U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

5.4 REACTOR COOLANT SYSTEM COMPONENT AND SUBSYSTEM DESIGN

REVIEW RESPONSIBILITIES

The organization responsible for review of reactor thermal-hydraulic systems in small modular reactors (SMR) has overall responsibility for the reviews performed under this Design-Specific Review Standard (DSRS) for the NuScale design. Primary and Secondary organizations responsible for the review of various components and subsystems associated with the reactor coolant system (RCS) are shown under Areas of Review below. RCS design bases, descriptions, evaluations, and necessary tests and inspections for the components or subsystems (including radiological considerations from the viewpoint of how radiation affects operation, and the viewpoint of how radiation levels affect the operators and capabilities of operation and maintenance) are to be evaluated for each of the specific Areas of Review below. Additional required evaluations are also specified under Areas of Review.

I. AREAS OF REVIEW

This section pertains to various components and subsystems within, or associated with, the RCS out to, and including, isolation valves. This is normally called the reactor coolant pressure boundary (RCPB), as defined in Title 10, Section 50.2(v), of the Code of Federal Regulations. The NuScale RCPB is reviewed as safety-related and risk-significant.

Principal components or subsystems might include the following:

- Steam generators (once-through helical coil)
- Reactor safety valves/low temperature overprotection valves
- Control rod drive mechanisms and instrument penetrations
- Main steamline flow restriction
- Pressurizer/Pressurizer relief discharge
- Chemical and volume control (CVCS) system
- RCS component supports
- Reactor vent valves (RVVs) and reactor recirculation valves (RRVs)
- RCS high-point vents
- Main steam lines and feedwater lines, including the decay heat removal (DHR) system

All RCS pipe penetrations such as feedwater for the steam generator or CVCS to process primary side water are above the top of the active fuel. RVVs and RRVs are attached directly to the reactor pressure vessel. Forged valve bodies are bolted directly to the reactor vessel and there is no piping between the valves and the vessel. The valves will be removable for inspection and repair. There will be valves at the reactor coolant pressure boundary that interface with the CVCS system.

ADAMS Accession No. ML15126A156

The specific areas of review, and interfaces with other DSRS sections, are provided below.

1 Once-through Helical Coil Steam Generators

A. The organization responsible for mechanical engineering reviews assesses methods of analysis (e.g. seismic and vibratory loadings, including those due to fluid flow and adverse flow conditions, and dynamic analyses), and the structural and functional integrity of steam generators under DSRS Sections 3.9.1, 3.9.2, and 3.9.3.

B. The organization responsible for the review of component integrity issues related to steam generator tubes evaluates the materials used to fabricate the steam generator, and the steam generator tube inservice inspection operational program. This evaluation of materials and design provisions is performed under DSRS Section 5.4.2.1, “Steam Generator Materials,” while the evaluation of the steam generator tube inservice inspection operational program is performed under DSRS Section 5.4.2.2, “Steam Generator Program.”

C. The organization responsible for the review of component integrity issues related to the reactor coolant pressure boundary (primary reviewer), and the organization responsible for component integrity issues related to the reactor vessel (secondary reviewer), assess the inservice inspection of the steam generator shell (i.e., the reactor pressure vessel) under DSRS Section 5.2.4.

D. The organization responsible for the review of transient and accident analyses for NuScale reviews the steam generator configuration and process design parameters and the response to various anticipated operational occurrences under DSRS Sections 15.1.1 - 15.1.4, “Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve,”.

E. The organization responsible for the review of transient and accident analyses for NuScale (primary reviewer) and the organization responsible for transient and accident analyses (secondary reviewer) assess the steam generator response to various anticipated operational occurrences and postulated accidents under DSRS 15.1.5, “Steam System Piping Failures Inside and Outside of Containment (PWR),”.

F. The organization responsible for the review of transient and accident analyses for NuScale evaluates the effects of a postulated feedwater line break under DSRS 15.2.8, “Feedwater System Pipe Break Inside and Outside Containment (PWR),”.

