Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

15.4.6 INADVERTENT DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT SYSTEM (PWR)

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

Unborated water can be added to the reactor coolant system (RCS), via the chemical volume and control system (CVCS), to increase core reactivity. This may be inadvertent due to operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. A specific plant design may include other inadvertent additions of unborated water to the RCS, like instrument flushing systems. During refueling, the NuScale RCS is disconnected from the CVCS and the Nuclear Power Module (NPM) is moved to the refueling pool where it will be disassembled and open to the refueling pool boron concentration. Therefore, the reviewer needs to evaluate any potential for an inadvertent dilution of the refueling pool. An inadvertent addition of unborated water to the RCS or refueling pool is possible due to operator error or malfunction of systems. The operator must stop this unplanned dilution before the shutdown margin is eliminated. As sequences of events that may occur depend on plant conditions at the time of the unplanned moderator dilution, the review includes conditions like refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown, and cold shutdown.

The specific areas of review are as follow:

1. The review of postulated moderator dilution events considers causes, initiating events, the sequence of events, the analytical model, the values of parameters in the analytical model, and predicted consequences of the event.

2. The sequence of events described in the applicant's safety analysis report (SAR) is reviewed. The reviewer concentrates on the need for the reactor protection system and required operator action to secure and maintain the reactor in a safe condition.

3. The analytical methods are reviewed for whether the mathematical modeling and computer codes have been accepted by the staff. If a referenced analytical method has

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not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model.

4. The predicted results of moderator dilution events are reviewed to assure that the consequences meet the acceptance criteria of subsection II of this Standard Review Plan (SRP) section. The results of the transients are also reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

5. <u>COL Action Items and Certification Requirements and Restrictions</u>. 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.

<u>Review Interfaces</u>

Other SRP 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. 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 SAR analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

4. Systems for emergency injection of borated cooling water are reviewed under DSRS Section 6.3.

5. Aspects of the sequence described in the SAR are reviewed to confirm that reactor and plant protection, engineered safety features (EFS) controls, interlocks, and other instrumentation and control systems important to safety will function as assumed in the safety analyses under DSRS Sections 7.0 through 7.2.

6. Functional and operational characteristics and potential failure modes of the CVCS are reviewed under DSRS Section 9.3.4. This includes the functional and operational characteristics and potential failure modes of other systems identified as having the potential for an inadvertent addition of unborated water into the RCS due to operator error

or system malfunction. The reviewer of this section makes use of this review to evaluate initiating causes and the expected sequence of events.

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

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 26, Reactivity Control System Redundancy and Capability.

5. The general objective of the review of moderator dilution events is to confirm either of the following conditions is met:

   A. The consequences of these events are less severe than those of another transient that results in an uncontrolled increase in reactivity and has the same anticipated frequency classification.

   B. The plant responds to events such that the criteria regarding fuel damage and system pressure are met and the dilution transient is terminated before the shutdown margin is eliminated.

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 and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity must be maintained so the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations with SRP Section 4.4.

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

4. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:

   A. During refueling: 30 minutes.

   B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.

5. The applicant's analysis of moderator dilution events should use an acceptable analytical model. Staff must evaluate any proposed unreviewed analytical methods. The reviewer initiates an evaluation of new generic methods. The following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown and cold shutdown. Parameters and assumptions in the analytical model should be suitably conservative. The following values and assumptions are acceptable:

   A. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent to account for power-measurement uncertainty. The analysis may use a smaller power-measurement uncertainty if justified adequately.

   B. The boron dilution is assumed to occur at the maximum possible rate.

   C. Core burnup and corresponding boron concentration should yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. The core burnup should be justified by either analysis or evaluation.

   D. All fuel assemblies are installed in the core.

   E. A conservatively low value is assumed for the reactor coolant or refueling pool volume.

   F. For analyses during refueling, all control rods are withdrawn from the core. An alternate assumption requires adequate justification and delineation of necessary controls so the alternate assumption remains valid.

   G. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1 percent) is assumed prior to boron dilution.

H. A conservatively high reactivity addition rate is assumed for each analyzed event to take into account the effect of increasing boron worth with dilution.

