U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

5.2.2 OVERPRESSURE PROTECTION

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of reactor thermal-hydraulic systems in SMRs.

Secondary - None

INTRODUCTION

The NuScale Power, LLC, (NuScale) small modular reactor (SMR) includes overpressure protection, which is reviewed as a safety-related and risk-significant system. There are two paths (two reactor safety valves for overpressure relief from the top of the pressurizer, which is integral to the reactor pressure vessel). Each path leads to a reactor safety valve that relieves pressure directly to the containment. Low temperature overpressure protection (LTOP) is provided by the two reactor vent valves (RVV) and are part of the emergency core cooling system, which is reviewed under Design Specific Review Standard (DSRS) 6.3.

A remotely operated reactor vessel vent valve and vent line from the pressurizer to the gaseous radwaste system is also provided.

I. AREAS OF REVIEW

The application of safety valves and the reactor protection system ensures overpressure protection for the reactor coolant pressure boundary (RCPB) during operation at power. The application of pressure-relieving systems that function during low-temperature operation ensures overpressure protection for the RCPB during low-temperature operation of the plant (startup, shutdown).

The pressure-retaining portions and supports of mechanical equipment shall be Safety Class 1 if they form part of the RCPB and have requirements that fall within the scope of Section III of Division 1 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). Different allowable stress limits (service limits) for pressure-retaining components provide different margins of failure, and hence, different reliability levels for the pressure-retaining function of the RCPB. Plant conditions and the operating status of the reactor determine the allowable ASME Code service limits. For normal plant operation, the design pressure limit (stress limit) of the RCPB shall not be exceeded. However, the ASME Code allows the design pressure to be exceeded by 10 percent for anticipated operational occurrences (AOOs) such as an inadvertent chemical and volume control system (CVCS) actuation. The transient pressure load allowances are based on the frequency and duration of the event and should be reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) staff.

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The specific areas of review are as follows:

1. System Design. The staff assesses information on the design of the overpressure protection system, including any subsystems and supporting systems, to gain familiarity with the design and operation of the system. Such information includes the following:

   A. The area of review for operation at power includes: the pressurizer; safety valves including the piping (if any) from these valves to the containment vessel on the primary side, as well as thesafety/relief valves on main steam line and steam generator safety valves, if applicable, on the secondary side (). The NuScale description of the basic design concept; the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and, process and instrumentation diagrams should be reviewed for power operation.

   B. The area of review for LTOP of the plant (startup, shutdown) includes two RVVs discharging into the primary containment, the CVCS, the normal shutdown cooling system (dumping steam to the main condenser) and the backup shutdown cooling system (i.e., the decay heat removal system (DHRS)) that may be operating when the primary system is water solid. The safety related DHRS is designed for post-accident operation that dumps heat to the reactor building pool. The NuScale description of the basic design concept; the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and process and instrumentation diagrams should be reviewed for low temperature operation.

2. Testing and Inspections. The areas of review include an examination of the adequacy of the proposed pre-operational and initial startup test programs.

3. Technical Specifications. The areas of review include technical specifications to assure that they are adequate with regard to limiting conditions of operation and periodic surveillance testing.

4. 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 Standard Review Plan (SRP) Section 14.3. The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been 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.

5. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. Review of seismic design criteria for components of the overpressure protection system (under SRP Sections 3.2.1 and 3.2.2).

2. Review of proposed in-service testing of valves to ensure that overpressure components will perform their safety functions (under SRP Section 3.9.6).

3. Review of seismic and dynamic qualification for components of the overpressure protection system (under SRP Section 3.10).

4. Review of environmental qualification for components of the overpressure protection system (under Design Specific Review Standard (DSRS) Section 3.11).

5. Review of the design of systems that interface with the reactor coolant system (RCS) with regard to the capability of the interfacing system to withstand full RCS pressure (under SRP Section 3.12).

6. Review of the fracture toughness of the RCPB and reactor vessel and the pressure-temperature limits and pressurized thermal shock analysis (under SRP Section 5.2.3 and DSRS Sections 5.2.4, 5.3.1, and 5.3.2).

7. Review of temperature and pressure conditions in containment due to pressurizer safety valve discharge (under DSRS Section 6.2.1.1.A).

8. Review of the CVCS to verify that LTOP requirements are met (under DSRS Section 9.3.4.

9. Evaluation of the adequacy of the design, installation, inspection, and testing of all electrical systems required to provide the safety-related and risk-significant functions for overpressure protection (under DSRS Sections 8.3.1 and 8.3.2).