G. The organization responsible for the review of transient and accident analysis for NuScale (primary reviewer) and the organization responsible for the review of atmospheric dispersion estimates (secondary reviewer) assess the steam generator response to various postulated accidents under DSRS Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses For Advanced Light Water Reactors,”.

2. <u>Reactor Coolant System Piping and Valves (RVVs and RRVs)</u>

A. The organization responsible for review of reactor thermal-hydraulic systems in NuScale reviews the system description and schematics, process flow features, and equipment arrangements of RCS piping and valves.

B. The organization responsible for mechanical engineering reviews assesses the design and analysis, and the preoperational testing, of the RCS piping and valves under SRP Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports."

C. The organization responsible for review of component integrity issues related to reactor coolant pressure boundary (primary reviewer), the organization responsible for review of component integrity issues related to reactor vessels (secondary reviewer), and the organization responsible for review of chemical engineering issues (secondary reviewer); assess reactor coolant system piping and valves and materials, under Standard Review Plan (SRP) Section 5.2.3.

D. The organization responsible for the review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of component integrity issues related to reactor vessels (secondary reviewer) evaluate periodic inspection and testing of ASME Code Class 1 RCS piping and valves (other than steam generator tubes) to assess their structural and leaktight integrity. These evaluations are performed under DSRS Section 5.2.4.

E. The organization responsible for the review of reactor coolant pressure boundary leakage detection reviews design features, analytical techniques, and procedural measures associated with reactor coolant pressure boundary leakage detection systems under DSRS Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection."

F. The organization responsible for the review of thermal-hydraulic systems assesses the functional aspects of valves within, and connected to, the reactor coolant pressure boundary under DSRS Section 5.4.7, "Decay Heat Removal (DHR) System."

G. The organization responsible for the review of thermal-hydraulic systems in the NuScale chemical and volume control system assesses the functional aspects of valves within, and connected to, the reactor coolant pressure boundary under DSRS Section 9.3.4, "Chemical and Volume Control System."

H. The organization responsible for the review of the NuScale emergency core cooling system (primary reviewer) and the organizations responsible for the review of other systems and technical areas related to the NuScale emergency core cooling system assess the functional aspects of valves within, and connected to, the reactor coolant pressure boundary under DSRS Section 6.3, "Emergency Core Cooling System."

I. The organization responsible for mechanical engineering reviews assesses methods of analysis (e.g., seismic and vibratory loadings, and dynamic analyses), and structural integrity of RCS (and subsystem) valves under SRP Sections 3.9.1, 3.9.2, and 3.9.3.

J. The organization responsible for the review of component performance and testing reviews the functional design and qualification provisions and IST programs for certain safety-related valves typically designated as Class 1, 2, or 3 under Section III of the ASME Code under SRP Section 3.9.6. The review may include other valves not categorized as ASME Code Class 1, 2, or 3 if the staff considers them to be safety related.

K. The organization responsible for mechanical engineering reviews (primary reviewer) and the organization responsible for reviews of electrical engineering and instrumentation and controls (secondary reviewer) assess the methods of test and analysis used to ensure the function of valves (including valve operators), associated electrical equipment, and instrumentation and controls, under the full range of normal and accident loadings (including seismic). These assessments are conducted under SRP Section 3.10.

L. The organization responsible for review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of materials engineering issues related to flaw evaluation and welding (secondary reviewer) assess materials of fabrication for RCS valves under SRP Section 10.3.6, "Steam and Feedwater System Materials."

M. The organization responsible for review of component integrity issues related to engineered safety features evaluates the compatibility of fluids with valve materials under DSRS Section 6.1.1, "Engineered Safety Features Materials."

N. The organization responsible for review of the inspection, testing, evaluation, and repair of mechanical equipment and components assesses the inservice inspection of ASME Code Class 2 and Class 3 valves under DSRS Section 6.6, "Inservice Inspection of Class 2 and 3 Components."

