5 REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 5, "Reactor Coolant System and Connecting Systems," of the NuScale Power, LLC (NuScale or applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report (FSAR)," Revision 2, issued October 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18310A326).

5.1 <u>Introduction</u>

The reactor coolant system (RCS) provides for the circulation of the primary coolant. The applicant's design relies on natural circulation flow for the reactor coolant and does not include reactor coolant pumps or an external piping system. The RCS is a subsystem of the NuScale Power Module (NPM). The RCS includes the reactor vessel (RV) and integral pressurizer (PZR), the reactor vessel internals (RVIs), the reactor safety valves (RSVs), RCS piping inside the containment vessel (CNV) (RCS injection, RCS discharge, PZR spray supply, and RV high-point degasification lines), the PZR control cabinet, and the RCS instruments and cables.

5.2 <u>Integrity of Reactor Coolant Boundary</u>

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with the Codes and Standards Rule, 10 CFR 50.55(a)

5.2.1.1.1 Introduction

This SER section addresses the use of acceptable codes (i.e., American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (ASME BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code)), code editions, and addenda required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and Standards," in the design certification (DC) for the NuScale Power Plant.

5.2.1.1.2 Summary of Application

DCA Part 2, Tier 1: The DCA Part 2, Tier 1, descriptions of several systems indicate that the systems and components will conform to the rules of construction of ASME BPV Code, Section III.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 5.2.1.1, "Compliance with 10 CFR 50.55a," discusses the Class 1 components of the NPM meeting the requirements of 10 CFR 50.55a and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality Standards and Records," and GDC 30, "Quality of Reactor Coolant Pressure Boundary." The applicant stated the ASME BPV Code of record as the 2013 Edition with no addenda. The applicant referenced DCA Part 2, Tier 2, Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," and Section 6.6, "Inservice Inspection and Testing of Class 2 and 3 Systems," for discussion of inservice inspection (ISI) requirements for satisfying the requirements in ASME BPV Code, Section XI. Additionally, the applicant discussed operational and maintenance inservice testing (IST) codes, standards, and guides as being in accordance with the ASME OM Code, 2012 Edition. SER Section 3.9.6 discusses the review of this item.

Inspections, Tests, Analyses, and Acceptance Criteria: DCA Part 2, Tier 1, addresses the proposed inspections, tests, analyses, and acceptance criteria (ITAAC), as required by 10 CFR 52.47(b)(1), based on the selection criteria in DCA Part 2, Tier 2, Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria."

Technical Specifications: There are no technical specifications (TS) for this area of review.

Technical Reports: There are no technical reports (TRs) for this area of review.

5.2.1.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 1, as it relates to the requirement that systems, structures, and components (SSCs) be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed

- 10 CFR 52.47(b)(1), as it relates to a DCA containing 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 plant that incorporates the DC is built and will operate in accordance with the DC; the provisions of the Atomic Energy Act of 1954, as amended (AEA); and the NRC's regulations

- 10 CFR 50.55a, as it relates to establishing minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of reactor coolant pressure boundary (RCPB) components and other fluid systems by complying with appropriate editions of published industry codes and standards and confirming that, according to 10 CFR 50.55a, components important to safety are subject to the following requirements:

  – that RCPB components must meet the requirements for Class 1 (Quality Group A) components specified in the ASME BPV Code, Section III, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2)

  – that components classified as Quality Groups B and C must meet the requirements for Class 2 and 3 components, respectively, specified in ASME BPV Code, Section III

Section 5.2.1.1, "Compliance with the Codes and Standards Rule, 10 CFR 50.55a," of NUREG-0800, Revision 4, (12/2016), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), lists the acceptance criteria adequate to meet the above requirements and includes review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- Regulatory Guide (RG) 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," as it relates to determining quality standards acceptable to the staff for satisfying GDC 1 for other (i.e., non-RCPB) safety-related components containing water, steam, or radioactive material in light-water-cooled nuclear power plants

- other system-specific acceptance criteria listed in SRP Section 5.2.1.1

5.2.1.1.4 Technical Evaluation

5.2.1.1.4.1 Compliance with GDC 1 and 10 CFR 50.55a.

GDC 1 requires that SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. For those SSCs defined as safety related, the NRC regulations specify special treatment requirements to provide reasonable assurance of the capability of those SSCs to perform their safety-related functions. One special treatment requirement is that applicable components meet the requirements in the ASME BPV Code and OM Code as incorporated by reference in 10 CFR 50.55a.

5.2.1.1.4.1.1 Code of Record

The staff has reviewed DCA Part 2 for compliance with the requirements presented above. DCA Part 2, Tier 2, Section 5.2.1.1, states that the code of record for the NuScale design is the 2013 Edition with no addenda, which has been incorporated by reference in 10 CFR 50.55a and is therefore acceptable to the staff.

5.2.1.1.4.1.2 ASME BPV Code Class 1, 2, and 3 Component Design Requirements and Inservice Inspection and Testing Requirements

The NRC regulations in 10 CFR 50.55a require that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III of the ASME BPV Code as incorporated by reference in 10 CFR 50.55a. This regulation also requires that pressure-retaining components of other fluid systems designated as ASME BPV Code Class 2 or Class 3 components meet the applicable requirements of the ASME BPV Code as incorporated by reference in 10 CFR 50.55a. Components within the scope of 10 CFR 50.55a are subject to ISI and IST in accordance with ASME BPV Code, Section XI, and ASME OM Code, respectively, and must be designed and provided with access to enable the performance of IST and ISI, as required in 10 CFR 50.55a(f) and 10 CFR 50.55a(g). The requirements of GDC 1 on quality standards are met by acceptable application of quality group classifications and quality standards. RG 1.26 describes a quality classification system that may be used to determine quality standards acceptable to the NRC staff for satisfying GDC 1 for ASME BPV Code Class 2 and 3 components. The staff evaluates quality group classifications and quality standards as part of the review discussed in SER Section 3.2.2.

5.2.1.1.4.1.3 Pressure Boundary Definition

DCA Part 2, Tier 2, Section 5.2, “Integrity of Reactor Coolant Boundary,” states that the RCPB for each NPM is consistent with the RCPB definition in 10 CFR 50.2, “Definitions.” The staff reviewed the proposed definition of the RCPB and finds it consistent with the definition in 10 CFR 50.2.

5.2.1.1.4.1.4 Clarification of Codes and Standards Requirements

Regulations in 10 CFR 50.55a(d) and 10 CFR 50.55a(e) provide requirements for Quality Group B and C components. DCA Part 2, Tier 2, Section 5.2.1.1, discusses meeting 10 CFR 50.55a, GDC 1, and GDC 30 for RCPB components designated as Class 1 (which the staff finds to be acceptable) but does not discuss Quality Group B or Quality Group C components. In DCA Part 2, Tier 2, Sections 3.2.2.1, 3.2.2.2, and 3.2.2.3 (Quality Group A, Quality Group B, and Quality Group C, respectively), the applicant states that Quality Group A, B, and C SSCs met the applicable conditions in 10 CFR 50.55a(b), in addition to the requirements of the appropriate ASME BPV Code, Section III, Class. The staff confirmed this information and finds that this satisfies the requirements of 10 CFR 50.55a(b), 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e).

The staff notes that existing combined license (COL) items in SRP Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints”; Section 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing”; and Section 6.6, “Inservice Inspection and Testing of Class 2 and 3 Components,” address the applicant’s compliance with an ISI/IST program in accordance with the ASME OM Code and ASME BPV Code, Section XI. The corresponding sections in this SER further discuss these topics.

5.2.1.1.4.1.5 Proposed Alternatives

In 10 CFR 50.55a(z), the NRC permits the use of alternatives to the requirements of 10 CFR 50.55a. The alternatives must be submitted and authorized before implementation. DCA Part 2, Tier 2, Section 5.2.1.1, does not discuss any proposed alternatives to compliance with 10 CFR 50.55a. In the applicant’s response dated August 31, 2017, to Request for Additional Information (RAI) 8914, Question 05.02.01.01-6 (ADAMS Accession No. ML17244A121), the applicant indicated that there are no exceptions to the ASME BPV Code for the RCPB. RAI 9335 was subsequently generated to supplement RAI 8914. In its response dated April 2, 2018, to RAI 9335, Question 05.02.01.01-7 (ADAMS Accession No. ML18092B091), the applicant noted that it would have identified any proposed alternatives to 10 CFR 50.55a under the “Augmented Design Requirements” column in DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structures, Systems, and Components,” and that there are, therefore, no proposed alternatives to compliance with 10 CFR 50.55a for Quality Group B and C components in the NPM. This clarification resolved the staff’s concern, as the responses to RAI 8914, Question 05.02.01.01-6, and RAI 9335, Question 05.02.01.01-7, collectively state that there are no proposed alternatives to the ASME BPV Code for Quality Group A, B, or C components. Based on the above response, and the incorporation of the response to RAI 8914, Question 05.02.01.01-6, in Revision 1 of DCA Part 2, the staff considers RAI 9335, Question 05.02.01.01-7, resolved and closed. SER Section 3.9.6 discusses relief and alternative requests to the ASME OM Code.

5.2.1.1.4.1.6 Inspections, Tests, Analyses, and Acceptance Criteria

The guidance in 10 CFR 52.47(b)(1) requires that a DCA contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the AEA, and the NRC's regulations. SRP Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(b)(1) will be met, in part, by identifying ITAAC for the top-level design features related to compliance with 10 CFR 50.55a.

DCA Part 2, Tier 1, addresses the proposed ITAAC, as required by 10 CFR 52.47(b)(1), based on the selection criteria in DCA Part 2, Tier 2, Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria.” The application contains ITAAC that verify compliance with ASME BPV Code, Section III, requirements for ASME BPV Code Class 1, 2, and 3 piping systems through inspection of ASME BPV Code, Section III, design reports for as-built piping systems. Other ITAAC verify that ASME BPV Code Class 1 and 2 and core support (CS) components conform to the rules of construction of ASME BPV Code, Section III, through inspection of the ASME BPV Code, Section III, data reports. As further discussed in SER Section 14.3.3, the applicant has adopted ITAAC from the standardized ITAAC guidance, specifically those for compliance with ASME BPV Code, Section III. As these ITAAC are aligned with the staff-approved standardized ITAAC guidance, these ITAAC are considered acceptable.

5.2.1.1.5 Combined License Information Items

There are no COL information items for this section.

5.2.1.1.6 Conclusion

The staff concludes that the applicant has met the requirements of 10 CFR 50.55a, specific to the RCPB, for the construction of SSCs important to safety to quality standards by ensuring that RCPB components, as defined by 10 CFR 50.55a, are classified properly as ASME BPV Code, Section III, Class 1 (Quality Group A) components. The staff concludes that the applicant has met the 10 CFR 50.55a requirements by properly specifying that non-RCPB components designated as Quality Group B and Quality Group C components are constructed as ASME BPV Code, Section III, Class 2 and 3 components, respectively. SER Section 3.2.2 documents the staff’s review of quality group classifications for components of safety-related fluid systems. The staff concludes that the applicant correctly identified the ASME BPV Code of record to apply to the NuScale design.

5.2.1.2 Applicable Code Cases

5.2.1.2.1 Introduction

This SER section discusses the use of Code Cases associated with the ASME BPV Code and OM Code. In general, ASME develops a Code Case based on inquiries from the nuclear industry associated with possible clarification or modification of the codes or alternatives to the code. The ASME BPV Standards Committee eliminated Code Case expiration dates as of March 11, 2005. Therefore, all Code Cases will be automatically reaffirmed and remain available for use unless annulled by the ASME BPV Standards Committee. The NRC staff publishes ASME Code Cases acceptable to it in RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III”; RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1”; and RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” as incorporated by reference in 10 CFR 50.55a(a)(3).