10. Review of the adequacy of controls and instrumentation for the automatic and manual actuation of overpressure protection components (under DSRS Sections 7.0, 7.1, and 7.2.

11. Review of proposed preoperational and initial startup test programs to ensure that overpressure components will perform their safety functions (under DSRS Section 14.2).

12. Review of proposed ITAAC associated with SSCs for RCPB overpressure protection (under SRP Section 14.3).

13. Review of technical specifications (under DSRS Section 16.0).

14. Review of the reliability assurance program is coordinated and performed under SRP Section 17.4.

15. Review of quality assurance requirements (under SRP Section 17.5).

16. Verification of SSC risk significance (under SRP Section 19.3).

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 1, Quality Standards and Records.

2. GDC 15, Reactor Coolant System Design.

3. GDC 30, Quality of Reactor Coolant Pressure Boundary.

4. GDC 31, Fracture Prevention of Reactor Coolant Pressure Boundary.

5. 10 CFR 52.47(a)(8) provides the requirement for design certification reviews to comply with the technically relevant portions of the requirements in 10 CFR 50.34(f).

6. 10 CFR 52.79(a)(17) provides the requirement for COL applications to comply with the technically relevant information in 10 CFR 50.34. This includes the Three Mile Island (TMI)-related requirements specified by 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi), which require that RCS safety valves meet TMI Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

7. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (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 design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the NRC's rules and regulations.

8. 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 be operated in conformity with the COL, the provisions of the Atomic Energy Act, and the NRC's rules and regulations.

9. 10 CFR 20.1406, which requires that facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

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. Material Specifications. The requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of material code cases, as described in Regulatory Guide (RG) 1.84.

   The specifications for permitted materials are identified in Appendix I to Section III of the ASME Code or described in detail in Parts A, B, and C of Section II of the ASME Code. RG 1.84 describes acceptable material code cases and guidelines for application in light-water-cooled nuclear power plants that may be used in conjunction with the above specifications.

2. Design Requirements for SMRs Operating at Power

   A. For overpressure protection during power operation of the NuScale reactor, the pressurizer should have sufficient capacity to preclude actuation of the safety valves during normal operational transients, when assuming the following conditions at the plant:

      i. The reactor is operating at the licensed core thermal power level.

      ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure.

      iii. All components, instrumentation, and controls function normally.

   B. The design of the safety valve should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Overpressure Protection". Also, sufficient available margin should account for uncertainties in the design and operation of the plant, assuming the following:

      i. The reactor is operating at a power level that will produce the most severe over-pressurization transient.

      ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

      iii. The second safety-grade signal from the reactor protection system initiates the reactor scram.

iv. The discharge flow is based on the rated capacities specified in ASME Code Article NB-7000 for each type of valve.

v. In addition, the design of the safety valve should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000.

C. A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.

Full credit is allowed for spring-loaded safety valve designed in accordance with the requirements of ASME Code Article NB-7511.1.

3. Design Requirements for SMRs Operating at Low Temperature (Startup, Shutdown). The design of the LTOP system should be in accordance with the requirements of Branch Technical Position (BTP) 5-2. The LTOP system should be operable during startup and shutdown conditions below the enable temperature defined in paragraph II.2 of BTP 5-2.

4. Testing and Inspections. The performance of tests and inspections should occur before operation and during startup to functionally demonstrate that the overpressure protection system, as installed, meets all design requirements.

5. Technical Specifications. The technical specifications should specify appropriate limiting conditions of operation and in-service surveillance to ensure continued system reliability, including, for SMRs, specific limiting conditions of operation and testing of the LTOP system as specified in NUREG-1430, Generic Letters No. 82-16 and 90-06.

6. TMI-2 Action Plan Requirements. Section II.D.1 of the TMI Action Plan requires an applicant submit a plant specific report regarding relief valve (RV) and safety valve (SV) testing. Section II.D.3 of the TMI Action Plan requires that RVs and SVs be provided with direct valve position indication. Generic Letter No. 82-16 requires sections II.D.1 and II.D.3 be covered by technical specifications while NUREG -0737 section II.K.3.3 specifies reporting for section II.D.1 and II.D.3.

7. Contents of applications; technical information: 10 CFR 52.47(b)(1) requires that a DC application contain proposed ITAAC necessary and sufficient to assure the plant is constructed and will be operated in conformity with the design certification. 10 CFR 52.97(b) requires that the COL identify the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP Section 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(b)(1) and 10 CFR 52.97(b) will be met, in part, by identifying ITAAC of the top-level design features of the overpressure protection system in the DC and COL applications, respectively.

