U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

15.5.1 - 15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are to identify which of the anticipated operational occurrences (AOOs) leading to a reactor coolant system (RCS) inventory increase are the most limiting and to verify that, for the most limiting transients, the plant responds to the RCS inventory increases in such a way that the criteria regarding fuel damage, RCS pressure, and escalation to a more serious event are met. The AOOs that decrease core boron concentration are addressed under Standard Review Plan (SRP) 15.4.6. A power level increase could result depending on the boron concentration, temperature of the injected water, and the response of the automatic control systems. Without adequate control, this event could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization could result. The reactor will trip from high water level, high flux, high pressure, low pressure, or from an engineering safeguards signal.

The events leading to an increase in reactor coolant inventory considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient. The reviewer concentrates on the reactor protection system, the engineered safety systems, and especially operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff, and have been applied in accordance with any limitations that may have been specified in the staff's acceptance. If a referenced analytical method has not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed. Results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within expected ranges.

For a design certification (DC) application, the reviewer addresses combined license (COL) action items and requirements and restrictions (e.g., interface requirements and site parameters).

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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 Design-Specific Review Standard (DSRS) sections interface with this section as follows:

1. Review of the chemical and volume control system is performed in DSRS Section 9.3.4
2. General information on transient and accident analyses is provided in DSRS Chapter 15
3. Review of transient and accident analysis methods in provided in SRP Section 15.0.2.

II. ACCEPTANCE CRITERIA

Requirements

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

1. Title 10 of the Code of Federal Regulations (10 CFR) 52.47(a)(2), which requires evaluations to show that safety functions will be accomplished. Descriptions shall be sufficient to permit understanding of the system design relationship to the safety evaluations.

2. General Design Criterion (GDC) 10, “Reactor Design”

3 GDC 13, “Instrumentation and Control”

4. GDC 15, “Reactor Coolant System Design”

5. GDC 26, “Reactivity Control System Redundancy and Capability”.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC’s regulations identified above are set forth below. The DSRS is not a substitute for the NRC’s regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (for DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with NRC regulations that underlie the DSRS acceptance criteria.

1. Identification of Causes and Frequency Classification: The applicant included a description of the events that lead to an increase in reactor coolant inventory. The frequency classification for this event is an AOO.

2. Sequence of Events and System Operation: The sequence of events, from initiation until a stabilized condition is reached, is reviewed to determine:

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

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

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

D. the extent to which operator actions are required (Note: an operator action to shut off the charging pump flow is normally required to terminate these AOOs.)

E. appropriate margin for malfunctions, such as stuck rods, is taken into consideration

3. Evaluation Model: The applicant's analysis of the event should be performed using an acceptable analytical model. If the applicant proposes analytical methods that have not been approved, then a generic review of the model is performed.

4. Input Parameters and Initial Conditions: The values of parameters used in the analytical model should be suitably conservative such that a limiting set of initial conditions are utilized and uncertainty is accounted for as follows.

   A. The initial power level is taken as the licensed core thermal power plus an allowance for uncertainty.

   B. Conservative SCRAM characteristics are assumed (i.e. maximum time delay with the highest worth rod held out of the core).

   C. The burnup is selected to yield the most limiting combination of reactor physics parameters and power profiles.

5. Results:

   Analysis results should address the following.

   A. The applicant should identify the limiting event scenario along with its basis. Any additional analyses addressing nonlimiting event scenarios need not be discussed in detail, but sufficient information should be provided to establish the basis for selecting the limiting event scenario.

   B. The applicant should present a time table of the sequence of events that occur during the AOO. At a minimum, this time table should address key phenomena, actuation of engineering safeguards, and credited operator actions.

   C. The applicant should present key parameters as a function of time during the course of the transient.

   D. Analysis acceptance criteria are taken from DSRS Section 15.0 and are provided below:

i. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

ii. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

iii. An AOO should not generate a more serious event (i.e. a postulated accident or infrequent event) without other faults occurring independently.

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 DSRS acceptance criteria provides sufficient detail to permit understanding of the system design relationship to the safety evaluation and compliance with 10 CFR 52.47(a)(2).

2. Compliance with DSRS acceptance Criteria 5(d)(ii) and 5(d)(iii) provides reasonable assurance that specified acceptable fuel design limits (SAFDLs) are not exceeded during AOOs and establishes compliance with GDC 10.

3. Compliance with DSRS acceptance Criteria 2(a-e) and 4(a-c) provides reasonable assurance that appropriate instrumentation and control are provided over appropriate ranges and establishes compliance with GDC 13.

4. Compliance with DSRS acceptance Criteria 5(d)(i) and 5(d)(iii) provides reasonable assurance that the design conditions of the reactor coolant pressure boundary are not exceeded and established compliance with GDC 15.

5. Compliance with GDC 26 requires reactivity control system redundancy and capability. The requirements of GDC 26 apply to this section because the appropriate mitigation for an AOO is a reactor trip (and in this case, manual action to end the mass addition to the RCS). Once shut down, the reactor should remain in a shutdown condition. Meeting this criterion provides reasonable assurance that AOOs will not result in fuel damage and subsequent fission produce release.