O. The organization responsible for mechanical engineering reviews assesses methods of analysis (e.g., seismic and vibratory loadings, and dynamic analyses) and structural integrity of the RVVs and the RRVs under SRP Sections 3.9.1, 3.9.2, and 3.9.3.

3. <u>Pressurizer</u>

A. The organization responsible for review of reactor thermal-hydraulic systems in NuScale reviews the configuration and process design parameters of the pressurizer, including related safety and relief valve capacities, under DSRS Section 5.2.2, "Overpressure Protection Review Responsibilities."

B. The organization responsible for the review of transient and accident analyses for NuScale reviews pressurizer system performance during anticipated operational occurrences and postulated accidents under DSRS 15.6.1, “Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve,” and other various DSRS Sections of Chapter 15.

C. The organization responsible for mechanical engineering reviews evaluates the structural integrity of the pressurizer and methods of analysis under SRP Sections 3.9.1, 3.9.2, and 3.9.3.

D. The organization responsible for the review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of component integrity issues related to reactor vessels (secondary reviewer) evaluate periodic inservice inspection and testing of pressurizers to assess their structural and leaktight integrity. These evaluations are performed under DSRS Section 5.2.4.

E. The organization responsible for review of component integrity issues related to reactor coolant pressure boundary (primary reviewer), the organization responsible for review of component integrity issues related to reactor vessels (secondary reviewer), and the organization responsible for review of chemical engineering issues (secondary reviewer) assess pressurizer materials under SRP Section 5.2.3.

4. <u>RCS High-Point Vents</u>

The organization responsible for the review of reactor thermal-hydraulic systems reviews the RCS high-point vents under SRP 5.4.12, “Reactor Coolant System High Point Vents.”

5. <u>Chemical and Volume Control System (CVCS) System</u>

The organization responsible for review of reactor thermal hydraulic systems in NuScale reviews the RTNSS function of the CVCS under DSRS Section 9.3.4.

6. <u>Reactor Safety Valves</u>

A. The organization responsible for review of reactor thermal-hydraulic systems in NuScale reviews setpoints and capacities of RCS safety and relief valves, and low-temperature overpressure protection systems / cold overpressure mitigation systems under DSRS Section 5.2.2.

B. The organization responsible for mechanical engineering reviews evaluates the structural integrity and methods of analysis of RCS pressure relief devices/ reactor coolant depressurization systems under SRP Sections 3.9.1, 3.9.2, and 3.9.3.

C. The organization responsible for the review of component performance and testing reviews the functional design and qualification provisions and IST programs for RCS pressure relief devices / reactor coolant depressurization systems typically designated as Class 1, 2, or 3 under Section III of the ASME Code under DSRS Section 3.9.6.

D. The organization responsible for mechanical engineering reviews (primary reviewer) and the organization responsible for reviews of electrical engineering and instrumentation and controls (secondary reviewer) assess the methods of test and analysis used to ensure the operability of safety and relief valves and associated electrical equipment, and instrumentation and controls, under the full range of normal and accident loadings (including seismic). These assessments are conducted under SRP Section 3.10.

E. The organization responsible for the review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of component integrity issues related to reactor vessels (secondary reviewer) evaluate the inservice inspection of ASME Code Class 1 RCS pressure relief devices / reactor coolant depressurization systems to assess their structural and leaktight integrity. These evaluations are performed under DSRS Section 5.2.4.

F. The organization responsible for review of the inspection, testing, evaluation, and repair of mechanical equipment and components assesses the inservice inspection of ASME Code Class 2 and Class 3 components under DSRS Section 6.6.

G. The organization responsible for review of component integrity issues related to reactor coolant pressure boundary (primary reviewer), the organization responsible for review of component integrity issues related to reactor vessels (secondary reviewer), and the organization responsible for review of chemical engineering issues (secondary reviewer), assess RCS pressure relief device/ reactor coolant depressurization system materials under SRP Section 5.2.3.