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 components be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed.

2. GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. The overpressure protection system maintains RCS pressure within acceptable design limits during certain analyzed transients. Application of GDC 15 to the overpressure protection system provides assurance that the RCPB will have an extremely low probability of failure during transients.

3. GDC 30 requires that the reactor coolant pressure boundary be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. Application of GDC 30 to overpressure protection system provides assurance that the reactor coolant pressure boundary will have an extremely low probability of failure because of manufacturing or design defects.

4. GDC 31 requires in part that the RCPB is designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. During certain conditions in which the RCPB might behave in a brittle manner, the overpressure protection system maintains the RCS pressure below brittle fracture limits. Application of GDC 31 to the overpressure protection system provides assurance that the RCPB will have an extremely low probability of failure because of brittle fracture.

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. System Design. The staff uses the procedures below during the review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report (SAR) meet the acceptance criteria in Subsection II of this DSRS section.

   The procedures below are used to verify that the initial design criteria and bases have been appropriately implemented in the final design in the final SAR (FSAR). The ASME

Code requires the latter report, which is the basis during the review for many of the individual review steps outlined below.

The reviewer can use previously reviewed designs as a guide, but must verify that any differences are justified.

A. SMRs Operating at Power

i. The reviewer examines the NuScale SMR system description and schematics or piping and instrumentation diagrams, if applicable, to determine the number, type, and location of safety valves in both the primary and secondary systems and in discharge lines, instrumentation, and other components that provide overpressure protection. An optimum -capacity pressurizer in the NuScale design, combined with other system design features, could eliminate the need for a power operated relief valve (PORV) in the SMR primary system. The reviewer verifies that PORV’s are not needed.

ii. The reviewer identifies all other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems. This includes any blowdown or heat dissipation system connected to the discharge side of any pressure-relieving device. The reviewer determines the effects of these other functions or systems on operation of the overpressure protection system.

iii. The reviewer identifies the capacities, set points, and set point tolerances for all primary and secondary safety valve or other overpressure protection system devices. The reviewer verifies that these constraints are adequate to provide overpressure protection to the RCPB at critical values of pressure and temperature based on RCPB material parameters.

iv. The reviewer identifies allowable power levels with one or more inoperable main-steam line safety valve to ensure that they are suitably conservative.

v. The reviewer identifies all of the reactor trip signals that occur during overpressure transients, including their set points and set point tolerances. The reviewer verifies that the second reactor trip signal, under worst-case conditions during an overpressure transient, is adequate to provide overpressure protection to the RCPB in conjunction with the installed overpressure protection systems or devices.

vi. The reviewer examines all transients analyzed in the accident analysis section of the SAR that result in an increase in the pressure experienced by the RCPB. The reviewer identifies predicated peak pressures and assesses the operating conditions and set points used in the analysis to ensure that they are suitably conservative.

vii. The reviewer ensures that nonsafety-grade, pressure-operated relief valves are not credited for the mitigation of events that result in an increase in reactor coolant inventory.

viii. The reviewer ensures that relevant industry codes and classifications applied to the system analysis for power operations should be clearly identified as specified in Regulatory Issue Summary (RIS) 2004-04. Assumptions used in the analysis, including the initial plant conditions and system parameters, should also be identified and justified. The reviewer should identify studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics. The reviewer should consult Section 5.2.2 of the FSAR for the NuScale SMR design to obtain insight into the overpressure protection methodology.

B. <u>SMRs Operating at Low Temperature (Startup, Shutdown)</u>

i. The reviewer examines the NuScale SMR system description and schematics to determine the number, type, and location of the safety valves in the primary system and of discharge lines, instrumentation, and other components in interfacing systems.

ii. The reviewer examines failures of the CVCS or the normal shutdown cooling systems to ensure overpressure protection during low-temperature operation of the plant.

iii. The reviewer identifies the capacities, set points, and set point tolerances for all safety valves designated for low-temperature overpressure protection. The reviewer verifies that these constraints are adequate to provide overpressure protection to the RCPB at critical values of temperature based on RCPB material parameters.

iv. Relevant industry codes and classifications applied to the system analysis for low-temperature operation should be clearly identified as specified in RIS 2004-04. Assumptions used in the analysis, including the initial plant conditions and system parameters, should also be identified and justified. The reviewer should evaluate the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics. The reviewer should consult Section 5.2.2 of the FSAR for the NuScale design to obtain insight on the overpressure protection methodology.