5.2.1.2.2 Summary of Application

DCA Part 2, Tier 1: There are no DCA Part 2, Tier 1, entries for this area of review.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 5.2.1.2, "Compliance with Applicable Code Cases," discusses the application of ASME BPV Code Cases to the NuScale Power Plant design. DCA Part 2, Tier 2, Table 5.2-1, "American Society of Mechanical Engineers Code Cases," lists the ASME BPV Code, Section III, Code Cases used by the applicant.

DCA Part 2, Tier 2, Table 5.2-1, lists the following Code Cases:

- ASME Code Case N-759-2, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Class 1, 2, and 3, Section III, Division 1," issued January 2008
- ASME Code Case N-62-7, "Internal and External Valve Items, Section III, Division 1, Classes 1, 2, and 3," issued February 2003
- ASME Code Case N-60-5, "Material for Core Support Structures, Section III, Division 1," issued February 1994

The applicant stated that those ASME BPV Code, Section III, Code Cases used for design, fabrication, and construction are those listed in Table 1 and 2 of RG 1.84, subject to the applicable provisions of the RG, as well as 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(b). The applicant discussed the potential use of ASME BPV Code, Section XI, and ASME OM Code Cases for preservice inspection (PSI) and ISI examinations, procedures, and testing but ultimately deferred them to the plant owner. The applicant additionally indicated that other Code Cases listed in RGs 1.84, 1.147, and 1.192 may be used.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: There are no TSs for this area of review.

Technical Reports: There are no TRs for this area of review.

5.2.1.2.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 1, as it relates to the requirement that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed, noting that this requirement is applicable to both pressure-retaining and nonpressure-retaining SSCs that are part of the RCPB, as well as other systems important to safety, and that, where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability
- 10 CFR 52.47(b)(1), as it relates to a DCA containing 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 plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the AEA, and the NRC's regulations
- 10 CFR 50.55a, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain

components of boiling-water reactor and pressurized-water reactor (PWR) nuclear power plants by requiring compliance with appropriate editions of specified published industry codes and standards

The guidance in SRP Section 5.2.1.2, "Applicable Code Cases," lists the acceptance criteria adequate to meet the above requirements and includes review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.84, as it relates to ASME BPV Code, Section III, Code Cases

- RG 1.147, as it relates to ASME BPV Code, Section XI, Code Cases

- RG 1.192, as it relates to ASME OM Code Cases

5.2.1.2.4 Technical Evaluation

5.2.1.2.4.1 Compliance with Regulatory Guide 1.84

The staff has reviewed DCA Part 2 for compliance with the requirements presented above. Acceptable ASME Code Cases that may be used for the NuScale Power Plant are those either conditionally or unconditionally approved in applicable NRC RGs, as incorporated into 10 CFR 50.55a(a)(3), and that are in effect at the time of the DC. DCA Part 2, Tier 2, Section 5.2.1.2, states that the ASME BPV Code, Section III, Code Cases used for design, fabrication, and construction are determined by those listed in the applicable ASME BPV Code Edition specified in 10 CFR 50.55a(a)(1)(i) or Tables 1 and 2 of RG 1.84 pursuant to 10 CFR 50.55a(a)(3)(i) and subject to the applicable provisions of 10 CFR 50.55a(b). It further states that Code Cases that are used and listed in Table 2 of RG 1.84 also meet the conditions established in the RG. SER Section 5.2.1.2.2 lists the ASME BPV Code, Section III, Code Cases used in the NuScale design.

The staff reviewed DCA Part 2, Tier 2, Section 5.2.1.2, and found that Table 5.2-1 lists ASME Code Cases that have been conditionally and unconditionally approved in accordance with RG 1.84. The staff finds the reference to conditionally and unconditionally approved Code Cases to be consistent with 10 CFR 50.55a and RG 1.84 and, therefore, acceptable.

Additionally, the staff noted that DCA Part 2, Tier 2, Section 5.2.1.2, does not identify the use of any ASME BPV Code, Section III, Code Cases related to Class 2 or 3 components. The staff is aware of other certified designs that have used such Code Cases. The staff issued RAI 8917, Question 05.02.01.02-1, requesting that the applicant either verify that no ASME BPV Code, Section III, Code Cases related to Class 2 or 3 components are to be used as part of the NuScale design or revise DCA Part 2, Tier 2, Section 5.2.1, "Compliance with Codes and Code Cases," to identify such Code Cases if they will be used as part of the design. In its response dated October 12, 2017, to RAI 8917, Question 05.02.01.02-1 (ADAMS Accession No. ML17285B452), the applicant confirmed that DCA Part 2, Tier 2, Table 5.2-1, was complete and accurate and that no additional ASME BPV Code Cases are used for any Class 1, 2, or 3 systems or components. Based on this response, the NRC staff considers this issue closed and resolved.

This information above supports the staff's determination that Code Cases are implemented in accordance with 10 CFR 50.55a, which incorporates by reference RG 1.84.

5.2.1.2.4.2 Compliance with Regulatory Guides 1.147 and 1.192

The staff observes that the applicant did not mention specific ASME Code Cases in accordance with RG 1.147 or RG 1.192. Therefore, no finding is necessary with respect to these RGs, as the specification of these ASME Code Cases has been deferred to the COL applicant. The applicant stated that the plant owner will identify the applicable ASME BPV Code, Section XI, and ASME OM Code Cases.

5.2.1.2.5 Combined License Information Items

A COL applicant may identify, within its COL application, the planned use of additional Code Cases or provide proposed alternatives to meeting 10 CFR 50.55a, provided they do not alter the staff's safety findings on the NuScale certified design. DCA Part 2, Tier 2, Table 1.8-2, does not need to include additional COL information items for Code Cases.

5.2.1.2.6 Conclusion

The staff finds that the ASME Code Cases identified in DCA Part 2 are acceptable as specified in the applicable NRC RGs, with conformance to conditions in the applicable RGs. The staff concludes that the information in DCA Part 2 with respect to the use of ASME Code Cases is acceptable and sufficient to support compliance with the requirements of GDC 1 and 10 CFR 50.55a that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

5.2.2 Overpressure Protection

5.2.2.1 Introduction

The NuScale design provides overpressure protection features to protect the RCPB, the primary side of auxiliary systems (such as the chemical and volume control system (CVCS)), the secondary side of the SGs (including the decay heat removal system (DHRS) and feedwater lines), and the control rod drive system (CRDS) cooling piping. Overpressure protection components include the pilot-operated RSVs, the battery-supplied power-operated reactor vent valves (RVVs), the thermal relief valves, and the PZR.

Two RSVs are connected to the top of the RV upper head above the PZR region to provide overpressure protection during startup, shutdown, and power operation. Also, three emergency core cooling system (ECCS) RVVs are connected to the top of the RV upper head and function as part of the low-temperature overpressure protection (LTOP) system to provide overpressure protection to ensure the pressure boundary behaves in a nonbrittle manner. One CRDS thermal relief valve provides overpressure protection for the CRDS cooling piping after a containment isolation event during plant operation. Two SG thermal relief valves in the feedwater piping provide overpressure protection during water-solid conditions that may occur while the NPM is shut down. The PZR and CVCS relief valves provide CVCS overpressure protection when RCS pressure changes from operating transients do not result in a reactor trip or containment isolation. The safety evaluation of these components appears in their respective sections of this report.

5.2.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 2, "Unit Specific Systems, Structures, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria," describes the SSCs that are specific to the operation of an NPM, including the overpressure protection features of pressure-relieving components of systems associated with the NPM. DCA Part 2, Tier 1, Table 2.1-2, "NuScale Power Module Mechanical Equipment," provides an inventory of the mechanical components, including the PZR, RSVs, RVVs, SG RVs, and CRDS RV for overpressure protection. DCA Part 2, Tier 1, Table 2.5-2, "Module Protection System Automatic Engineered Safety Feature Functions," includes the LTOP actuation function, which is linked with the low-temperature interlock with a high-pressure condition. DCA Part 2, Tier 1, Table 2.8-1, "Module Specific Mechanical and Electrical/I&C Equipment," lists the components identified above for equipment qualification. The applicant gave the ITAAC for overpressure protection in DCA Part 2, Tier 1, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria"; Table 2.5-7, "Module Protection System and Safety Display and Indication System Inspections Tests, Analyses, and Acceptance Criteria"; and Table 2.8-2, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria."

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 5.2.2, "Overpressure Protection," describes the features to protect the RCPB from overpressurization, including the primary side of auxiliary systems connected to the RCS and the secondary side of the SGs. Also, this section describes the overpressure protection of the CRDS cooling piping during a containment isolation event. Because the overpressure protection function includes pressure relief components of other systems, the applicant gave additional information on these devices in DCA Part 2, Tier 2, Sections 3.9.3.2, "Design and Installation of Pressure Relief Devices"; 5.4.1, "Steam Generators"; 5.4.5, "Pressurizer"; and 6.3, "Emergency Core Cooling System."

ITAAC: The applied described overpressure protection in DCA Part 2, Tier 1, Chapter 2, and testing of the pressure-relieving devices includes ITAAC Tables 2.1-4 (NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria), 2.5-7 (Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria), and 2.8-2 (Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria).

Technical Reports: TR-1015-18177-P, "Pressure and Temperature Limits Methodology," Revision 0, issued December 2016

Technical Specifications:

- NuScale TS 2.1.2, "RCS Pressure SL," which is the RCS pressure safety limit (SL)
- NuScale TS 3.4.1, "RCS Pressure and Temperature Critical Heat Flux (CHF) Limits," which includes RCS CHF parameters for PZR pressure
- NuScale TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," which includes RCS pressure
- NuScale TS 3.4.4, "Reactor Safety Valves (RSVs)," which is the TS related to the RSVs
- NuScale TS 3.5.1, "Emergency Core Cooling System (ECCS)," which includes operation of the RVVs

- NuScale TS 5.6.3, “Core Operating Limits Report (COLR),” which discusses the reporting of the COLR and lists TS 3.4.1

- NuScale TS 5.6.4, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” which discusses the PTLR

Initial Test Program:

- NuScale DCA Part 2, Tier 2, Table 14.2-47, “Emergency Core Cooling System Test # 47”

- NuScale DCA Part 2, Tier 2, Table 14.2-63, “Module Protection System Test # 63,” which provides instrument information signals for LTOP actuation

5.2.2.3 Regulatory Basis

SRP Section 5.2.2, “Overpressure Protection,” SRP Section 5.4, Reactor Coolant System Component and Subsystem Design,” and Branch Technical Position (BTP) 5-2, “Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures,” contains the following relevant requirements and the acceptance criteria, as well as review interfaces with other SRP sections:

- GDC 15, “Reactor Coolant System Design,” as it relates to designing the RCS and associated auxiliary, control, and protection systems with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs)

- GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” as it relates to designing the RCPB with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fractures is minimized

- 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi), which reference the Three Mile Island (TMI) Action Plan Items II.D.1 and II.D.3 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980

- 10 CFR 52.47(a)(8), which requires DC reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f)

- 10 CFR 52.47(b)(1), which requires that a DCA 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 plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the AEA, and the NRC’s regulations

5.2.2.4 Technical Evaluation

The overpressure protection system is designed to prevent RCPB pressure from exceeding 110 percent of design pressure during full power, startup, and shutdown conditions, including

AOOs, and 120 percent for postulated accidents. In addition, the design provides sufficient capacity to ensure that the design limits are not exceeded during an anticipated transient without scram (ATWS).

