**U.S. NUCLEAR REGULATORY COMMISSION**

# DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

## 4.5.2 REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS

**REVIEW RESPONSIBILITIES**

**Primary -** Organization responsible for review of component integrity issues related to reactor vessel internals

**Secondary -** None

I. AREAS OF REVIEW

Section 50.55a, "Codes and Standards," of Title 10 of the *Federal Code of Regulations* (10 CFR) Part 50, and General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 require that structures, systems, and components (SSCs) important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed. The purpose of this design specific review standard (DSRS) section is to review and evaluate the adequacy of the materials selected for the construction of the reactor internal and core support structures, and to assure that the reactor internal and core support structures meet these regulations. The reactor internal and core support structures reviewed under this DSRS section include all structures and components as defined in NG-1120 of Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"). In addition, for the NuScale design, any structure or component within the reactor vessel that is either a reactor coolant pressure boundary, or has emergency core cooling or safe shutdown capability, shall be treated as such according to its safety function, and not as a reactor internal. This includes, but is not limited to the steam generator (including tubes and tubesheet), riser, and control rod drives. Any supporting structures for these components, for example, control rod guide supports, guide tubes, baffle plates, etc. may be considered reactor internals/core supports. The reactor internal and core support structure forms the safety related part of a system.

The NuScale reactor internal and core support structure materials include the following classifications of equipment:

1. Safety-related, risk-significant equipment

2. Safety-related, nonrisk-significant equipment

3. Nonsafety-related, risk-significant Regulatory Treatment of Nonsafety Systems (RTNSS) equipment

4. Nonsafety-related, nonrisk-significant equipment.

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The NuScale application will include the classification of SSCs, a list of risk-significant SSCs, and a list of RTNSS equipment. Based on this information, the staff will review according to NUREG-0800, Standard Review Plan (SRP) Section 17.4, Reliability Assurance Program," 19.3, "Regulatory Treatment of Non-Safety Systems (RTNSS) for Passive Advanced Light Water Reactors" and DSRS Section 3.2.2, "System Quality Group Classifications," to confirm the determination of safety-related and risk-significant SSCs.

This DSRS section covers the material, component design, fabrication and inspection to assure structural integrity in compliance with Section 50.55a and General Design Criterion 1 of Appendix A to 10 CFR Part 50.

The following areas in the applicant's safety analysis report (SAR) relating to reactor internal and core support structure materials are reviewed; specific areas of review are as follows:

1. Materials. The review includes the acceptability of the materials, including weld materials, to be used for the reactor internals and core support structures.

   The adequacy and suitability of the materials specified for the reactor internals and core support structures are reviewed in terms of their fracture toughness, stress corrosion resistance, fabricability, and other mechanical properties.

2. Controls on Welding. The review includes the controls on welding for reactor internals and core support structures.

3. Nondestructive Examination. The review includes information submitted by the applicant on the nondestructive examination procedures used for inspection of each product form.

4. Austenitic Stainless Steel. Austenitic stainless steels are primarily used for the construction of the reactor internals and core support structures. These steels may be used in a variety of product forms, including several stabilized product forms. Unstabilized austenitic stainless steels, such as Types 304 and 316, are frequently specified.

   Since unstabilized compositions are susceptible to stress corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid sensitization of the material, and to minimize exposure of the stainless steel to contaminants that lead to stress corrosion cracking. The review includes information submitted by the applicant as described in DSRS Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

5. Other Materials. Materials other than austenitic stainless steels are reviewed and evaluated in terms of their fracture toughness, corrosion resistance, fabricability, and other mechanical properties.

6. 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 SSCs related to this DSRS section in accordance with 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 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.

7. 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. The review of the adequacy of programs for assuring the integrity of bolting and threaded fasteners is performed under DSRS Section 3.13, "Threaded Fasteners - ASME Code 1, 2, and 3."

2. The evaluation of corrosion and compatibility of reactor internals and core support structure materials with the expected environment during service is performed using the guidance in DSRS Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

3. The review of acceptability of the reactor coolant chemistry and associated chemistry controls (including additives such as inhibitors) as it relates to corrosion control and compatibility with materials to be exposed to reactor coolant is performed under DSRS Section 9.3.6, "Reactor Coolant Inventory and Purification System."

4. The review of reactor internal and core supports with respect to dynamic testing and analysis is performed under SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components." Guidance is provided in Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," for the vibration assessment program to verify the structural integrity of reactor internals/core supports due to flow-induced vibration. This review may include other components and structures in the reactor coolant flow path.

5. The review of the adequacy of design fatigue curves for reactor internals and core support structures materials with respect to cumulative reactor service-related environmental and usage factor effects and consideration of each combination of loadings is performed under DSRS Sections 3.9.1, "Special Topics for Mechanical Components," and 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

6. The review of the reactor internals and core support structures with respect to their mechanical design adequacy to withstand design and service loading combinations is performed under DSRS Section 3.9.5, "Reactor Pressure Vessel Internals."

