U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of transient and accident analyses for NuScale Power

Secondary - None

I. AREAS OF REVIEW

The purpose of this report is to use the risk informed approach to classify the set of design basis events for a single module that will be analyzed in the NuScale design certification (DC) application. Some of the anticipated operational occurences (AOOs) may be reclassified as Infrequent Events (IE).

The specific areas of review are as follows:

1. The types of control rod misoperations that are assumed to occur. For NuScale, this may include one or more rods moving or displaced from normal or allowed control bank positions (such as dropped rods and rods left behind when inserting or withdrawing banks, or single rod withdrawal) and may include the automatic control system attempting to maintain full power.

2. Descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations.

3. Descriptions of the sequence of events occurring during each anticipated operational occurrence (AOO), e.g., rod drop followed by automatic return to full power with possible power overshoot, including the effect of important feedback mechanisms and trips.

4. Descriptions of the calculational models used and justification of their validity and adequacy.

5. The input to the calculations, including rod worths, power distributions, and feedback coefficients and evidence of the conservatism of the input.

6. Results of the analyses, including, for each of the AOOs considered, plots of the time history of reactor power, reactor vessel pressure, critical heat flux for the limiting fuel

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rod, and maximum fuel centerline temperature or linear heat generation rate.

7. COL Action Items and Certification Requirements and Restrictions. For a 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 design specific review standard (DSRS) Section 15.0.

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

3. Review of the limiting anomalies for NuScale with normal modes of control rod operation (subcritical, low power, start-up and at power) is performed under Design-Specific Review Plan (DSRS) Sections 15.4.1 and 15.4.2.

4. Uniform cladding strain and fuel centerline temperatures (for PWRs) are reviewed under DSRS Section 4.2.

5. Reactivity coefficients and control rod worths are reviewed under DSRS Section 4.3

6. Thermal margin limits are reviewed under DSRS Section 4.4.

7. Safety systems required to prevent misoperations, as required by General Design Criterion (GDC) 25, as well as the control rod system are reviewed under DSRS 7 Sections. The purpose of the review is to determine the events that are to be included as single error malfunctions (e.g., single rod withdrawal).

The specific acceptance criteria and review procedures are contained in the referenced DSRS sections.

II. ACCEPTANCE CRITERIA

Requirements

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

The following GDCs apply:

1. GDC 10, Reactor Design.

2. GDC 13, Instrumentation and Control.

3. GDC 20, Protection System Functions.

4. GDC 25, Protection System Requirements for Reactivity Control Malfunctions.

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.

The requirements of General Design Criteria 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

1. The thermal margin limits (departure from nucleate boiling ratio) as specified in DSRS Section 4.4, subsection II.1, are met.

2. Fuel centerline temperatures as specified in DSRS Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.

3. Uniform cladding strain as specified in DSRS Section 4.2, subsection II.A.2(b), does not exceed 1%.

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

   GDC 10 is applicable to this section because the reviewer evaluates the effects and consequences of a control rod misoperation due to system malfunction or operator error for a BWR or a PWR to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. This DSRS section and DSRS Sections 4.2, 4.3, 4.4, 7, 15.4.1, and 15.4.2 provide guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits, thereby ensuring that specified acceptable fuel design limits are not exceeded.

Meeting the requirements of GDC 10 provides reasonable assurance that specified acceptable fuel design limits are not exceeded during either normal operations or AOOs (e.g., control rod misoperation) due to system malfunction or operator error.

2. Compliance with 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 projection 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. Compliance with GDC 20 requires that each reactor protection system 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 anticipated operational occurrences, 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 a control rod misoperation 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 (RCS), to terminate the AOOs analyzed in this DSRS section. AOOs such as those caused by a control rod misoperation are terminated in a timely manner so that acceptable specified fuel design limits are not exceeded. This DSRS section and DSRS Sections 4.2, 4.3, 4.4, 7, 15.4.1, and 15.4.2 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

   Meeting the requirements of GDC 20 provides reasonable assurance that specified acceptable fuel design limits will not be exceeded when the reactor protection system initiates operation of appropriate systems to terminate AOOs caused by control rod misoperations due to system malfunction or operator error.