Overpressure Protection during Power Operations

The RSVs' main design function is to provide overpressure protection for the RV as part of the RCPB. The RSVs are redundant, and safety related, designed to maintain pressure below 110 percent of the design pressure of 162.4 kilograms per square centimeter absolute (2,310 pounds per square inch absolute). Each RSV is designed to provide 100 percent of the required relief capacity. The RSV is a pilot-operated relief valve designed in accordance with ASME requirements. During power operations, RCS pressure is routed to the RSV chamber located above the pilot disc, where it expands the pilot valve bellows and seats the pilot disc. Relief pressure is determined by the spring preload of the pilot valve and the main valve closing startup tests spring pressure. However, the staff was unable to locate an ITAAC that tests the capability of the RSV to perform its functions. The proposed ITAAC in NuScale DCA Part 2, Tier 1, Table 2.1-4, are not sufficient to verify the capability of the NPM safety-related valves during preoperational testing.

The staff noted that the design commitment should specify that the NPM safety-related valves change position under design-basis temperature, differential pressure, and flow conditions. Also, the inspections, tests, and analyses should specify that a diagnostic stroke test will be performed of the NPM safety-related valves under preoperational temperature, differential pressure, and flow conditions. The acceptance criteria should specify that each NPM safety-related valve strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions with sufficient diagnostic data to correlate valve performance to its design-basis capability, as established by the type test performed in accordance with the applicable functional qualification ITAAC. To address this concern, the staff submitted RAI 9132 for the applicant to discuss the intent of its proposed ITAAC in comparison to the standardized ITAAC or revise the proposed ITAAC to verify the capability of the NPM safety-related valves during preoperational testing. SER Chapter 14 evaluates RAI 9132. The staff is tracking this as **Open Item 5.2.2-1**.

The RSVs are designed with sufficient capacity to limit the RCPB pressure to 110 percent of the design pressure under normal conditions and AOOs and 120 percent under postulated accident conditions. NuScale DCA Part 2, Tier 2, Section 5.2.2.6, "Applicable Codes and Classification," states that the RSVs are designed in accordance with ASME Code, Section III, Subarticle NB 3500, and function to satisfy the overpressure protection criteria described in ASME Code, Section III, Article NB 7000. This is discussed in Chapter 3 of the SER report.

However, as noted in NuScale DCA Part 2, Tier 2, Section 5.2.2.2.1, "Overpressure Protection during Power Operations," the turbine trip at full power without bypass capability is the most severe AOO and is the bounding event used in the determination of RSV capacity and the RV overpressure analyses. As part of the CIV-RSV Audit, the staff reviewed the RSV detailed design description, including specific design aspects, to address the capacity certification for the various fluid conditions over their full range of operating conditions, up to and including design-basis accident conditions.

The RSVs sizing calculation uses a methodology based on identifying the maximum volumetric surge rate into the pressurizer that occurs during the turbine trip transient. In their analysis, the applicant uses three general methods of adding conservatism to the RSV capacity: (1) increase the heat addition to the reactor coolant system, (2) decrease the heat lost from the reactor coolant system, and (3) increase the coefficient of thermal expansion of the coolant by initiating the transient with coolant at a higher temperature. These are discussed below.

The calculation used the maximum steady state reactor power of 163.2 MW including a 2% instrument uncertainty. This higher reactor power generates a higher volumetric surge rate into the pressurizer to produce the most severe transient in accordance with NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B.i.

In addition, the analysis uses a reactor trip that is artificially delayed until after the RSVs lift, which is after the second trip signal. Both the high steam pressure and high pressurizer pressure trip signals occur before the RSV setpoint is reached. Therefore, only the pressurizer surge rate data prior to the RSV lift is used in the sizing calculation for the RSVs, so the reactor trip is not credited in the analysis for rapid response. This satisfies the intent of NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B.iii.

In the analysis, the steam generator pressure and feedwater temperature are set to 42.2 kilograms per square centimeter absolute (600 pounds per square inch absolute) and 150.3 degrees Celsius (302.5 degrees Fahrenheit), respectively. Both of these parameters are set higher than the expected operating values; thus, the elevated temperature and pressure reduces the ability for the secondary side to remove heat, which increases the volumetric surge rate into the pressurizer.

Also, the hot leg temperature is set to 321.1 degrees Celsius (610 degrees Fahrenheit), which is the maximum value allowed by the analytical limits. This hotter RCS coolant results in an increased coefficient of thermal expansion, which increases the surge rate into the pressurizer.

Finally, the heat up and pressurization during the transient will tend to reduce core power due to reactivity feedback associated with an increase in the moderator temperature. However, the reactivity feedback is not credited in this analysis to ensure that core power is not reduced. This will result in a greater rate of coolant expansion.

Therefore, the staff finds the analysis and methodology used to determine the RSV flow capacity is acceptable because it complies with the NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B. In addition, the staff finds RSV design satisfies the single failure criteria as defined in NUREG-0800, Section 5.2.2, Acceptance Criteria 3.C because each RSV is designed to provide 100 percent of the required relief capacity.

Low-Temperature Overpressure Protection System

The primary purpose of LTOP is to prevent the RCPB pressure from exceeding the limiting pressure when operating below the LTOP enabling temperature. This will ensure that the RV is maintained below brittle fracture stress limits during operating, maintenance, testing, or postulated accident conditions at low temperatures. The LTOP limit is determined as a function of RCS cold temperature and is based on the worst case low-temperature overpressure transient. The applicant’s analysis has determined that the spurious actuation of the PZR heaters with a heat input of 880 kilowatts from the PZR heaters and 2 megawatts from core decay heat is the limiting event. This limiting event assumes the initial conditions that maximize the rate of PZR level increase as it approaches a water-solid condition, resulting in a maximum pressurization rate. The analysis results indicate the peak pressure remains below the brittle fracture stress limit.

The RVVs provide the overpressure protection during low-temperature conditions with the assumption that it results in excessive heat being added to the RCS. Based on the worst-case results, the RVVs are designed with sufficient capacity to prevent RCPB pressure from exceeding the limiting pressure while operating below the LTOP enabling temperature of 162.78 degrees Celsius (C) (325 degrees Fahrenheit (F)). The RVVs are capable of opening during startup and shutdown conditions while discharging directly from the RCS to containment to provide LTOP protection. Each RVV is configured with a trip valve and a reset valve, which are solenoid pilot valves constructed in accordance with the ASME Code. When operating at normal power and RCS pressure, the probability of an inadvertent opening of the RVVs is minimized by an inadvertent actuation block feature in the RVV actuators. The inadvertent actuation block feature does not prevent the LTOP actuation of the RVVs when the LTOP enabling setpoint is reached because the inadvertent actuation block arming setpoint is above the limiting RV pressure at LTOP conditions.

Three RVVs and associated valve controls ensure LTOP protection is maintained, assuming a single active component failure. The RVVs are designed with sufficient pressure relief capacity to accommodate the most limiting single active failure, assuming the most limiting allowable operating condition and system configuration. With regard to ITAAC, the staff noted that NuScale DCA Part 2, Tier 1, Table 2.1-4, proposes ITAAC 19 to verify the loss of motive power capability of RVV safety-related valves. However, the ITAAC should verify that each RVV will perform its function to fail to (or maintain) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions sufficient to correlate valve performance with its design-basis capability, as established by the type test performed in accordance with the applicable functional qualification ITAAC. The NRC has prepared standardized ITAAC to verify the loss of motive power capability for safety-related valves in new reactors for the conditions stated above. Therefore, in RAI 9132, Question 30711, the NRC staff requested that the applicant discuss the intent of its proposed ITAAC in comparison to the standardized ITAAC or revise the proposed ITAAC to verify the capability of the RVVs during preoperational testing. SER Chapter 14 evaluates the applicant's response in its letter dated November 21, 2017. The staff is tracking this as **Open Item 5.2.2-1**. SER Section 6.3 contains the primary review of the RVVs.

In addition, the staff issued RAI 9469, Question 06.03-6 as a follow-up to RAI 8820, Question 03.09.06-1 and initiated an audit to address the ECCS valve design characteristics including the RVVs function related to LTOP operation. Because these characteristics, including the valve sizing and flow capacity, are assumed in demonstrating the LTOP capability during the limiting event of the spurious actuation of the PZR heaters, this issue remains open pending resolution of the above RAIs and audit issues. The staff is tracking this as **Open Item 5.2.2-2**.

5.2.2.4 Combined License Information Items

SER Table 5.2.2-1 lists the COL information item number and description related to the RCPB and secondary system adequate overpressure protection features, from DCA Part 2, Tier 2, Table 1.8-2.

Table 5.2.2-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 5.2.2

Item No.	Description	DCA Part 2 Tier 2 Section
COL Item 5.2-2	A COL applicant that references the NuScale Power Plant design certification will provide a certified Overpressure Protection Report in compliance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Subarticles NB-7200 and NC-7200 to demonstrate the reactor coolant pressure boundary and secondary system are designed with adequate overpressure protection features.	5.2.2

5.2.2.5 Conclusion

The staff reviewed the overpressure protection for full-power and low-temperature conditions, as described in DCA Part 2, Tier 2, Section 5.2.2, and TR-1015-18177-P, to determine whether overpressure protection is sufficient to ensure that 110 percent of the design pressure is not exceeded during normal power operations and during LTOP conditions. However, the staff is unable to make a safety finding until the open items described above are resolved. The staff will update SER Section 5.2.2 to reflect the final disposition of DCA Part 2, Tier 2.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Introduction

This section addresses the materials that make up the RCPB. RCPB materials are fabricated and selected to maintain pressure boundary integrity for the design life of the plant. Ferritic low-alloy and carbon steel RCPB components have either austenitic stainless steel or nickel-based alloy cladding on surfaces exposed to the reactor coolant.

5.2.3.2 Summary of Application

By application dated December 31, 2016, as supplemented by letter dated August 3, 2017 (ADAMS Accession No. ML17215A977), the applicant has described the RCPB materials in accordance with 10 CFR 52.47, “Contents of Applications; Technical Information.”

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, “NuScale Power Module,” includes the information associated with this section. The Tier 1 information requires that the RCPB provide the second barrier against radioactive product leakage. DCA Part 2, Tier 1, Tables 2.1-1 through 2.1-3, “NuScale Power Module Piping Systems” and “NuScale Power Module Electrical Equipment,) respectively contain detailed Tier 1 mechanical, electrical, and instrumentation information associated with the RCPB. DCA Part 2, Tier 1, Table 2.1-4, specifies the ITAAC for the RCPB.

DCA Part 2, Tier 2: The applicant provided a Tier 2 description of the materials used in the RCPB in DCA Part 2, Tier 2, Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.” On August 3, 2017, the applicant provided supplemental information that clarified information in the original application and responded to issues identified by the staff. The docketed responses are treated as supplemental information to the design control document (DCD); any responses that include modifications to the FSAR are treated as confirmatory items and are indicated as such.

The paragraphs below summarize the information in the DCD, in part.

Material Specifications and ASME Code Compliance

The application gave materials specifications, including the grade or type and final metallurgical conditioning of materials used for the RCPB. Generally, the RCPB is composed of low-alloy steel, austenitic stainless steel, and nickel alloy components. The three tables below summarize the RCPB materials.

