U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

6.2.7 FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of component integrity issues related to Containment

Secondary - None

I. AREAS OF REVIEW

The reactor containment pressure boundary relates to the reactor containment system. The reactor containment system design must include the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant to postulated accidents. This design-specific review standard (DSRS) section reviews fracture prevention of the reactor containment pressure boundary materials of the NuScale integral pressurized water reactor (iPWR).

The reactor containment pressure boundary, as addressed in the U.S. Nuclear Regulatory Commission (NRC) licensing review process, consists of those ferritic steel parts of the reactor containment system which sustain loading and provide a pressure boundary in the performance of the containment function under the operating, maintenance, testing and postulated accident conditions cited by Title 10 *Code of Federal Regulations* Part 50 (10 CFR Part 50), Appendix A, General Design Criterion (GDC) 51, "Fracture Prevention of Containment Pressure Boundary." Within this context, typically reviewed are the ferritic materials of components such as freestanding containment vessels, equipment hatches, personnel airlocks, containment penetration sleeves, process pipes, end closure caps and flued heads, and penetrating-piping systems connecting to penetration process pipes and extending to and including the system isolation valves.
The specific areas of review are as follows:

1. The containment vessel and all penetration assemblies or appurtenances attached to the containment vessel; all piping, and valves attached to the containment vessel, or to penetration assemblies out to and including the pressure boundary materials of any valves required to isolate the system and provide a pressure boundary for the containment function.

2. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this DSRS section in accordance with NUREG-0800, Standard Review Plan (SRP) Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of the application has been

ADAMS Accession No. ML15112A517

reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

3. 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 SRP and DSRS sections interface with this section as follows:

1. Review of the design of steel containments will be performed under DSRS Section 3.8.2, “Steel Containment.”

2. Review of the adequacy of programs for assuring the integrity of bolting and threaded fasteners will be performed under DSRS Section 3.13, “Threaded Fasteners – ASME Code Class 1, 2, and 3.”

3. Determination of SSC risk significance under SRP Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors.”

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criteria (GDC) 1, found in Appendix A to Part 50, as it relates to the quality standards for design and fabrication.

2. GDC 16, as it relates to the prevention of the release of radioactivity to the environment.

3. GDC 51, as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

4. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the NRC's regulations.

5. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that

the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the AEA, and the NRC's regulations.

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.

To meet the requirements of GDCs 1, 16 and 51, NuScale containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified in the appropriate American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Article, consistent with ASME Code Section III Article NCA-2130. For example, should the containment pressure boundary be designed to ASME Code Class 1, Article NB of Section III, Division 1 of the ASME Code would apply.

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 1 requires that SSCs be designed, fabricated, erected and tested commensurate with the importance of the safety functions to be performed. This DSRS section evaluates the fracture toughness of the containment pressure boundary ferritic materials to ensure they are not subject to brittle fracture. For example, ASME Code Section III, Division 1, Class 1, NB-2300 criteria are used in the performance of ASME Code Class 1 fracture toughness evaluations. The containment pressure boundary is one of the barriers that prevents the release of radioactivity to the environment in the event of an accident, and therefore, fulfills a vital safety-related role. Use of appropriate design and fabrication standards in conjunction with the appropriate fracture toughness testing provides assurance that containment will not fail due to brittle behavior and will thus be capable of preventing the release of radioactivity to the environment.

2. GDC 16 requires reactor containment and associated systems to be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Containment must be leak-tight and withstand accidents because it is the final barrier against the release of radioactivity to the environment in the event of a loss-of-coolant accident. To ensure leak-tightness, containment must not be subject to brittle fracture even under the most severe postulated conditions. Meeting GDC 16 provides assurance that containment will satisfactorily fulfill its safety role and that significant radioactivity will not be released to the environment.

3. GDC 51 requires, in part, that the reactor containment boundary be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. As the final barrier against the release of radioactivity to the environment, containment must not be subject to brittle failure or rapidly propagating fracture, either of which could cause a breach of containment integrity. Meeting GDC 51 will ensure that the containment pressure boundary remains intact during the harshest expected conditions, thereby precluding the release of radioactivity to the environment.

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.

The licensing review process assesses the fracture toughness of the materials of the components of the NuScale reactor containment pressure boundary identified in Section I of this document, within the context of compliance with the criteria of the appropriate ASME Code Article. For example, if the vessel is designed and built to ASME Code Class 1, then the criteria of NB-2300 of Section III, Division 1 of the ASME Code apply.

The reviewer reviews the information provided by the applicant for the materials of the components of interest. Such information should consist of construction drawings, piping system diagrams and related supplemental information, ASME Code data reports and certified material test reports.

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. 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 design control document (DCD), 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 DCD.

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).

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

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 SER. The reviewer also states the bases for those conclusions. The staff concludes that reasonable assurance has been provided that the materials of the reactor containment pressure boundary, under operating, maintenance, testing and postulated accident conditions, will not undergo brittle fracture, and that the probability of rapidly propagating fracture will be minimized, so that the requirements of GDCs 1, 16, and 51 will be met. This conclusion is summarized as follows (provide the finding that applies):

Based on its review, the staff finds that the materials of the reactor containment pressure boundary were (or will be, where appropriate) acceptably tested and demonstrated to meet the fracture toughness requirements for the appropriate ASME Code Class of the components as specified in the application. 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. In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

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, §50.55a, Codes and Standards.

2. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

3. 10 CFR Part 50, Appendix A, GDC 16, "Containment Design."

4. 10 CFR Part 50, Appendix A, GDC 51, "Fracture Prevention of Containment Pressure Boundary."

5. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

6. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

7. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."