4. Compliance with GDC 25 requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

   GDC 25 is applicable to this section because the reviewer evaluates the effects and consequences of a control rod misoperation due to system malfunction or operator error at power. One criterion specifies that the reactor protection system be designed to ensure that specific acceptable fuel design limits are not exceeded during normal operations or during an AOO, including the event of a single malfunction of the RCS. The reactor protection system operates in a manner that automatically terminates the AOOs analyzed in this DSRS section. This DSRS section and DSRS Sections 4.2, 4.3, 4.4, 7, 15.4.1, and 15.4.2 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of AOOs.

Meeting the requirements of GDC 25 provides reasonable assurance that a single malfunction of the reactivity control system, together with anticipated operational occurrences caused by the initiating event of a control rod misoperation due to system malfunction or operator error during either normal operation or an AOO, will not cause specified acceptable fuel design limits to be exceeded.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

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: 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.

3. The reviewer, in determining whether the criteria are met, should determine the AOOs to be considered for this event. Generally, the list of errors should include: (1) inadvertently withdrawing one or several rods, (2) leaving one or several rods behind during bank withdrawal, and (3) inserting one or several rods with power compensation in other portions of the core. In addition to these events, the reviewer must also decide, by postulating single failures in equipment or errors in operation, whether additional single rod malfunctions can be created. Once the list of AOOs has been established, the reviewer must determine acceptability in accordance with the criteria of subsection II of this DSRS section.

4. For each failure event analyzed, the cases which result in a limiting fuel rod condition should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. Conditions of first-order importance for any time in cycle are initial power level and distribution, initial rod configuration, reactivity addition rate, moderator temperature, fuel temperature, and void reactivity coefficients.

5. For each event, the analytical methods used by the applicant are reviewed. Those steady-state and AOO methods that are primarily based on reactor physics considerations are the responsibility of the organization responsible for reactor systems. Where thermal-hydraulic methods are involved, review assistance may be requested as described in DSRS Section 4.4. In either case, the reviewer should determine whether the applicant's evaluation methods are acceptable. This may be done by using one or more of the following procedures:

    A. Determine whether the method has been reviewed and approved previously by considering past safety evaluation reports (SERs) and reports prepared in response to specific technical assistance requests.

    B. Perform an independent review of the method (usually described in a separate licensing topical report and often completed, on a generic basis, outside the scope of the review for a particular facility).

C. Perform auditing-type calculations with methods available to the staff.

D. Request additional bounding calculations from the applicant to confirm the validity of those portions of the applicant's analytical method that have not already been fully reviewed and approved.

6. For each event, the results are evaluated. In addition to verifying conformance to the acceptance criteria of subsection II above, the reviewer determines that:

   A. Input conditions (e.g., pressure, temperature, flow rate) are at the adverse end of the range of values specified as the operating range.

   B. Initial power is 102% of licensed core thermal power, unless a lower power level is justified by the applicant.

   C. Output signals (power, temperature, flux perturbation) provide adequate alarm or scram signals.

   D. Nuclear conditions that interact with this event (e.g., Doppler coefficient, void coefficient) have been calculated as described in DSRS Section 4.3.

7. 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 final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR 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 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 (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 possibilities for single failures of the reactor control system which could result in a movement or misposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod misposition configurations, the course of the resulting AOOs or steady-state conditions, and the instrumentation response to the AOO or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects due to moderator

and fuel temperature changes, have been examined. (If audit calculations have been done, they should be summarized.)

The staff concludes that the requirements of General Design Criteria 10, 13, 20, and 25 have been met. This conclusion is based on the following:

1. GDC 10, ensuring that the specified acceptable fuel design limits are not exceeded.

2. GDC 13, ensuring 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. GDC 20, ensuring that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded.

4. GDC 25, ensuring that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded.

   These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and clad strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that limiting configurations and AOOs for single error control rod malfunctions have been analyzed, that the analysis methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be 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 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. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. 10 CFR Part 50, Appendix A, General Design Criterion 13, “Instrumentation and Control.”

3. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions"

4. 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."