Reactor Coolant Pressure Boundary Materials

Base Metal Specification	Grade/Type
SB-163, SB-166, SB-167, SB-168, SB-564	Alloy 690 (UNS N06690)
SB-637	Alloy 718
SA-182	Grade F304/F304L
SA-193	B8, Class 1
SA-240, SA-312, SA-479	Type 304/304L
SA-312	Type 316L; Seamless
SA-479	Type 316/316L
SA-508	Grade 3, Class 1
SA-508	Grade 3, Class 2
SA-533	Type B, Class 2
SA-965	Grade F304LN

Weld Material Specification	Grade/Type
SFA-5.5, SFA-5.23, SFA-5.28, SFA-5.29	TBD (see below)
SFA-5.4, SFA-5.9, SFA-5.22	E/ER308, E/EQ/ER308L, E/ER309, E/EQ/ER309L, E/ER316, E/ER316L

Weld Material Specification	Grade/Type
SFA-5.11, SFA-5.14	E/EQ/ERNiCrFe-7, EQ/ERNiCrFe-7A

In Revision 2 of the application NuScale removed the RCS check valves, RCS Excess Flow Check Valves, RCS Injection and Discharge Isolation Valves from DCA, Part 2, Tier 2, Table 5.2-4. By deleting the reference in Table 5.2-4, Table 5.2-4 is incomplete as it does not list all of the RCPB components. The staff are following up with NuScale on this deletion and it is being tracked as Open Item 5.2.3-1.

The applicant did not specify the weld filler material for the joining low-alloy steel components. The applicant stated that the weld filler material will have the same chemical composition as the base metal and will exceed the mechanical properties of the base metal. The staff finds the applicant's requirements to be generally consistent with the requirements in the ASME Code and to be sufficiently narrow to prevent the use of material that the staff would find unacceptable or acceptable with conditions. The NRC still requires the applicant to meet the ASME Code in accordance with 10 CFR 50.55a. The staff finds that the selection of RCPB materials is adequate and consistent with 10 CFR 50.55a.

The applicant stated that the RCPB materials will meet the requirements for ASME Code, Section III, Division 1, Class 1, including the material requirements in Article NB-2000. The staff evaluated the applicant's use of the ASME Code as it relates to conformance to 10 CFR 50.55a in SER Section 5.2.1.

Compatibility of Materials with the Reactor Coolant

The applicant stated that the RCPB materials are selected with consideration of the reactor coolant chemistry to avoid degradation or failure in environmental conditions associated with normal operations, maintenance, testing, and postulated accidents. The RCS water chemistry is controlled in accordance with the Electric Power Research Institute (EPRI) TR 3002000505, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Revision 7, issued April 2014 (EPRI Guidelines). The EPRI Guidelines minimize negative impacts of chemistry on materials integrity, fuel rod corrosion, fuel design performance, and radiation fields, and is routinely analyzed for verification.

The applicant stated that the CVCS maintains the primary water chemistry in accordance with the EPRI Guidelines. The applicant stated that chemical additions to the primary water will include boric acid for reactivity control, lithium hydroxide (enriched with lithium-7) for pH control, hydrogen for oxygen scavenging during normal operations, hydrazine for oxygen scavenging at low temperatures during startup, and zinc to reduce radiation levels and reduce primary water stress-corrosion cracking (PWSCC) initiation rates.

Fabrication and Processing of Ferritic Materials

The applicant described controls for ferritic steel materials to justify the material's sufficiency for a 40-year period of operation. These controls include specifying a minimum fracture toughness of the material, describing controls for welding to prevent hydrogen embrittlement and cold cracking, and including acceptance criteria for corrosion-resistant cladding that is welded onto the ferritic steel to protect the base metal from the water in the primary coolant system or the ultimate heat sink.

Fracture toughness of ferritic materials is provided by complying with 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” and ASME Code, Section III, Subarticle NB-2300. Welding of ferritic materials will be qualified in accordance with ASME Code, Section IX, and will meet the requirements in ASME Code, Section III, Subarticle NB-4300. Preheat and interpass temperature controls for ferritic steels will meet the requirements of ASME Code, Section II, Division 1, Nonmandatory Appendix D. In addition, the applicant committed to placing controls on preheat for low-alloy steel forging, consistent with RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” Revision 1, issued March 2011. Welding of ferritic steel under limited accessibility conditions will be controlled in accordance with the guidance in RG 1.71, “Welder Qualification for Areas of Limited Accessibility.” Finally, postweld heat treatment of low-alloy RV materials will be at 607.22 degrees C (1,125 degrees F (+/- 25 degrees F)).

Ferritic low-alloy and carbon steels used in principal pressure-retaining applications have either austenitic stainless steel or nickel-based alloy corrosion-resistant cladding on all pressure-retaining surfaces that are exposed to the reactor coolant. The RV is clad with at least two layers of austenitic stainless steel on its interior and clad at least one layer on its exterior. After cladding, the RV receives a postweld heat treatment, as required by ASME Code, Section III, Subsubarticle NB-4622. Clad low-alloy steel forgings will have American Society for Testing and Materials (ASTM) grain size 5 or finer. The applicant committed to meeting RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” which requires controls to limit underclad cracking and weld qualification to demonstrate that underclad cracking is prevented.

Fabrication and Processing of Austenitic Stainless Steel Materials

The applicant described controls for austenitic stainless steel materials to justify the material’s sufficiency for a 40-year period of operation. Controls for austenitic stainless steel components primarily focus on preventing sensitization of unstabilized stainless steels (specifically, type 304 and 316) in the base material and heat-affected zone of weldments.

The applicant stated that unstabilized austenitic stainless steel base materials will be solution annealed and water quenched (or rapidly cooled), in accordance with the guidance in RG 1.44, “Control of the Processing and Use of Stainless Steel.” Austenitic stainless steels heated above 426.67 C (800 degrees F) for post-weld heat treatment or other purposes, other than locally by welding operations, for more than 60 minutes will be tested for nonsensitization in accordance with ASTM A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” Practice A or E. If the base metal is cooled after solution annealing by a means other than water quenching, then the applicant will verify that the material is free of sensitization by testing the material in accordance with ASTM A262, Practice A or E.

Most austenitic stainless steel base metal and weld metal have carbon content less than 0.03 weight percent. One component, the RPV instrument seal assemblies set screws, will have carbon content above the “L Grade” limits. However, this component will not be welded during fabrication and will not be post weld heat treated. As such, the base metals in the NuScale design are resistant to stress corrosion cracking.

Cold working of austenitic stainless steel will be avoided to the extent practicable. If cold working is performed, then the yield strength of the base material shall not exceed 620.53 megapascals (MPA) (90,000 pounds per square inch (psi)).

Austenitic stainless steel weld materials for the RCPB system will have delta ferrite number (FN) between 5 and 20 in accordance with the guidance found in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 4, issued October 2013. Welding of austenitic stainless steels will be qualified in accordance with ASME Code, Section IX, and will meet the requirements of ASME Code, Section III, Subsection NB-4300. Welding of austenitic stainless steel under limited accessibility conditions will be controlled in accordance with the guidance in RG 1.71.

<u>Fabrication and Processing of Nickel-Based Alloy Materials</u>

The applicant uses alloy 690 base materials for primary pressure-retaining applications and will clad some low-alloy ferritic steel components with weld filler 52/152. Nickel-based alloys will have a sulfur content limit of a maximum 0.02 weight percent. The alloy 690 base metal for SG tubes is used in the solution annealed and thermally treated condition to optimize resistance to intergranular corrosion.

<u>Cleaning of Components and Systems</u>

The applicant committed to the cleaning, handling, storing, and shipping requirements in ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," Subpart 2.1, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," and Subpart 2.2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants," for austenitic stainless steel components.

ITAAC: Items 1 and 2 in DCA Part 2, Tier 1, Table 2.1-4 state that inspections will verify that ASME Code, Section III, design reports and data reports for RCPB components in DCA Part 2, Tier 1, Tables 2.1-1 and 2.1-2 exist and conclude that the requirements of ASME Code, Section III, are met.

Technical Specifications: There are no TS for this area of review.

5.2.3.3 Regulatory Basis

SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," includes the following relevant requirements of the NRC regulations for this area of review, and the associated acceptance criteria, as well as review interfaces with other SRP sections:

- GDC 1 and 30, as they relate to quality standards for design, fabrication, erection, and testing
- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the compatibility of components with environmental conditions
- GDC 14, "Reactor Coolant Pressure Boundary," and GDC 31, as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB

- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criterion XIII, “Handling, Storage, and Shipping,” as it relates to onsite material cleaning control

- 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB

- 10 CFR 50.55a, as it relates to quality standards applicable to the RCPB

Acceptance criteria adequate to meet the above requirements include the following:

- RG 1.31, as it relates to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems

- RG 1.34, ”Control of Electroslag Weld Properties,” as it relates to acceptable solidification patterns and impact test limits and the criteria for verifying conformance during production welding

- RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” as it relates to acceptance criteria for compatibility of austenitic stainless steel with thermal insulation

- RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” as it relates to the quality of water used for final cleaning or flushing of finished surfaces during installation

- RG 1.43, as it relates to criteria to limit the occurrence of underclad cracking in low-alloy steel safety-related components clad with stainless steel

- RG 1.44, as it relates to the compatibility of RCPB materials with the reactor coolant and the avoidance of stress-corrosion cracking (SCC)

- RG 1.50, as it relates to criteria for preheating low-alloy steel to prevent cold cracking

- RG 1.71 as it relates to welder requalification

- RG 1.84, as it relates to alternate design methodologies, fabrication practices, and materials that have been accepted by the NRC and the ASME Code

5.2.3.4 Technical Evaluation

Material Specifications and ASME Code Compliance

The staff reviewed the material specifications and the grades and types of materials selected for the RCPB to ensure that the RCS is manufactured to the “highest quality standards practical.”

The applicant has selected RCPB base metal materials, which are included in ASME Code, Section II-D, Tables 2A or 2B, as is required by ASME Code, Section III, paragraph NB-2121. The applicant selected weld filler materials conforming to SFA specifications, which are specifications endorsed by the ASME Code and are compliant with ASME Code, Section III, paragraph NB-2400. The staff concludes that the applicant has selected materials that are allowed for ASME Code, Section III, applications.

The applicant stated that all materials will meet the requirements of ASME Code, Section III, Article NB-2000. In doing so, the applicant ensured that the ASME Code requirements for material certification, identification, heat treatment, testing, repair welding, examination, and quality systems will be met.

As stated in DCA Part 2, Tier 2, Section 5.2.1.1, all RCPB components will meet the regulatory requirements found in 10 CFR 50.55a, including the design, construction, stamping, and overprotection requirements in ASME Code, Section III. ASME Code compliance also requires meeting the fabrication (including welding) requirements of ASME Code, Section III, Article NB-4000, the nondestructive examination (NDE) requirements of NB-5000, the overpressure protection requirements of NB-6000, and the quality assurance requirements of Subsection NCA. Verification of ASME Code compliance is achieved by ITAAC Items 1 and 2 in DCA Part 2, Tier 1, Table 2.1-4.

Compatibility of Materials with the Reactor Coolant

The staff reviewed the information in the DCD to ensure that (1) the materials selected by the applicant are compatible with the reactor coolant and (2) the water chemistry of the primary coolant is adequately controlled.