7. The review of the plant design, including the selection of materials to minimize activation products, to verify that occupational radiation exposures will be as low as is reasonably

achievable (ALARA) is performed under DSRS Section 12.1, “Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable.”

8. The review of operational programs required by regulations for COL application, including inservice and preservice inspection and reactor vessel material surveillance performed under DSRS Section 13.4, “Operational Programs.”

9. Review of the Probabilistic Risk Assessment is performed under SRP Section 19.

## II. ACCEPTANCE CRITERIA

### Requirements

The design, fabrication, and testing of the materials used in the reactor internals and core support structures are reviewed and evaluated to meet codes and standards commensurate with the safety functions to be performed such that the relevant requirements of 10 CFR 50.55a and GDC 1 are met.

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

1. 10 CFR 50.55a, "Codes and Standards,” which requires that SSCs shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," which requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. GDC 1 also requires that appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

3. 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 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 regulations.

4. 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 (AEA), and the U.S. Nuclear Regulatory Commission’s (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.

1. Materials. For core support structures and reactor internals, the permitted material specifications are those given in the ASME Code, Section III, Division 1, Subsubarticle NG-2120. The properties of these materials are specified in Tables 2A, 2B and 4 of Section II, Part D, Subpart 1, of the Code for Class components.

   Additional permitted materials and their applications are identified in ASME Code Cases approved for use as described in RG 1.84, "Design, Fabrication, and Material Code Case Acceptability, ASME Code, Section III."

2. Controls on Welding. Methods and controls for core support structures and reactor internals welds shall be in accordance with ASME Code, Section III, Division 1, Article NG-4000. The examination requirements and acceptance criteria for these welds are specified in Article NG-5000.

3. Nondestructive Examination. Nondestructive examinations shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-2500. The nondestructive examination acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-5300.

4. Austenitic Stainless Steels. The acceptance criteria for this area of review are given in DSRS Section 5.2.3, Subsections II.2 and II.4.a, b, d, and e.

   RG 1.44 provides acceptance criteria for preventing intergranular corrosion of stainless steel components. In conformance with this guide, furnace-sensitized material should not be allowed. Methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding.

5. Other Materials. All materials used for reactor internals and core support structures must be selected for compatibility with the reactor coolant, as specified in Subsubarticles NG-2160 and NG-3120 of Section III, Division 1 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be selected and specified to provide assurance that these materials will not deteriorate in service. For example, acceptable heat treatment temperatures are 565°C - 595°C (1050°F - 1100°F) for aging of Type 17-4 PH and 565°C (1050°F) for tempering of Type 410 stainless steel.

   Other materials shall have similar appropriate heat treat and fabrication controls in accordance with strength and compatibility requirements.

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:

> GDC 1 and 10 CFR 50.55a require that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. 10 CFR 50.55a also incorporates by reference the applicable editions and addenda of the ASME Code. The reactor internals and core support structures include SSCs that perform safety functions and/or whose failure could affect the performance of safety functions by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the primary reactor coolant system). Application of 10 CFR 50.55a and GDC 1 to the materials of construction provides assurance that established standard practices of proven or demonstrated effectiveness for selecting materials, fabrication, and testing/ inspection of SSCs are used to achieve a high likelihood that these safety functions will be performed.

## 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, "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 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. Materials. The list of the materials for reactor internals and core support structures that are exposed to the reactor coolant is reviewed.

   The materials identified for each component or part used in the reactor internals and core support structures are compared with the materials identified as being acceptable in Sections II and III of the ASME Code and/or acceptable ASME Code Cases identified in Regulatory Guide 1.84, as described in the acceptance criteria. The reviewer verifies that any exceptions to the ASME Code-specified materials are clearly identified. The reviewer evaluates the basis for the exceptions, taking into account precedents set in earlier cases, and determines the acceptability of such materials.

4. Controls on Welding. The reviewer verifies that welding methods and controls for the reactor internals and core support structures are in accordance with the procedures of ASME Code, Section III, Division 1, Article NG-4000. The reviewer verifies that welding controls submitted by the applicant are in conformance with the welding controls in DSRS Section 5.2.3, which are also considered applicable to welding of reactor internals. The reviewer assures that any special welding processes or welding controls conform to the qualification requirements of ASME Code, Section IX, or that justification is made for any deviation.

5. Nondestructive Examination. The information submitted by the applicant is reviewed to determine methods used for nondestructive examination. The reviewer verifies that the nondestructive examination methods proposed by the applicant are in conformance with

the examination methods specified by the ASME Code. Section III, Division 1, Subarticle NG-2500 of the ASME Code specifies that examination by either radiographic or ultrasonic examination plus surface examinations as required is acceptable.

