Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale^{TM} SMR DESIGN

15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of transient and accident analyses for SMR.

Secondary - None

I. AREAS OF REVIEW

A malfunction of the reactivity control system or the control rod drive mechanism (CRDM) may cause an uncontrolled withdrawal of a control rod bank, or banks, from the reactor while at power. During the inadvertent withdrawal and the subsequent insertion of positive reactivity, the reactor power level and fuel and coolant temperatures may rapidly increase as a result of the mismatch between power generation and heat removal. Depending on the magnitude of the positive reactivity insertion and the resulting response of the reactor systems, the resulting power surge may lead to overheating of the fuel with resulting fuel damage and radiological release.

Either the reactor protection system trips the reactor terminating the event or pre-exsiting margins are available to prevent the specified acceptable fuel design limits (SAFDLs) from being violated. Examples of reactor trips include high neutron flux, high neutron flux rate, pressurizer high pressure and high pressurizer water level. Prior to the trip the increase in fuel temperature, resulting in an insertion of negative reactivity, counters the positive reactivity due to the withdrawal. Following the trip the reactor becomes subcritical, fuel rod heat fluxes decease and the system transitions to decay heat removal. The objective of the analysis is to demonstrate that the reactor trip setpoints or pre-exsiting margins prevent the departure from nucleate boiling and that the acceptable fuel design limits are not exceeded.

The specific areas of reviews are as follows:

1. The reviewer evaluates the effects and consequences of an uncontrolled withdrawal of a control rod bank or banks at power to ensure conformance with the requirements of General Design Criteria (GDC) 10, 13, 17, 20, and 25 under this design-specific review standard (DSRS) section. The review under this DSRS section covers the description of the causes of the anticipated operational occurrence (AOO) and of the AOO itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the AOOs as compared with the acceptance criteria.

ADAMS Accession No. ML15118A600

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 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. Fuel centerline temperatures are reviewed under DSRS Section 4.2, Subsections II.A.2(a) and (b).

4. Uniform cladding strain as specified in DSRS Section 4.2, subsection II.A.2(b).

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

6. The thermal margin limits (departure from nucleate boiling ratio (DNBR)) are reviewed under DSRS Section 4.4.

II. ACCEPTANCE CRITERIA

Requirements

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

The following GDC apply:

1. General Design Criteria (GDC) 10, Reactor Design.

2. GDC 13, Instrumentation and Control.

3. GDC 17, Electric Power Systems.

4. GDC 20, Protection System Functions.

5. 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 GDC 10, 17, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

1. The DNBR thermal margin limits 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. GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

   GDC 10 is applicable to this section because the reviewer evaluates the effects and consequences of an uncontrolled withdrawal of a control rod bank or banks at power to ensure that SAFDLs are not exceeded during normal operation, including the effects of AOOs. DSRS Section 15.4.2 as well as DSRS Sections 4.2, 4.3, and 4.4 provide guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

   Meeting the requirements of GDC 10 provides reasonable assurance that SAFDLs are not exceeded for AOOs caused by an uncontrolled control rod bank or banks withdrawal at power.

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 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. Compliance with GDC 17 requires (in part) that an onsite and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (1) SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

   GDC 17 is applicable to DSRS Section 15.4.2 because this section reviews an AOO to which the GDC is applicable.

   Meeting the requirements of GDC 17 provides reasonable assurance that an uncontrolled control rod bank or banks withdrawal at power, in combination with a loss-of-offsite power, will not result in a reactor transient that could cause the reactor coolant pressure boundary design conditions or the fuel design limits to be exceeded.

4. Compliance with GDC 20 requires that the protection system be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that SAFDLs 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 uncontrolled withdrawal of a control rod bank or banks at power. The reactor protection system (RPS) automatically initiates the operation of appropriate systems in a timely manner to terminate the AOOs analyzed in this DSRS section. The AOOs are terminated in a timely manner so that acceptable specified fuel design limits are not exceeded. DSRS Section 15.4.2 as well as DSRS Sections 4.2, 4.3, and 4.4 provide guidance for ensuring that SAFDLs are not exceeded as a result of AOOs.

   Meeting the requirements of GDC 20 provides reasonable assurance that SAFDLs are not exceeded by ensuring that the RPS initiates the operation of appropriate systems in a timely manner to terminate AOOs caused by an uncontrolled control rod bank or banks withdrawal at power.

