiation of those protection, engineered safety, and other systems needed to limit transient consequence to acceptable levels is reviewed. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints.

4. To the extent deemed necessary, the reviewer evaluates the effects of system and component single active failures that may alter the course of the transient. In this phase of the review, the system reviews are as described in the DSRS sections for technical submittal Chapters 5, 6, 7, and 8. The reviewer considers possible single failures in systems that replenish or maintain the reactor coolant inventory.

5. The applicant's mathematical models to evaluate core performance and to predict system pressure in the RCS are reviewed to determine whether they have been reviewed and found acceptable by the staff. If not, the reviewer initiates a generic review of the applicant's proposed model.

6. The values of system parameters and initial core and system conditions as input to the model also are reviewed. Of particular importance are the reactivity coefficients and control rod worths in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The applicant's justification showing that the selected core burnup yields the minimum margins is evaluated.

7. The results of the analysis are reviewed and compared to the acceptance criteria of Subsection II for the maximum pressure in the RCS. The following transient parameters are reviewed: reactor power, heat fluxes (average and maximum), RCS pressure, pressurizer water volume, minimum DNBR, core coolant flow rate, coolant conditions (inlet temperature, core average temperature, average exit and hot assembly exit temperatures, and steam fractions), containment pressure, RVV and RRV flow rates, and flow rate from the RCS to the containment system.

8. Values of the more important of these parameters for the transient caused by the inadvertent RVV or RRV openings are compared to those predicted for other similar plants, if applicable, to determine whether they are within the expected range.

9. Upon request from the reviewer, other responsible organizations provide input for the areas of review stated in Subsection I of this DSRS section. The reviewer uses the input requested as required to complete the review procedure.

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. 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 final safety analysis report (FSAR).

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, 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 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 SER. The reviewer also states the bases for those conclusions.

The applicant evaluated this transient using a mathematical model previously reviewed and found acceptable by the staff. The input parameters for this model were reviewed and found suitably conservative. The results showed SAFDLs maintained by a minimum DNBR not below the DNBR limit value____ and a maximum pressure within the reactor coolant and main steam systems not in excess of 110 percent of the design pressures.

The staff concludes that the analysis or evaluation of this AOO is acceptable and meets the relevant requirements of GDC 10, 13, 15, 20, 26, 29, and 35 and the applicable paragraphs of 10 CFR 50.34(f)(1). This conclusion is based on the following findings:

1. The applicant meets the requirements of GDC 10, 20, 26, and 35 by demonstrating that resultant fuel integrity is maintained because the SAFDLs 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 RCPB limits were not exceeded by the event and that resultant leakage is within acceptable limits. This requirement is met because the maximum pressure within the reactor coolant did not exceed 110 percent of the design pressure.

4. The applicant meets GDC 20 requirements by demonstrating that the protection system is designed to initiate automatic systems, including the reactivity control systems, to assure the SAFDLs are not exceeded as a result of AOOs.

5. The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during the event with appropriate margin for stuck rods because the SAFDLs were not exceeded.

6. The applicant meets GDC 29 requirements by demonstrating that the protection and reactivity control systems were designed in a way to assure an extremely high probability of accomplishing their safety functions in the event of an inadvertent operation of the ECCS.

7. The applicant meets GDC 35 requirement for removal of heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

8. The staff has determined that, in meeting GDC 10, 13, 15, 20, 26, and 35, the analysis used a mathematical model previously reviewed and accepted by the staff. The input parameters for this model were reviewed and found suitably conservative. In addition, we have determined further that the positions of RG 1.53 on single-failure criterion and RG 1.105 for instrument setpoints also are satisfied.

9. The applicant has shown that this AOO would not develop into a postulated accident without other faults occurring independently.

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 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, "General Design Criteria for Nuclear Power Plants," Appendix A, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

2. U.S. Nuclear Regulatory Commission, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," NUREG-0718.

3. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980, ADAMS Accession No. ML051400209.

4. U.S. Nuclear Regulatory Commission, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'" for Westinghouse-designed nuclear steam supply systems, Generic Letter 85-12, June 28, 1985.

5. U.S. Nuclear Regulatory Commission, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'" for Babcock and Wilcox-designed nuclear steam supply systems, Generic Letter 86-05, May 29, 1986.

6. U.S. Nuclear Regulatory Commission, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'" for Combustion Engineering-designed nuclear steam supply systems, Generic Letter 86-06, May 29, 1986.

7. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection against Overpressure."

8. U.S. Nuclear Regulatory Commission, "Instrument Setpoints for Safety-Related Systems," RG 1.105, ADAMS Accession No. ML003740318.

9. U.S. Nuclear Regulatory Commission, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems,” RG 1.53, ADAMS Accession No. ML003740182.