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

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP 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 acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. Fuel design limits are established to assure the integrity of fuel cladding as a fire protection (FP) barrier.

   In PWRs, boron is added to the reactor coolant in sufficient concentrations for reactivity control. PWR conditions of normal operation include startup, power operation, hot standby, shutdown (hot and cold), and refueling modes. Because of the frequency of boron dilution events (one or more times during the life of the nuclear power unit) without other concurrent failures or incidents, regulatory requirements for AOOs apply to their analyses or evaluations. Uncertainties of quantification or measurement of relevant boron dilution event parameters are verified by analyses including appropriate design margins.

   Thus, GDC 10 assures that analyses demonstrate, under all operating, shutdown, and refueling modes, whether the reactor core and its coolant, control, and protection systems are designed with sufficient margins for postulated boron dilution events to maintain fuel cladding integrity.

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 instrumental indicators.

3. GDC 15 requires design of the RCS and its auxiliary, control, and protection systems with sufficient margin so RCPB design conditions are not exceeded during any normal operation, including AOOs. Design conditions (*e.g.,* pressure limits for transients) of the RCPB are established to assure its integrity. The RCPB provides a FP barrier, confined volume for the inventory of reactor coolant, and flow paths for core cooling.

   In PWRs, boron is added to the reactor coolant in sufficient concentrations for reactivity control. PWR conditions of normal operation include startup, power operation, hot standby, shutdown (hot and cold), and refueling modes. Because of the frequency of boron dilution events (one or more times during the life of the nuclear power unit) without other concurrent failures or incidents, regulatory requirements for AOOs apply to their analyses or evaluations.

RCS pressure transients of power increases caused by postulated boron dilution events are analyzed for whether pressure limiting design features, including conservatively assumed responses of control and protection systems, maintain pressures below the RCPB design pressure limits for transients. Uncertainties of quantification or measurement of relevant boron dilution event parameters are verified by analyses including appropriate design margins.

Thus, GDC 15 assures that analyses demonstrate, under conditions of normal operation, including the effects of postulated boron dilution events, design of the RCS and its auxiliary, control, and protection systems with sufficient margin to maintain RCPB integrity.

4. GDC 26 requires control rods to control reactivity changes to assure that acceptable fuel design limits are not exceeded under normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods. Fuel design limits are established to assure the integrity of fuel cladding as a FP barrier.

   In PWRs, a control rod system is provided for reactivity control. Boron is also added to the reactor coolant in sufficient concentrations for reactivity control. PWR conditions of normal operation include startup, power operation, hot standby, shutdown (hot and cold), and refueling modes; however, the control rods may reduce reactivity only when withdrawn and operable (*i.e.* during startup and power operation). Because of the frequency of boron dilution events (one or more times during the life of the nuclear power unit) without other concurrent failures or incidents, regulatory requirements for AOOs apply to their analyses or evaluations. Uncertainties of quantification or measurement of relevant boron dilution event parameters are verified by analyses including appropriate design margins. To address single failures not attributable to a common cause/mode, the control rod system is designed for a specified minimum shutdown margin without credit for the functioning of the highest worth control rod.

   Thus, GDC 26 assures that analyses demonstrate whether the control rods reliably control reactivity changes with appropriate margin for malfunctions like stuck rods under applicable conditions of normal operation (startup and power operation), including the effects of postulated boron dilution events, to maintain fuel cladding integrity.

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 SAR describes moderator dilution transients reviewed by the staff for the occurrences leading to the initiating events. The sequence of events, from initiation to a stabilized condition, is reviewed for:

   A. The extent that normally operating plant instrumentation and controls are assumed to function. Alarms that alert operators to the unplanned boron dilution are of particular importance.

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

   C. The credit taken for normally operating plant systems.

   D. The operation of required engineered safety systems.

   E. The extent of required operator actions.

   F. The appropriate margin for malfunctions (*e.g.,* stuck rods) is accounted for.

4. The reviewer confirms that analyses are included for a boron dilution incident during each of the following plant initial conditions: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown, and cold shutdown. Refueling condition analyses the dilution of the refueling pool when the reactor vessel head is removed. For each reviewed incident, the applicant must consider all causes and justify the cause selected for analysis as allowing the operator the least time to take corrective action.