Components in the RCPB are fabricated from low-alloy steels, austenitic stainless steels, and nickel-based alloys. The major components in the RCPB (RV vessel shells, flanges, and heads) are fabricated from low-alloy steel forgings. Low-alloy steel materials are susceptible to corrosion if the base metal is exposed to the borated primary coolant water or the borated ultimate heat sink water. The applicant has chosen to clad the low-alloy steel components with austenitic stainless steel or nickel weld filler 52/152 to protect the base metal from corrosion. The use of nickel weld filler 52/152 as cladding is limited to the area around the steam plenum cap. All other cladding in the RCPB is type E308 or E309 austenitic stainless steel. Currently operating nuclear power plants use alloys 52/152, E308, and E309, which join components in the NuScale design. Later SER sections further discuss welding of austenitic stainless steel and nickel-based alloys.

The staff reviewed the process controls for the welding of cladding to verify that the process will produce high-quality cladding. The applicant will require a minimum of two layers of cladding for the interior surface of the ferritic RCPB components and one layer for the exterior surface of ferritic RCPB components. The requirement for two layers of cladding on the interior surfaces as described in the FSAR is sufficient for the safety functions that the cladding performs. The multiple-layer process provides additional assurance that the cladding will resist damage. In addition, if the cladding is damaged, the primary water chemistry program would detect elevated iron levels in the water. In this manner, the NuScale design provides additional margin and monitoring for the cladding adequacy. The use of multiple-layer cladding on the interior surfaces is sufficient, considering that the elevated temperature boric-acid environment would result in significant corrosion of the ferritic material if cladding damage would occur. Additionally, a single layer of cladding may not provide significant assurance, considering the difficulty in inspecting the cladding on the upper head. The single layer of cladding on the exterior of the ferritic RCPB components is sufficient, considering that the annulus between the RV and the CNV is held in a vacuum during normal operation. In this manner, the RCPB exterior components are not exposed to a significant corrosive environment for a sufficient amount of time.

In the August 3, 2017, supplement to the DCD, the applicant stated that austenitic stainless steel cladding may use electroslag welding. The applicant did not commit to meeting RG 1.34. The staff agrees that RG 1.34 is not applicable because it is specific to electroslag joining processes, and the RG states that "the qualification of electroslag welding process for purposes of cladding is not addressed." Furthermore, RG 1.34 addresses crack susceptibility associated with deep pools of molten weld metal. The electroslag welding process for cladding is similar to a submerged arc weld process and would not produce a deep weld pool. The use of electroslag welding for cladding will be qualified in accordance with ASME Code, Section IX, QW-200.

The applicant also committed to meeting RG 1.43, which contains guidance to prevent underbead cracking. The applicant further stated that cladding will receive a postweld heat treatment meeting the requirements of ASME Code, Section III, Subsubarticle NB-4622, and that clad low-alloy steel forgings will have ASTM grain size 5 or finer. An ASTM grain size of 5 is a "fine-grain" microstructure, as described in SA-508, Article 4, which is consistent with guidance found in RG 1.43.

The applicant committed to meeting the latest version of the EPRI Guidelines. The applicant uses lithium hydroxide to control pH and hydrogen or hydrazine to control dissolved oxygen, consistent with the EPRI Guidelines. The applicant described the primary water chemistry limitations in DCA Part 2, Tier 2, Table 5.2-5, "Reactor Coolant Water Chemistry Controls," which are consistent with the EPRI Guidelines.

Fabrication and Processing of Ferritic Materials

The staff reviewed the DCD information to ensure that (1) brittle failure of ferritic materials is prevented, (2) the use of RGs is consistent with the intent described in the SRP, and (3) postweld heat treatments for ferritic materials are defined and meet the ASME Code.

The applicant committed to 10 CFR Part 50, Appendix G, and ASME Code, Section III, Subarticle NB-2300, which contain requirements for fracture toughness. The requirements in Appendix G and NB-2300 will ensure that RCPB materials have sufficient ductility to prevent rapidly propagating fracture or gross rupture.

The applicant committed to meeting RG 1.50 and RG 1.71. These RGs ensure that sufficient preheat will be used for low-alloy steels and that welding qualification will consider limited accessibility conditions. The applicant did not commit to meeting RG 1.34 for the RCPB. The staff finds this acceptable because the applicant prohibited using electroslag welding to join components.

The applicant stated that preheat and interpass temperature controls for ferritic steels will meet the requirements of ASME Code, Section II, Division 1, Nonmandatory Appendix D. The postweld heat treatment of low-alloy RV materials will be 607.22 degrees C (1,125 degrees F (+/- 25 degrees F)). The staff verified that the postweld heat treatment temperature meets the requirement in ASME Code, Section III, Table NB-4622.1-1, "Mandatory Requirements for Postweld Heat Treatment of Welds."

The applicant stated that the welding of corrosion-resistant cladding to the ferritic components will be in accordance with ASME Code, Section III, Subarticle NB-4300. The qualification of welding for cladding will use procedures meeting the requirements of ASME Code, Section IX. The staff finds the controls for the welding of cladding to be acceptable and consistent with the ASME Code and 10 CFR 50.55a.

The staff finds that the applicant provided sufficient controls on the fabrication and processing of ferritic materials, and the information in the DCD meets the acceptance criteria in the SRP.

<u>Fabrication and Processing of Austenitic Stainless Steel Materials</u>

The staff reviewed the information in the DCD to ensure that (1) corrosion is adequately considered in the design of the NuScale plant, (2) the applicant provided welding process controls to prevent hot cracking and SCC, (3) thermal embrittlement of cast austenitic steel components is prevented, (4) the applicant's use of RGs is consistent with the acceptance criteria in the SRP, and (5) the applicant incorporated appropriate NDE provisions on tubular products.

The applicant did not elect to increase the thickness of piping to compensate for erosion, corrosion, abrasion, or other environmental effects. The staff agrees that a corrosion allowance is not necessary for the NuScale design because (1) austenitic stainless steels (including cladding) are resistant to corrosion and erosion, (2) the potential for erosion in the NuScale design is relatively less than in operating plants because of natural circulation flow, and (3) the applicant committed to the EPRI Guidelines, which ensure that the potential for corrosion is minimized.

Controls for the welding of austenitic stainless steels are necessary to prevent hot cracking and SCC. Hot cracking is prevented by using weld filler material with sufficient delta ferrite. The applicant stated that welding material will have a delta ferrite content between 5 FN and 20 FN, which is consistent with the staff guidance in RG 1.31. Conformance to this RG is sufficient to prevent hot cracking and meets the SRP acceptance criteria.

The applicant uses unstabilized austenitic stainless steels, which can be susceptible to SCC. The applicant's three-step approach to SCC prevention, which is consistent with staff expectations in RG 1.44, is to (1) control material chemistry to prevent sensitization, (2) place limitations on cold work, and (3) control the reactor coolant chemistry. The applicant committed to meeting RG 1.44 and provided details on how regulatory guidance is met.

All austenitic stainless steel materials in the NuScale design are procured in the solution annealed and rapidly quenched state, as required by ASME Code, Section II and Section III. The applicant stated that unstabilized austenitic stainless steel materials exposed to sensitization temperatures subsequent to the solution heat treatment will have a carbon content no greater than 0.03 weight percent. The carbon content provision also applies to weld filler material, which ensures that all pressure boundary austenitic stainless steel base metals and weldments will be consistent with the staff guidance in RG 1.44.

The applicant committed to avoiding cold work to the extent practicable. If cold working must be done, it is limited to a yield strength less than 620.53 megapascals (MPA) (90,000 pounds per square inch (psi)) as determined by the 0.2 percent offset method. The controls on cold working are identical to SRP Section 5.2.3, Acceptance Criterion (4)(B). RG 1.44 cites the SRP guidance on cold work as adequate to limit stress concentration in the material.

As previously mentioned, the applicant committed to meeting the latest version of the EPRI Guidelines. Operating experience has shown that the guidelines are sufficient in monitoring and correcting water chemistry before material degradation can occur.

Operational experience with cast austenitic stainless steel components has shown that the base material can lose ductility when exposed to high temperatures or significant radiation fields over extended periods of time. The applicant stated that all cast austenitic stainless steel components used in the RCPB are at locations where neutron fluence is low enough to prevent irradiation embrittlement. The applicant also stated that the delta ferrite levels described in DCA Part 2, Tier 2, Table 6.1-3, “Pressure Retaining Materials for RCPB and Engineered Safety Feature (ESF) Valves,” are consistent with staff guidance described in the letter on “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,” from the NRC’s Christopher Grimes to the Nuclear Energy Institute (NEI), dated May 19, 2000 (ADAMS Accession No. ML15223A635). As such, the information in DCA Part 2, Tier 2, conforms to staff guidance on the use of cast austenitic stainless steel materials.

The applicant did not commit to meeting RG 1.34 or RG 1.36, which are RGs specified in the SRP for Section 5.2.3, for the RCPB. The staff agrees that these RGs are not applicable because the applicant prohibits using electroslag welding for joining components, and the applicant does not use insulation for the RCPB components. The applicant committed to meeting RG 1.71, which ensures that welding qualification will consider limited accessibility conditions.

The applicant committed to meeting ASME Code Subarticles NB-2550 and NB-2570 for austenitic stainless steel tubular components, which is consistent with the SRP acceptance criteria.

Fabrication and Processing of Nickel-Based Alloy Materials

The staff reviewed the information in the DCD to ensure that operating experience associated with PWSCC of nickel-based alloys is appropriately considered in the design of the NuScale plant.

The applicant committed to welding nickel-based alloys in accordance with ASME Code, Section III, Article NB-4300, and qualify welding processes in accordance with ASME Code, Section IX. Additional requirements for alloy 690 RCPB components will meet all requirements in ASME Code, Section III. The applicant committed to meeting the staff guidance in RG 1.71 when accessibility of weldments is restricted.

In DCA Part 2, Tier 2, Section 5.2.3.1, and as shown in DCA Part 2, Tier 2, Table 5.2-4, “Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances,” the applicant’s design uses alloy 690 base metal and alloy 52/152 weld filler materials for the RCPB. In laboratory testing, this alloy has higher resistance to corrosion, more resistance to fracture, and better mechanical properties at elevated temperatures when compared to alloy 600/82/182.

To ensure optimal performance of alloy 690/52/152 materials, the applicant committed to several industry best practices. These include (1) controlling the water chemistry, as described in the EPRI Guidelines, (2) optimizing the microstructure of the material by requiring components to be thermally treated, and (3) limiting sulfur content to 0.02 weight percent for components in contact with the reactor coolant.

The staff considers the use of alloy 690/52/152 with the described industry best practices to be consistent with the SRP acceptance criteria and therefore, acceptable.

Cleaning of Components and Systems

Surface contamination of RCPB components can introduce chemicals that are known to cause corrosion.

SRP Section 5.2.3 recommends cleaning procedures and processes based on RG 1.37. However, the staff has withdrawn RG 1.37 (ADAMS Accession No. ML13345A259) and now includes the guidance in RG 1.28, “Quality Assurance Program Criteria (Design and Construction).” RG 1.28 endorses the ASME nuclear quality assurance program standard NQA-1, including NQA-1, Subpart 2.1.

For all RCPB components, the applicant committed to meeting the requirements in NQA-1, Subpart 2.1. The scope of NQA-1, Subpart 2.1, includes cleaning water chemistry limitations, prohibitions on the use of ferritic grinding wheels on corrosion-resistant materials, and requiring the evaluation of process fluids for harmful chemicals. The applicant committed to meeting Cleanness Class B for all interior surfaces of the RCPB and Cleanness Class C for all exterior surfaces. The staff finds the cleanness classes chosen by the applicant to be consistent with the guidance in NQA-1. Additionally, the applicant committed to meeting the handling, storage, and shipping requirements in ASME NQA-1, Subpart 2.2.