6. Austenitic Stainless Steel. The materials and fabrication procedures used for reactor internals are reviewed. The areas of review and review procedures include those described in DSRS Section 5.2.3. The reviewer verifies that environmental conditions are controlled and welding procedures are developed such that the probabilities of sensitization and microfissuring are minimized. DSRS Section 4.5.1, "Control Rod Drive Structural Materials," Subsection III.2, identifies an acceptable alternate to the methods described in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," for verifying the degree of sensitization that occurs during welding. In addition, the reviewer verifies that materials are selected to assure compatibility with the compositions of the reactor coolant, and that the fabrication and cleaning controls imposed on stainless steel components are adequate to prevent contamination with chloride and fluoride ions.

   Where cast austenitic stainless steels are proposed for use, the reviewer verifies that, under the expected environmental conditions, the selected material will provide adequate fracture toughness over its design life (e.g., considering thermal aging due to exposure to reactor coolant operating temperatures and the effects of radiation).

7. Other Materials. The reviewer verifies that the heat treatment and welding controls provided in the material specifications and fabrication procedures are appropriate for the material. The reviewer verifies that the fabrication and cleaning controls will preclude contamination of nickel-base alloys by chloride ions, fluoride ions, or lead.

   Operating experience has indicated that certain nickel-chromium-iron alloys (e.g. Alloy 600 and associated weld materials, Alloy 82 and 182) are susceptible to stress corrosion cracking, as documented in NUREG-1823 and NRC Generic Letter 97-01. Alloy 690, and associated weld materials Alloy 52 and 152, have improved corrosion resistance in comparison to Alloy 600, which is used in pressurized-water reactor (PWR) coolant applications. Where nickel-chromium-iron alloys are proposed for use in PWR reactor coolant applications, use of Alloy 690 materials is preferred. If Alloy 600 material is proposed, the reviewer must be able to verify that an acceptable technical basis is either identified (based upon demonstrated satisfactory use in similar applications) or presented by the applicant to support use of the material under the expected environmental conditions (e.g., exposure to the reactor coolant), in order for Alloy 600 material to be acceptable. In addition, the reviewer should verify whether acceptable augmented inspection requirements have been proposed based on operating experience and service conditions. For all reactor coolant environments, particular review emphasis is placed upon the corrosion resistance and stress corrosion cracking resistance properties of the proposed nickel-chromium-iron alloy(s).

8. 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 technical submittal meets the acceptance criteria. DCs have referred to the technical submittal 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 technical

submittal.

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).

9. 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 technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the SER. The reviewer also states the bases for those conclusions.

1. The staff concludes that the materials used for the reactor internals and core support structures are acceptable and meet the requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, General Design Criterion 1. This conclusion is based upon the following considerations:

   The applicant has selected, and identified by specification, materials for the reactor internals and core support structures that satisfy the requirements of Subsubarticle NG-2120 of Section III, Division 1 and Tables 2A, 2B and 4 of Section II of the ASME Code. For materials not in accordance with ASME Code provisions, the applicant has selected materials of construction that are approved for use by NRC-accepted ASME Code Cases, as identified in RG 1.84, or that have otherwise been demonstrated acceptable for the application. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

   The applicant has demonstrated that the design, fabrication, and testing of the materials used in the reactor internals and core support structures are of high quality standards and are adequate to assure structural integrity. The controls imposed upon austenitic stainless steel components satisfy the positions of RG 1.44, and the related criteria provided in DSRS Section 5.2.3.

   The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internals and core support structures will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

   The material selection, fabrication practices, examination and testing procedures, and control practices provide reasonable assurance that the materials used for the reactor internals and core support structures will be in a metallurgical condition that will preclude inservice deterioration.

   Conformance with relevant requirements of the ASME Code, or accepted Code Cases, and the recommendations of RG 1.31 and 1.44 and the related criteria in DSRS Section

5.2.3, constitutes an acceptable basis for meeting the relevant requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1.

2. 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.

3. 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. <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 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 50.55a, "Codes and Standards."

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants," General Design Criterion 1, "Quality Standards and Records."

3. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

4. RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

5. RG 1.44, "Control of the Use of Sensitized Stainless Steel."

6. RG 1.84, "Design, Fabrication, and Material Code Case Acceptability, ASME Section III."

8. RG 1.215, Guidance for ITAAC Closure under 10 CFR Part 52.

9. ASME Boiler and Pressure Vessel Code, Section II, "Materials," Tables 2A, 2B and 4; Section III, "Rules for Construction of Nuclear Facility Components," Division 1; and Section IX, "Welding and Brazing Qualifications." American Society of Mechanical Engineers.

10. ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, "Core Support Structures."

11. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plants."

12. NUREG-1823, "U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials." U.S. Nuclear Regulatory Commission. Washington, DC. April 2005.

13. NRC Letter to All Licensees of Pressurized Water Reactors (PWRs), "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations" (Generic Letter 97-01). April 1, 1997 (ML031110036).