5. The staff reviews the timing of initiation of protection, engineered safety, and other systems needed to limit the consequences of each boron dilution incident to acceptable levels. The reviewer compares the predicted variations of system parameters to various trip and system initiation setpoints. The reviewer also evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation where the SAR states that operator action is needed or expected.

6. When necessary, the reviewer evaluates the effects of single active failures of systems and components that may affect the course of the transient. This phase of the review uses SRP system review procedures for SAR Chapters 5, 6, 7, 8, and 9. In particular, redundant alarms that alert the operator to the unplanned dilution are confirmed.

7. The mathematical models used by the applicant to evaluate core performance and reactivity status are reviewed to determine whether these models have been accepted by staff. If not, a generic review of the model proposed by the applicant is initiated.

8. The staff reviews system parameter values and initial core and system conditions used as input to the model. The reactivity coefficients and control rod worths used by the applicant are of particular importance. The reviewer evaluates the justification provided by the applicant to show that the selected core burnup condition, boron concentration, and rod worths yield the minimum margins. The staff reviews the reactivity parameter values in the applicant's analysis under SRP Section 4.2. The value of core reactivity as a function of time following each analyzed incident is confirmed by comparison to an acceptable analysis for another plant, by comparison to staff calculations for typical plants, or by independent calculations of the reviewer.

9. The assumed dilution flow rates are reviewed, taking into consideration the system parameters acting to limit the flow. The reviewer examines the flow limiting equipment characteristics provided by the applicant to justify flow rate assumptions (*e.g.*, if the flow is limited by the charging pump capacity, the assumed flow is compared to that for all charging pumps acting at full capacity). If a lesser flow value is assumed (*e.g.,* not all pumps operating or flow limited by a valve), justification is necessary. The secondary reviewer is consulted concerning any interlocks for which credit is taken.

10. The results of the analyses are reviewed and compared to the acceptance criteria of subsection II of this SRP section regarding the time available for the operator to take corrective action. Variations with time during the transient of important parameters are compared to those predicted for other similar plants for compliance with the expected range. Parameters of particular importance are core reactivity, boron concentration, rate of addition of unborated water, power level, core pressure, and minimum DNBR.

11. 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.

Various operator errors and system malfunctions that could lead to an inadvertent boron dilution incident have been reviewed. Operator errors and system malfunctions that allow the operator the shortest time for corrective action have been analyzed from plant conditions of startup, power operation (automatic and manual), hot standby, hot shutdown, cold shutdown, and refueling. The applicant evaluated these events using a mathematical model previously reviewed and found suitably conservative. Results of the analyses showed that the operator has ___ minutes to take corrective action if a boron dilution incident occurs during refueling and __ minutes if at-power operation (automatic and manual), startup, hot standby, hot shutdown, and cold shutdown. In the latter case, the most severe transient results in a minimum DNBR of __ and reactor coolant and main steam system pressures of less than 110 percent of design.

The staff concludes that analysis for the decrease in the reactor coolant boron concentration event is acceptable and meets GDCs 10, 13, 15, and 26 requirements. This conclusion is based on the following findings:

1. The applicant meets GDC 10 requirements by demonstrating that the specified acceptable fuel design limits are not exceeded for this event. This requirement is met as the results of the analysis show that the thermal margin limits (minimum DNBR for PWRs) are satisfied as indicated by SER Section 4.4.

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 whether the RCPB limits are not exceeded for this event. This requirement is met as the analysis show that the maximum pressure in the reactor coolant and main steam systems did not exceed 110 percent of the design pressure.

4. The applicant meets GDC 26 requirements by demonstrating that the control rod system can overcome the effects of boron dilution events during reactor operation. The applicant fulfills these requirements by showing under the postulated operational occurrence conditions, and with appropriate margins for stuck rods, that the specified acceptable fuel design limits are 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 SRP 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 15, "Reactor Coolant System Design."

4. 10 CFR Part 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

5. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Overpressure Protection 60," ASME.