ITAAC: Items 1 and 2 in DCA Part 2, Tier 1, Table 2.1-4 state that inspections will be performed to verify that ASME Code, Section III, design reports and data reports for RCPB components in DCA Part 2, Tier 1, Tables 2.1-1 and 2.1-2 exist and conclude that the requirements of ASME Code, Section III, are met. Item 6 in DCA Part 2, Tier 1, Table 2.1-4 states that a COL applicant will verify that the certified material test report of the RV beltline material concludes that the material has a charpy upper-shelf energy (USE) greater than 101.6 joules (J) (75 foot-pounds (ft-lb)).

5.2.3.5 Combined License Information Items

SER Table 5.2.3-1 lists item numbers and descriptions from DCA Part 2, Tier 2, Table 1.8-2.

Table 5.2.3-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 5.2.3

Item No.	Description	DCA Part 2 Tier 2 Section
5.2-4	A COL applicant that references the NuScale Power Plant design certification will develop and implement a Strategic Water Chemistry Plan. The Strategic Water Chemistry Plan will provide the optimization strategy for maintaining primary coolant chemistry and provide the basis for requirements for sampling and analysis frequencies, and corrective actions for control of primary water chemistry consistent with the latest version of the EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines.	5.2.3.2
5.2-5	A COL applicant that references the NuScale Power Plant design certification will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for	5.2.3.2

Item No.	Description	DCA Part 2 Tier 2 Section
	subsequent service, monitoring of the containment atmosphere for RCS leakage, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.	

The staff determines the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder.

5.2.3.6 Conclusion

The staff reviewed the reactor coolant pressure boundary materials, as described in DCA Part 2, Tier 2, Section 5.2.3. However, the staff is unable to make a determination on compliance with NRC regulations until the open item described above is resolved. The staff will update SER Section 5.2.3 to reflect the final disposition of DCA Part 2, Tier 2.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

5.2.4.1 Introduction

Nuclear power plants periodically use ISI and IST to assess the structural and leak-tight integrity of the RCPB throughout the operating lifetime of the facility. As required by 10 CFR 50.55a(g)(3), reactor designs certified on or after July 1, 1974, are required to provide access to enable the performance of ISI of ASME Code Class 1 RCPB components. Typically, a design should be developed that implements an ISI program consistent with the provisions of ASME Code, Section XI, Division 1, “Rules for Inservice Inspection of Nuclear Power Plant Components,” as supplemented by augmented ISI requirements in 10 CFR 50.55a. However, based on the specific attributes of a reactor design, additional augmented ISI may need to be proposed, and designed for, to support the design’s compliance with GDC 14, which requires the RCPB to be, “designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.”

5.2.4.2 Summary of Application

The application dated December 31, 2016, as supplemented by letters dated November 20, 2017, and June 12, 2018, contains the following sections:

DCA Part 2, Tier 1: There are no DCA Part 2, Tier 1, entries for this area of review. The system-based descriptions of DCA Part 2, Tier 1, Chapter 2, address design-related ASME Code requirements for system components.

DCA Part 2, Tier 2: The applicant has provided a DCA Part 2, Tier 2, description of its ISI program for Class 1 RCPB components in DCA Part 2, Tier 2, Section 5.2.4, as summarized below, in part.

In accordance with GDC 32, “Inspection of Reactor Coolant Pressure Boundary,” the ASME Code Class 1 components are designed and provided with access to permit periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity.

DCA Part 2, Tier 2, Section 5.2.4, states that the PSI and ISI are to be conducted in accordance with ASME Code, Section XI. Aside from describing the use of ASME Code, Section XI, DCA Part 2, Tier 2, Section 5.2.4, addresses the ISI requirements for the components and configurations unique to the NuScale design. An example of a unique system is the Class 1 piping, such as the RCS injection and discharge piping that exits containment. While these pipes are relatively small, at a nominal pipe size of 2 inches, they represent a direct path from the core to the environment.

5.2.4.3 Regulatory Basis

SRP Section 5.2.4 includes the relevant requirements of NRC regulations for this area of review and the associated acceptance criteria, summarized as follows, as well as review interfaces with other SRP sections:

- GDC 32, as it relates to periodic inspection and testing of the RCPB
- 10 CFR 50.55a, as it relates to the requirements for inspecting and testing ASME Code Class 1 components of the RCPB, as specified in ASME Code, Section XI
- ASME Code Case N-729-4, as modified by 10 CFR 50.55a(g)(6)(ii)(D) for RV head inspection

5.2.4.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Section 5.2.4, in accordance with SRP Section 5.2.4. DCA Part 2, Tier 2, Section 5.2.4, details the proposed requirements for the ISI and IST of the Class 1 components, including the pressure vessel and the Class 1 piping but excluding the SG tubing. The PSI and ISI are to be conducted in accordance with ASME Code, Section XI. The proposed initial ISI program will incorporate the latest edition and addenda of ASME Code, Section XI, approved in 10 CFR 50.55a(b).

The proposed ISI of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in RG 1.147 that are incorporated by reference in 10 CFR 50.55a(b), subject to the conditions listed in 10 CFR 50.55a(b)).

5.2.4.4.1 System Boundary Subject to Inspection

The definition of the system boundary subject to inspection is acceptable if it is in agreement with the definition of RCPB in 10 CFR 50.2. In accordance with 10 CFR 50.2, the RCPB for a PWR includes all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, which are as follows:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

(i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,

(ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment, and

(iii) The reactor coolant system safety and relief valves.

The examination requirements of ASME Code, Section XI, Subsection IWB, apply to all Class 1 pressure-retaining components and their welded attachments.

DCA Part 2, Tier 2, Section 5.2.4.1, "Inservice Inspection and Testing Program" covers the process for inspecting all components defined as Class 1 by the ASME Code. (DCA Tier 2, Part 2, Section 6.6, describes the inspection requirements for Class 2 and 3 components.) The DCA sufficiently describes the Class 1 components and follows the guidance in RG 1.26.

5.2.4.4.2 Accessibility

GDC 32 requires a design for accessibility. The design and arrangement of system components are acceptable if they provide adequate clearance in accordance with ASME Code, Section XI, Subarticle IWA-1500. Regulations in 10 CFR 50.55a(g)(3)(i) require Class 1 components, including supports, to be designed and be provided with access to enable the ISI of these components, in addition to meeting the preservice examination requirements set forth in the editions and addenda of Sections III or XI of the ASME Code of record.

The DCD sufficiently explains that the RCPB components will be designed to allow for the inspection and testing of important areas and features pursuant GDC 32. The DCD states that the NuScale design allows inspection, testing, and maintenance of the RSV, PZR heaters, SG primary and secondary sides, instruments, electrical connections, and other components located inside the RCPB of the NPMs.

5.2.4.4.3 Examination Categories and Methods

The examination categories and methods specified in DCA Part 2 are acceptable if they meet the requirements in ASME Code, Section XI, Article IWB-2000. Every area subject to examination falling within one or more of the examination categories in Article IWB-2000 must be examined, at least to the extent specified. The requirements of Article IWB-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

DCA Part 2, Tier 2, Table 5.2-6, "Reactor Pressure Vessel Inspection Elements," and Table 5.2-8, "American Society of Mechanical Engineers Class 1 Piping Inspection Elements" sufficiently describe the Class 1 components, including the vessel, nozzle, and piping welds; their examination category; and the examination methods.

5.2.4.4.4 Inspection Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of ASME Code, Section XI, Article IWA-2000.

As stated in DCA Part 2, Tier 2, Section 5.2.4, the inspections will be conducted in accordance with ASME Code, Section XI, which requires the use of 10-year inspection intervals. DCA Part 2, Tier 2, Section 5.2.4, also refers to the inspection interval as 10 years. This is acceptable because it meets the applicable requirements of ASME Code, Section XI.

5.2.4.4.5 Evaluation of Examination Results

The standards for the evaluation of examination results for Class 1 components are acceptable if they are in accordance with the requirements of ASME Code, Section XI, Article IWB-3000. DCA Part 2, Tier 2, Section 5.2.4, states that the examinations will be conducted in accordance with ASME Code, Section XI.

The proposed program on repair or replacement of components containing defects is acceptable if the program is in accordance with the requirements of ASME Code, Section XI, Article IWA-4000. ASME Code, Section XI, Article IWB-3000, describes the criteria that establish the need for repair or replacement. This is acceptable because it meets the applicable requirements of ASME Code, Section XI.

5.2.4.4.6 System Pressure Tests

The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program is in accordance with the requirements of ASME Code, Section XI, Article IWB-5000, and the TS requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. The pressure tests verify pressure boundary integrity in conjunction with ISI.

DCA Part 2, Tier 2, Section 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing,” adopts the requirements for system pressure tests as defined in ASME Code, Section XI, Articles IWA-5000 and IWB-5000. These tests verify the pressure boundary integrity in conjunction with the ISI program. This is acceptable, as the applicant’s methodology for performing pressure testing of the Class 1 boundary and components meets the requirements of the ASME Code; the methodology for performing system pressure testing is, therefore, acceptable.

5.2.4.4.7 Code Exemptions

Exemptions from ASME Code examinations should be permitted if the criteria in Subsubarticle IWB-1220 are met.

This section contains no requests for ASME Code exemptions.

5.2.4.4.8 Code Cases

ASME Code Cases referenced by the COL application will be reviewed for acceptability and compliance with RG 1.147. The NRC will review and accept Code Cases not specifically referenced in RG 1.147 on a case-by-case basis.

This section does not describe any specific Code Cases, although the DCD does refer to the optional ASME Code Cases listed in RG 1.147.

5.2.4.4.9 Augmented Inservice Inspection To Protect against Postulated Piping Failures

The DCD does not describe whip restraints for any Class 1 piping. This is acceptable, as the design does not include any significant length of Class 1 piping that would require a whip restraint.

5.2.4.4.10 Other Inspection Programs

For PWRs, the DCD needs to describe a program to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” dated March 17, 1988.

The Boric Acid Corrosion Detection and Control Program is considered part of DCA Part 2, Tier 2, Section 5.2.3.2.1, “Reactor Coolant Chemistry,” under COL Item 5.2-5, which states the following:

> A COL applicant that references the NuScale Power Plant design certification will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for subsequent service, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.

5.2.4.4.11 Inspections, Tests, Analyses, and Acceptance Criteria

The ITAAC associated with DCA Part 2 Tier 2, Section 5.2.4, are given in several sections of DCA Part 2, Tier 1. These ITAAC indicate that inspections will be performed on as-built components and piping and that reports exist that conclude the following:

- As-built ASME Code components, piping, and supports are designed and constructed in accordance with ASME Code, Section III, requirements.
- The ASME Code, Section III, requirements are met for NDE of the pressure boundary welds in as-built ASME Code components and piping.
- The results of hydrostatic testing of the as-built ASME Code components and piping conform with ASME Code, Section III, requirements.

The ITAAC described for this section are evaluated in DCA Part 2, Tier 2, Section 14.3.3.

5.2.4.4.12 Combined License Action Items

DCA Part 2, Tier 2, Section 5.2.4, contains one COL item, COL 5.2-6.

Item No.	Description	DCA Part 2 Tier 2 Section
5.2-6	A COL applicant that references the NuScale Power Plant design certification will develop site specific preservice examination, inservice inspection, and inservice testing program plans in	5.2.4

	accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and will establish implementation milestones. If applicable, a COL applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the augmented inservice inspection program. The COL applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a.	

The staff determines the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder.

5.2.4.4.13 Operational Program Description and Implementation

The NRC reviews the description of the operational program and proposed implementation milestone(s) for the PSI, ISI, and IST programs in accordance with 10 CFR 50.55a(g) and 10 CFR Part 50, Appendix A.