5. Compliance with GDC 25 requires that the protection system be designed to assure that SAFDLs 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, as discussed in Section II above, the reviewer evaluates the effects and consequences of an uncontrolled withdrawal of a control rod bank or banks at power. One criterion specifies that the RPS be designed to ensure that acceptable fuel design limits are not exceeded during normal operation or AOOs, including in the event of a single malfunction of the RCS. The RPS operates in a timely manner to initiate automatic termination of the AOOs analyzed in this DSRS section. DSRS Section 15.4.2 as well as DSRS Sections 4.2, 4.3 and 4.4 provides guidance for ensuring that SAFDLs are not exceeded as a result of operation or AOOs.

   Meeting the requirements of GDC 25 provides reasonable assurance that a single malfunction of the reactivity control system, together with AOOs caused by the initiating event of an uncontrolled control rod bank or banks withdrawal at power, will not cause SAFDLs to be 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: 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 NRC reviewer considers the entire power range from low to full power and the allowed extreme range of reactor conditions during the operating (fuel) cycle, including rod configurations, power distribution, and associated reactivity feedback components. The continuous withdrawal of normal configurations of rods should be assumed for the initial conditions in the AOO calculation. For NuScale this may be one or more control banks. The review considers a full range of bank withdrawals and reactivity rates, up to the maximum bank worth and rate of reactivity addition.

   The exact analysis of the AOO would ideally involve three-dimensional, coupled neutron kinetics, thermal-hydraulics calculation. However, acceptable results may be obtained with suitable approximate calculations.

4. Because power distributions in the course of the AOO can frequently be predicted conservatively using design-limit peaking factors, point kinetics may be used for the nuclear AOO. The nuclear AOO is coupled, however, to core and system thermal-hydraulic response to the power changes (fuel and moderator thermal feedback and system instrumentation response).

   The NRC reviewer ascertains that a full range of AOO conditions are analyzed; the AOO calculation models are adequate; and that scram response of the flux, temperature, or pressure instrumentation is correctly calculated. The range of parameters to be considered includes:

   A. Initial power levels from low to full power.

   B. Reactivity insertion rates from very low to maximum possible for the control system, including allowance for uncertainties.

   C. Fuel and moderator feedback reactivity coefficients covering the range expected throughout the cycle, including allowance for uncertainties.

   D. Power peaking factors at design limits for the initial power level conditions.

. The NRC reviewer determines whether the applicant's analytical methods and models are acceptable, including steady-state, AOO, system response, and fuel response models. 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 SERs and reports prepared in response to technical assistance requests.

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

C. Perform auditing-type calculations using 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 methods that are not fully reviewed or approved.

5. For new application reviews, the analysis must consider a loss of offsite power in conjunction with the limiting single active failure when assessing the consequences of the AOO.

6. The results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, scram or rod block actions that occur, reactor temperatures and pressures, maximum heat flux levels, and related fuel duty (operating conditions and performance). The latter are compared with the acceptance criteria in Subsection II of this DSRS section.

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 applicant's 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 applicant's 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 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 safety evaluation report. The reviewer also states the bases for those conclusions.

1. The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods at power conditions have been reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worth, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions and reactivity feedback effects due to moderator and fuel temperature

changes, have been examined. If audit calculations have been done, they should be summarized.

2. The staff concludes that the requirements of General Design Criteria 10, 13, 17, 20, and 25 have been met. This conclusion is based on the following:

   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.

   The applicant has met the requirements of GDC 10 that the SAFDLs are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that SAFDLs are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the SAFDLs to be exceeded with and without offsite electrical power availability in accordance with the requirements of GDC 17.

   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 assure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analyses of the maximum AOOs for single error control rod bank(s) withdrawal at power condition have been confirmed, that the analytical methods and input data are reasonably conservative and that SAFDLs 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, GDC 10, "Reactor Design."

2. 10 CFR Part 50, Appendix A, GDC 13, “Instrumentation and Control.”

3. 10 CFR Part 50, Appendix A, GDC 17, "Electric Power Systems."

4. 10 CFR Part 50, Appendix A, GDC 20, "Protection System Functions."

5. 10 CFR Part 50, Appendix A, GDC 25, "Protection System Requirements for Reactivity Control Malfunctions."