4. Testing and Inspections. To ensure operational readiness, the overpressure protection system should be testable. The reviewer should verify the following:

   Tests for safety valve operability are scheduled to be conducted as specified in Section III of the ASME Code Article NB-7000.

5. Technical Specifications. The review includes the technical specifications to ensure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing. The review should include the following:

A. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable because of repairs and maintenance.

B. Verify that the frequency and scope of periodic surveillance testing are adequate.

C. Verify compliance with the technical specification guidance of Generic Letter No. 90-06 for the LTOP system.

D. Verify compliance with Action Plan Item II.K.3.3 of NUREG-0737, Clarification of TMI Action Plan Requirements, regarding reporting of safety valves challenges and failures. Generic Letter No. 82-16 provides descriptions of this NUREG-0737 item, include guidance regarding appropriate technical specifications to address the reporting requirements of II.K.3.3 of Section 5.6.4 of Standard Technical Specification NUREG-1430 regarding monthly operating reports, and offer related guidance on an appropriate technical specification to address this issue for those applicants implementing improved technical specifications.

E. Verify that appropriate references to the pressure-temperature limit report (PTLR) exist and that PTLR technical specification administrative controls are established as required by Generic Letter No. 96-03.

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 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).

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 staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's 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.

> The overpressure protection system, which includes the reactor safety valves, prevents overpressurization of the RCPB under the most severe transients and limits the reactor pressure during normal operational transients. The reactor safety valves discharge to the primary containment vessel. The reactor safety valves and the pressurizer capacity, in conjunction with the safety valves on the secondary side (if taken credit in the analysis), and the reactor protection and safeguards systems will protect the NuScale reactor coolant pressure boundary against overpressure in the event of a complete loss of heat sink. The NuScale ECCS RVVs also provide a means for overpressure protection as discussed in DSRS Section 6.3.
>
> The peak primary system pressure following the most severe anticipated transient is limited to the ASME Code allowable (110 percent of the design pressure) with no credit taken for nonsafety-grade relief systems. The assumption was that the NuScale plant is operating at design conditions (percent of rated power) and the NuScale reactor is shut down by a scram. The calculated pressure at the bottom of the vessel is ____kPag (___ psig), a value within the code allowable of ___ kPag (___ psig) (110 percent of vessel design pressure).
>
> Overpressure protection during low-temperature operation (defined in BTP 5-2) of the NuScale plant is provided by the reactor vent valves.
>
> In addition, the applicant has incorporated into the design the recommendations of Task Action Plan Items II.D.1 and II.D.3, as described in NUREG-0737, and has met the related requirements of 10 CFR 50.34(f)(2)(x), 10 CFR 50.34(f)(2)(xi), or 10 CFR 52.47(a)(8), as applicable.

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, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

3. 10 CFR Part 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

4. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

5. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

6. 10 CFR 50.34(f), "Additional TMI-related Requirements."

7. 10 CFR 52.47, "Contents of Applications; Technical Information."

8. 10 CFR 52.79, “Contents of Applications; Technical Information in Final Safety Analysis Report.”

9. 10 CFR 52.97, “Issuance of Combined Licenses.”

10. 10 CFR 50.55a, “Codes and Standards.”

11. ASME Boiler and Pressure Vessel Code, Section II, “Materials Specifications.”

12. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Facility Components.”

13. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, “Overpressure Protection,” American Society of Mechanical Engineers.

14. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7511.1, “Spring-Loaded Valves.”

15. Branch Technical Position (BTP) 5-2, “Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures.”

16. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

17. Deleted

18. NUREG-1511, Supplement 2, “Reactor Pressure Vessel Status Report.”

19. NRC Letter to All Pressurized Power Reactor Licensees, Generic Letter 82-16, “NUREG-0737 Technical Specifications,” September 20, 1982.

20. NRC Letter to All Pressurized Water Reactor Licensees and Construction Permit Holders, “Resolution of Generic Issue 70, ‘Power-Operated Relief-Valve and Block Valve Reliability,’ and Generic Issue 94, ‘Additional Low-Temperature Overpressure Protection for Light Water Reactors,’” Generic Letter No. 90-06, June 25, 1990.

21. NRC Regulatory Issue Summary 2004-04, “Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits,” RIS 2004-04, April 5, 2004.

22. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, “Relocation of the Pressure Temperature Limit Curves and Low-Temperature Overpressure Protection System Limits,” Generic Letter 96-03, January 31, 1996.

23. Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”

24. Regulatory Guide 1.29, “Seismic Design Classification.”

25. Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

26. Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

27. Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

28. Regulatory Guide 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52.”