DCA Part 2, Tier 2, Section 5.2.4, describes the PSI, ISI, and IST programs. These programs are acceptable because they meet or exceed the requirements of 10 CFR 50.55a(g) and 10 CFR Part 50, Appendix A.

5.2.4.5 Conclusion

The design of the RCPB incorporates provisions for access to enable the performance of ISI examinations in accordance with 10 CFR 50.55a(g)(3) and ASME Code, Section XI. The final ISI program is required to meet the latest ASME Code, Section XI, Edition and Addenda incorporated by reference 18 months before the date scheduled for initial loading of fuel. Suitable equipment will be developed and installed to facilitate the remote inspection of these areas of the RCPB that are not readily accessible to inspection personnel. The final ISI program will consist of a PSI and ISI plan. The periodic inspections and pressure testing of pressure-retaining components of the RCPB are performed in accordance with the requirements in applicable subsections of Section XI of the ASME Code and provide reasonable assurance that evidence of structural degradation or loss of leak-tight integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the PSI and ISI program required by the ASME Code constitutes an acceptable basis for satisfying, in part, the requirements of GDC 32.

The staff concludes that the description of the PSI and ISI program is acceptable and meets the inspection and testing requirements of GDC 32 and 10 CFR 50.55a. This conclusion is based on the applicant meeting the requirements of ASME Code, Section XI, Division 1, as reviewed by the staff and determined to be appropriate for this application.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

5.2.5.1 Introduction

DCA Part 2, Tier 2, Section 5.2.5, “Reactor Coolant Pressure Boundary (RCPB) Leakage Detection,” discusses the RCS leakage detection systems, which are designed to detect and, to the extent practicable, identify the source of reactor coolant leakage. The RCS leakage detection systems conform to the guidance of RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” on detecting, monitoring, quantifying, and identifying reactor coolant leakage.

5.2.5.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.3, “Containment Evacuation System,” describes the containment evacuation system (CES) and design commitments relating to RCS leakage detection.

DCA Part 2, Tier 2: The applicant discussed RCPB leakage detection in DCA Part 2, Tier 2, Section 5.2.5, and Section 9.3.6, “Containment Evacuation System and Containment Flooding and Drain System.” DCA Part 2, Tier 2, Section 5.2.5, provides information on the leakage monitoring by using CES condensate and containment pressure. DCA Part 2, Tier 2, Section 9.3.6, describes applicable GDC for RCPB leakage detection and important components for the systems description. DCA Part 2, Tier 2, Section 5.2.5.2, “Leak-Before-Break,” and Section 3.6.3.5, “Leak Detection,” provide information about leak detection in the leak-before-break (LBB) evaluations. DCA Part 2, Tier 2, Figure 9.3.6-1, “Containment Evacuation System Diagram” shows a simplified CES diagram.

ITAAC: DCA Part 2, Tier 1, Table 2.3-1, “Containment Evacuation System Inspection, Test, Analyses, and Acceptance Criteria,” provides the ITAAC for RCS leakage detection.

Initial Test Program: DCA Part 2, Tier 2, Table 14.2-41, “Containment Evacuation System,” Test #41-3, addresses the initial test program for RCS leakage detection monitoring capability.

Technical Specifications: NuScale DCA Part 4, “Technical Specifications,” provides plant TS in Limiting Condition for Operation (LCO) 3.4.5, “RCS Operational Leakage,” to verify RCS operational leakage within specified limits and in LCO 3.4.7, “RCS Leakage Detection Instrumentation,” to address the RCS leakage detection instrumentation requirement.

Technical Reports: There are no TRs associated with the RCS leakage detection systems.

5.2.5.3 Regulatory Basis

Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection,” of “Design-Specific Review Standard for NuScale SMR [small modular reactor] Design,” Revision 0, issued June 2016 (ADAMS Accession No. ML15355A505) (DSRS), gives the relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, as follows:

- GDC 2, “Design Bases for Protection against Natural Phenomena,” as it relates to SSCs important to safety being designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, seiches, and tsunami without loss of capability to perform their safety functions

- GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated pipe rupture

- GDC 30, as it relates to the components that are part of the RCPB being designed, fabricated, erected, and tested to the highest quality standards practical, noting that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage

- 10 CFR 50.36(c)(2)(ii), which requires that a TS LCO of a nuclear reactor be established for each item meeting one or more of the specified Criteria 1, 2, 3, or 4.

- 10 CFR 52.47(b)(1), which requires that a DCA 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 plant that incorporates the DC has been constructed and will be operated in accordance with the DC, the provisions of the AEA, and NRC regulations.

5.2.5.4 Technical Evaluation

DCA Part 2, Tier 2, Section 5.2.5.1, “Leakage Detection and Monitoring,” states that there are two primary methods of leakage monitoring to detect leakage into containment—a change in containment pressure and condensate collected from the CES. The NuScale design uses a different approach for the RCPB leakage monitoring than other current PWR designs.

The staff reviewed the RCPB leakage detection systems described in DCA Part 2 in accordance with the NuScale design DSRS Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection.” As indicated in NuScale DSRS Section 5.2.5, the alternative leakage detection systems should be reviewed in sufficient detail according to the guidance in RG 1.45. GDC 30 requires that the components, which are part of the RCPB, 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. Although the proposed means for NuScale may not be the same as for the current PWR designs, the capability of the proposed means must be equivalent to the current PWR designs. NuScale DSRS Section 5.2.5 states that, for GDC 30, the review of RCPB leakage detection is based on meeting the guidance in RG 1.45, which is the same guidance being used for the current PWR designs.

5.2.5.4.1 Leakage Detection Capability, Sensitivity, and Response Time

RG 1.45 states that the capability of the leakage monitoring system includes overall response time, detector sensitivity, and accuracy. The instrument should be able to detect leakage of 1 gallon per minute (gpm) within an hour.

DCA Part 2, Tier 2, Section 9.3.6.3, “Safety Evaluation” states the following:

> Regulatory Positions C.2.1 and C.2.2, in RG 1.45 are satisfied in that:
>
> - Leakage to the primary reactor containment from unidentified sources can be detected, monitored, and quantified for flow rates greater than or equal to 0.05 gpm using containment vessel (CNV) pressure or containment evacuation system (CES) sample tank level timing.

- Leakage detection response time (not including transport delay time) is less than one hour for a leakage rate greater than 1 gpm using CNV pressure or CES sample tank level timing.

The CNV pressure and CES tank level indirectly measure the reactor coolant leakage and require correlations to relate to this leakage.

In the response, dated July 24, 2017 (ADAMS Accession No. ML17205A650), to RAI 8841, Question 05.02.05-02, the applicant provided additional information on how the instrument output of CNV pressure and CES tank level timing correlate to the reactor coolant leakage rate and to demonstrate that RG 1.45, Regulatory Positions C.1.2 and C.2.2, relating to sensitivity and response time of the RCS leakage were satisfied. In the response, the applicant provided explanations and equations to correlate CNV pressure and CES level to RCS leakage:

> The leakage into the containment vessel (CNV) can be calculated using pressure instruments by solving the ordinary differential equation for flow in a vacuum system for the leak rate. … Given a pressure reading, and assuming the leaking fluid is at RCS conditions, a volumetric flow rate for the leak can be calculated. The module control system will perform this calculation automatically.
>
> As leakage enters the CNV from the Reactor Coolant System (RCS), the fluid pressure goes below the vapor pressure and the fluid vaporizes. … As the vapor passes through the vacuum pump, it is condensed back to liquid in the condenser. The CES sample vessel collects the liquid. … Given that the dimensions of the CES sample vessel are known, the volumetric flow rate can be calculated.

In addition, the applicant provided a calculation to demonstrate the sensitivity of 0.05 gpm and the response time well within the criterion of 1 gpm leakage within 1 hour to be consistent with the guidance in RG 1.45, Regulatory Positions C.2.1 and C.2.2.

Based on DCA Part 2, Tier 2, Section 9.3.6.3, and the response to RAI 8841, Question 05.02.05-02, the staff found that the applicant has adequately demonstrated that the CNV pressure and CES tank level for reactor coolant leakage detection satisfy the quantitative criteria specified in RG 1.45, Regulatory Positions C.2.1 and C.2.2. Therefore, RAI 8841, Question 05.02.05-02, is resolved and closed.

5.2.5.4.2 Leakage Detection Systems

RG 1.45, Regulatory Position C.2.3, provides the following guidance on additional leakage detection systems:

> In addition to the monitoring systems detailed in the technical specifications, the plant should use other systems to detect and monitor for leakage, even if it does not have the capabilities specified in Regulatory Position C.2.2. These supplemental instruments/ methods may include, but are not limited to, the following:
>
> a) monitoring airborne gaseous radioactivity,
>
> b) monitoring the humidity of the containment,

c) monitoring the temperature of the containment,

d) monitoring the pressure of the containment,

e) monitoring acoustic emission, and

f) conducting video surveillance.

In addition, RG 1.45, Section B, on the subject of methods for monitoring leakage and identifying its source, states the following:

> Effective methods for monitoring (including detecting) any leakage and locating its source are important….
>
> … Because of the need to identify the source of leakage to assess its safety significance, plants should install monitoring systems to assist in locating the source of leakage during reactor operation. Plants can accomplish this, in part, by installing a number of instruments throughout containment and monitoring the response of each of these instruments to leakage. An instrument that is closer to a leak is likely to respond sooner than an instrument that is further away, assuming that the two instruments have similar capabilities (e.g., sensitivity).

DCA Part 2, Tier 2, Section 5.2.5, discusses CES gaseous discharge radioactivity monitoring, which can be used to identify the source of leakage with respect to leakage from the primary or secondary side. The staff found that DCA Part 2, Tier 2, Section 5.2.5, does not contain sufficient information to address the "supplemental instruments/methods" described in RG 1.45, Regulatory Position C.2.3. The staff notes that the supplemental methods, which do not have to satisfy the quantitative criteria in RG 1.45, Regulatory Positions C.2.1 and C.2.2, provide an indication for locating the leakage and identifying its source (e.g., from coolant of the primary side or secondary side). Therefore, the staff requested the applicant in RAI 8915 to address the supplemental methods for leakage detection that will provide diverse qualitative leakage information.

In the response, dated August 14, 2017, to RAI 8915, Question 05.02.05-05 (ADAMS Accession No. ML17226A352), NuScale provides additional information on the supplemental methods of chemistry analysis and the inventory mass balance method.

However, DCA Part 2, Tier 2, Section 5.2.5, did not describe or reference the information in the RAI response where it demonstrates leakage detection methods and associated capabilities. The staff requested the applicant in the follow-up RAI 9391 to update the DCA Part 2, Tier 2, Section 5.2.5 to reflect the information regarding supplemental methods and the capability to locate the leakage. In a response, dated May 1, 2018, to RAI 9391 (ADAMS Accession No. ML18121A452), NuScale discussed the bases and justification for not having detailed, in DCA Part 2, Tier 2, Section 5.2.5.1, the additional leak detection methods used in the design as described in the response to RAI 8915. NuScale also explained that it is not feasible for NuScale RCPB leakage detection systems to locate the leakage. A revised COL Item 5.2-7 reflected the limitation for locating the leakage based on the RAI response. Once a potential leak is identified, a containment entry is required to perform a visual inspection to determine the exact location. Additionally, a pressure test followed by a visual leakage exam (performed by the continuous monitoring of the CES in accordance with ASME Code, Section XI, IWA-5241(c)), is required for the Class 1 pressure-retaining boundary during each refueling outage before startup in accordance with ASME Code, Section XI, Division 1, Table IWB 2500-1 B-P. This test verifies the leak tightness of the RPV and connected piping in containment before reactor restart. Based on the above, the staff found that NuScale has adequately addressed RG 1.45, Regulatory Position C.2.3, on supplemental leakage detection systems. Based on above, RAI 8915 and RAI 9391 are resolved and closed.