H. The organization responsible for review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of materials engineering issues related to flaw evaluation and welding (secondary reviewer) assess the materials of fabrication for RCS pressure relief devices / reactor coolant depressurization systems under DSRS Section 10.3.6. These reviewers also evaluate the compatibility of fluids with valve materials under DSRS Section 6.1.1.

I. The organization responsible for the review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of component integrity issues related to reactor vessels (secondary reviewer) evaluate the inservice inspection program for ASME Code Class 1 RCS pressure relief devices / reactor coolant depressurization systems to assess their structural and leaktight integrity. These evaluations are performed under DSRS Section 5.2.4.

J.

7. <u>Reactor Coolant System Component Supports</u>

A. The organization responsible for mechanical engineering reviews assesses methods of analysis (e.g. seismic and vibratory loadings and dynamic analyses) and structural integrity of RCS (and subsystem) component supports under SRP Sections 3.9.1, 3.9.2, and 3.9.3.

B. The organization responsible for the review of component performance and testing reviews the functional design and qualification provisions and IST programs for certain safety-related dynamic restraints (snubbers) typically designated as Class 1, 2, or 3 under Section III of the ASME Code under SRP Section 3.9.6. The review may include other dynamic restraints not categorized as ASME Code Class 1, 2, or 3 if the staff considers them to be safety related.

8. Main Steam Lines and Feedwater Piping

A. The organization responsible for the review of power conversion systems reviews the functional and related requirements for the main steam line piping under DSRS Section 10.3, "Main Steam Supply System."

B. The organization responsible for the review of power conversion systems reviews the functional and related requirements for feedwater piping under DSRS Section 10.4.7, "Condensate and Feedwater System."

C. The organization responsible for mechanical engineering reviews assesses the design and analysis and preoperational testing of the main steam line and feedwater piping under SRP Section 3.12.

D. The organization responsible for review of component integrity issues related to reactor coolant pressure boundary (primary reviewer) and the organization responsible for the review of materials engineering issues related to flaw evaluation and welding (secondary reviewer) assess the materials of fabrication for main steam line and feedwater piping under SRP Section 10.3.6.

E. The organization responsible for mechanical engineering reviews evaluates potential adverse flow effects on reactor, steam, feedwater, and condensate systems resulting from hydrodynamic loads, acoustic pressure fluctuations, and vibrations in accordance with the guidance in SRP Sections 3.9.2 and 3.9.5, "Reactor Pressure Vessel Internals."

9. Main Steam Line Flow Restrictions

A. The organization responsible for mechanical engineering reviews assesses methods of analysis (e.g., seismic and vibratory loadings, and dynamic analyses), and structural and functional integrity of the main steam line flow restrictions under SRP Sections 3.9.1, 3.9.2, and 3.9.3.

B. The organization responsible for the review of transient and accident analyses for PWRs/BWRs reviews the functional requirements of the main steam line flow restrictions under DSRSP Section 15.1.5.

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 DSRS Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” and DSRS Section 14.3.4, “Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria.” 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 DSRS Section 14.3.

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.

II. ACCEPTANCE CRITERIA

Requirements

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

1. Specific requirements are identified in the applicable DSRS sections.

2. 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 plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

3. 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 Atomic Energy Act, and the NRC's regulations.

4. 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.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas/components identified in Section I are included in the individual DSRS sections of the components.

III. REVIEW PROCEDURES

For each area of review specified in Subsection I of this DSRS section, the review procedure is contained in the specified DSRS section. 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, and in the specified DSRS sections.

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. 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) meet 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.

4. 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).

5. For review of both DC and COL applications, DSRS 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 technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the Standard Review Plan (SRP) revision in effect six months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed small modular reactor (SMR) designs, however, differ significantly from large light-water nuclear reactor power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405) (SRM). In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated pre-application activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for the evaluation of a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design or siting assumptions.

VI. REFERENCES

None.