III. REVIEW PROCEDURE

The reviewer selects 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 reviews the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Selected Programs and Guidance—In accordance with the guidance in NUREG–0800, "Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports

for Nuclear Power Plants: Light-Water Small Modular Reactor Edition" (NUREG-0800, Intro Part 2), as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800, Intro Part 2, the NRC requirements that must be met by an SSC do not change under the small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain SSCs, the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified, where applicable, as part of completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

- 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
- 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
- 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
- 10 CFR 50.55a, Code Design, Inservice Inspection, and Inservice Testing (ISI/IST) Programs
- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, "Technical Specifications"
- Availability Controls for SSCs Subject to Regulatory Treatment of Nonsafety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8), (21), and (22), and 10 CFR 52.79(a)(17), (20), and (37), for DC or COL applications submitted under 10 CFR Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933, "Resolution of Generic Safety Issues," current on the date up to 6 months before the docket date of the application and which are technically relevant to the design, (2) demonstrate how the operating experience insights have been incorporated

into the plant design, and (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v), for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v),for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

The applicant's description of events leading to an increase in reactor coolant inventory is reviewed with respect to the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to determine the following:

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

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

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

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

E. the extent to which operator actions are required (Note: an operator action to shut off the charging pump flow is normally required to terminate these AOOs.)

F. that appropriate margin for malfunctions, such as stuck rods (see Subsection II.3.b), is taken into consideration

The applicant should present a quantitative analysis in the safety analysis report (SAR) of the most limiting events that lead to an increase in reactor coolant inventory. Such an analysis should demonstrate that AOOs could not develop into more serious events. The reviewer examines the timing of the initiation of those protection and engineered safety systems, and operator actions needed to limit the consequences of the event to acceptable levels. The reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the models is initiated.

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

The results of the applicant's analysis are reviewed in accordance with the acceptance criteria presented in Subsection II regarding maximum pressure in the reactor coolant and main steam systems, the minimum critical heat flux ratio (MCHFR) DNBR and the possibility of escalation to a more serious event. The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR)); core and recirculation coolant flow rates, coolant conditions (inlet temperature, core average temperature (PWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system are reviewed, as applicable. The reviewer also compares values of the more important of these parameters for the events leading to an increase in reactor coolant inventory with those predicted for other similar plants to confirm that they are within the expected range.

For review of a DC application, the reviewer follows 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), meet the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer considers 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 are 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 staff concludes that the analysis of a transient resulting in an increase in reactor coolant inventory is acceptable and meets the requirements of GDC 10, 15, and 26 and the guidance of ANS standards. This conclusion is based on the following:

1. In meeting GDC 10, 13, 15, and 26 as discussed below, the staff has determined that the applicant's analysis was performed using a mathematical model that has been reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The staff has further determined that the positions of Regulatory Guide (RG) 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems," for the single-failure criterion and RG 1.105, "Setpoints for Safety-Related Instrumentation," for instruments have also been satisfied.

2. The applicant has met the requirements of GDC 10, and 26 with respect to demonstrating that resultant fuel damage is maintained because the specified acceptable fuel design limits were not exceeded for this event.

3. The applicant has met the 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.

4. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

5. The applicant has met the requirements of GDC 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for malfunctions because the specified acceptable fuel design limits were not exceeded.

6. The applicant has satisfied the ANS design criteria that prohibits the escalation of an AOO to a more serious incident without other incidents, occurring independently.

For DC and COL reviews, the findings 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. <u>IMPLEMENTATION</u>

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," Part 50, Appendix A, Title 10, "Energy."

2. U.S. Nuclear Regulatory Commission, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Regulatory Guide 1.70, Revision 3, May 14, 2001, Agencywide Documents Access and Management System (ADAMS) Accession No. ML011340122.

3. American Nuclear Society (ANS), "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor [PWR] Plants," ANS 51.1, 1983 (replaces ANSI N18.2-1973) (withdrawn in 1998).

4. American National Standards Institute/ANS, ""Nuclear Safety Criteria for the Design of Stationary Boiling-Water Reactor Plants," ANSI/ANS-52.1-1983, (withdrawn in 1998).

5. U.S. Nuclear Regulatory Commission, "Anticipated Transients that Could Develop into More Serious Events," Regulatory Issue Summary 2005-29, December 14, 2005, ADAMS Accession No. ML051890212.

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

7. U.S. Nuclear Regulatory Commission, "Review Standard for Extended Power Uprates," RS-001, Revision 0, December 2003 (Note 8 of Matrix 8 of Section 2.1), ADAMS Accession No. ML033640024.

8. U.S. Nuclear Regulatory Commission, "Classification of Transients and Accidents for the NuScale Power Small Modular Reactor," NuScale Report, NP-WP-0613-3803-P, June 2015, Revision 0, ADAMS Accession No. ML13255A482.