5.2.5.4.3 Leakage Instrumentation in the Main Control Room

RG 1.45, Regulatory Position C.3.3, provides guidance on the leakage monitoring systems in the main control room (MCR).

> The plant should provide output and alarms from leakage monitoring systems in the main control room. Procedures for converting the instrument output to a leakage rate should be readily available to the operators. (Alternatively, these procedures could be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.)

The staff requested the applicant in RAI 8842 to address the leakage monitoring systems in the MCR. In the response dated July 26, 2017, to RAI 8842, Question 05.02.05-03 (ADAMS Accession No. ML17207A914), NuScale revised DCA Part 2, Tier 2, Section 5.2.5.1, to address the alarms in the MCR. Also, NuScale pointed out COL Item 5.2-7 to address the RAI with regards to the procedures. The staff found the RAI response acceptable. In addition, the NRC staff confirmed that the applicant implemented the marked-up changes in the RAI response in Revision 1 of DCA Part 2. Based on the above, the NRC staff found that NuScale has adequately addressed RG 1.45, Regulatory Position C.3.3, on leakage monitoring systems in the MCR.

5.2.5.4.4 GDC 2 and Seismic Qualification

GDC 2 requires that SSCs important to safety be designed to withstand the effects of seismic events and other natural phenomena without losing the capability to perform their intended safety functions. The RCPB leakage detection system detects leakage after an earthquake for an early indication of degradation so that corrective action can be taken before such degradation becomes severe enough to result in a leak rate greater than the capability of the makeup system to replenish the coolant loss. Application of GDC 2 to the RCPB leakage detection system ensures that plant operators have the capability to detect and respond to RCPB leakage after an earthquake. The prompt detection of, and response to, RCPB leakage after an earthquake reduces the possibility of a severe loss-of-coolant accident (LOCA). Specifically, RG 1.45 describes an acceptable method for RCPB leakage detection systems.

RG 1.45, Regulatory Position C.2.4, indicates that “at least one of the leakage monitoring systems required by the plant technical specifications…should be capable of performing its function(s) following any seismic event that does not require plant shutdown.”

In reviewing the RCPB leakage detection with respect to RG 1.45, Regulatory Position C.2.4, the staff noticed the information in DCA Part 2, Tier 2, Section 5.2.1, which states the following:

> The containment evacuation system (CES) inlet pressure instrumentation is designed to Seismic Category I and ensures that these components maintain the

> capability to perform their safety leak monitoring function during and after a safe shutdown earthquake. Therefore, the CES inlet pressure instrumentation is also capable of detecting changes in the containment atmospheric conditions, including leakage from the RCPB, during a seismic event that does not result in an NPM shutdown.

However, DCA Part 2, Tier 2, Table 3.2-1, indicates that, if the CES is being used for RCPB leakage detection, the system, other than the pressure indicator, is seismic Category III, which is not expected to survive a seismic event. The staff requested the applicant in RAI 8832 to demonstrate the CES leakage detection system has the capability to perform its intended function(s) following any seismic event that does not require plant shutdown.

In the response dated July 24, 2017, to RAI 8832, Question 05.02.05-1 (ADAMS Accession No. ML17205A648), the applicant provided additional information to address RG 1.45, Regulatory Position C.2.4, relating to the following requests:

- Clarify how the pressure instrument can "maintain the capability to perform its leak monitoring function" if the nonseismic portions of the CES fail following a safe shutdown earthquake.
- In the CES diagram, identify the location of the pressure instrument and the piping seismic classification associated with the pressure instrument.

In the RAI response, NuScale clarified its design as follows:

> In the event of a failure of the Seismic Category III portion of the Containment Evacuation System (CES), as the result of a seismic event, containment pressure may increase due to a failure of the CES vacuum pumps or CES pressure boundary. If the failure of the CES or leakage is sufficiently severe, this could result in a loss of vacuum event covered in Chapter 15.1.6 of the Final Safety Analysis Report. A loss of vacuum would result in a shutdown of the affected module and monitoring of leakage would no longer be required consistent with RG 1.45 position C.2.4. The CES inlet pressure instrumentation which is seismically qualified to Seismic Category I will be used as the primary means of leak detection following a seismic event that does not result in a module shutdown.
>
> …FSAR Figure 9.3.6-1 has been revised to include the CES inlet pressure instrumentation location.

The NRC staff reviewed the above responses and found them acceptable because the design is consistent with RG 1.45, Regulatory Position C.2.4, and the revised DCA Part 2, Tier 2, Figure 9.3.6-1, provided the requested information. Therefore, GDC 2, with respect to the leakage monitoring systems, is satisfied. The staff also confirmed that Revision 1 of DCA Part 2 contains the revised DCA Part 2 language.

5.2.5.4.5 Identified and Unidentified Leakage

In the current PWR designs, the reviewer verifies whether the provisions for collecting, detecting, and monitoring unidentified leakage are separate from those for identified leakage. NuScale DSRS Section 5.2.5 indicates that, if separation is not practicable for NuScale, all leakage will be conservatively assumed to be unidentified leakage. The total leakage flow rate will be established and monitored as specified in RG 1.45.

DCA Part 2, Tier 2, Section 5.2.5, states the following:

> For each NPM, distinguishing between RCS identified and unidentified leakage inside the containment is not practicable with the installed instrumentation …reactor coolant leakage, whether from a known or unknown source, into containment quickly vaporizes and disperses within the containment atmosphere. Upon vaporization, there is no feasible means to monitor separately the flow rates of identified and unidentified leakage from inside the containment. Therefore, containment leakage is treated as unidentified until the source location is known and leakage quantified by other means.

TS LCO 3.4.5 on RCS operational leakage specifies the limits for unidentified leakage and identified leakage to be 0.5 gpm and 2 gpm, respectively.

The staff found that the applicant's approach treating both identified and unidentified leakage as unidentified leakage is conservative because unidentified leakage has more stringent criteria. This is consistent with the guidance in DSRS Section 5.2.5.

5.2.5.4.6 Prolonged Low-Level Reactor Coolant System Leakage and Combined License Information Item

The operating experiences at Davis Besse Nuclear Power Station (NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002) indicated that prolonged low-level unidentified reactor coolant leakage inside containment could cause corrosion and material degradation such that it could compromise the integrity of a system, leading to the gross rupture of the RCPB. The regulatory position in RG 1.45 on operations-related positions provides guidance to address the issue. The plant should establish procedures for responding to prolonged low-level RCS leakage. The procedures should specify operator actions in response to prolonged low-level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the TS limits to provide operators sufficient time to take action before the TS limit is reached. The procedures would include identifying, monitoring, trending, and managing prolonged low-level leakage.

The applicant identified the following COL Item 5.2-7 to address the concern of the prolonged low-level RCS leakage.

> A COL applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.

The staff found proposed COL Item 5.2-7 to be acceptable because it is consistent with the guidance in RG 1.45, Revision 1, issued May 2008, about managing the prolonged low-level RCS leakage. In response to RAI 9391, the applicant revised the wording by changing "locating…" to "trending…" to resolve the inconsistency. The staff found this acceptable in Section 5.2.5.4.2 above, and this portion of RAI 9391 on inconsistencies is resolved and closed.

5.2.5.4.7 Intersystem Leakage

The regulatory positions in RG 1.45 state that the plant should monitor intersystem leakage for systems connected to the RCPB. NuScale DSRS Section 5.2.5 indicates that the applicant should identify all potential intersystem leakage paths and the instrumentation to monitor the intersystem leakage.

The applicant addressed the intersystem leakage in DCA Part 2, Tier 2, Section 5.2.5.3, "Reactor Pressure Vessel Flange Leak-Off Monitoring"; DCA Part 2, Tier 2, Section 5.2.5.4, "Reactor Safety Valve and Emergency Core Cooling System Valve Leakage Monitoring"; DCA Part 2, Tier 2, Section 5.2.5.5, "Chemical and Volume Control System Intersystem Leakage Monitoring"; DCA Part 2, Tier 2, Section 5.2.5.6, "Reactor Component Cooling Water System Leakage Monitoring"; and DCA Part 2, Tier 2, Section 5.2.5.7, "Primary to Secondary Leakage Monitoring."

The staff reviewed the above information and determined it is acceptable because the proposed approach is consistent with the guidance in NuScale DSRS Section 5.2.5, Table 1, "Systems and Components Connected to Reactor Coolant System and Needing Inter-System Leakage Monitoring."

5.2.5.5 Initial Test Program

DCA Part 2, Tier 2, Table 14.2-41, Test #41-3, addresses the initial test program for RCS leakage detection monitoring capability. The test will verify the CES level instrumentation and CES pressure instrumentation capability to detect unidentified RCS leakage of 1 gpm within 1 hour. SER Section 14.2 documents the staff evaluation of the initial test program for the DC review.

5.2.5.6 Inspections, Tests, Analyses, and Acceptance Criteria

DCA Part 2, Tier 1, Table 2.3-1, provides the ITAAC for RCS leakage detection. This ITAAC calls for inspections to verify the design of the RCS leakage detection systems. The inspection includes (1) verifying the CES level increase in the CES tank, which correlates to a detection of an unidentified RCS leakage rate of 1 gpm within 1 hour, and (2) verifying the CES pressure increase in the CES inlet pressure instrumentation, which correlates to a detection of an unidentified RCS leakage rate of 1 gpm within 1 hour.

SER Section 14.3 discusses all NuScale ITAAC.

5.2.5.7 Technical Specifications

RG 1.45, Regulatory Position C.4.1, provides guidance on the content of TS addressing RCPB leakage by stating the following:

Plant technical specifications should include the limiting conditions for identified, unidentified, RCPB, and intersystem leakage, and they should address the availability of various types of instruments to ensure adequate coverage during all phases of plant operation (not including cold shutdown and refueling modes of operation).

NuScale DCA Part 4 provides plant TS in LCO 3.4.5 to specify allowable operational leakage limits and in LCO 3.4.7 to specify operability requirements for instruments of diverse monitoring principles during plant operating modes 1, 2, and 3:

LCO 3.4.5: RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,
b. 0.5 gpm unidentified LEAKAGE,
c. 2 gpm identified LEAKAGE from the RCS, and
d. 150 gallons per day primary to secondary LEAKAGE.

LCO 3.4.7: Two of the following RCS leakage detection instrumentation method shall be OPERABLE:

a. Two CES condensate channels,
b. Two CES inlet pressure channels, and
c. One CES gaseous radioactivity monitor channel.

In RAI 9276, the staff requested the applicant demonstrate that the action statements in TS LCO 3.4.7 provide at least two detection methods that have the acceptable quantitative detection capabilities are operable, considering the gaseous radioactivity monitor does not have the acceptable quantitative detection capability. In the response dated January 22, 2018, to RAI 9276 (ADAMS Accession No. ML18022A473), NuScale responded to an inadequacy in the logic of action statement of TS LCO 3.4.7 such that RG 1.45, Regulatory Position C.2.3, is not satisfied. The plant TS should identify at least two independent and diverse methods that have the detection capabilities specified in Regulatory Position C.2.2. However, one of the detection methods, radioactivity monitor, does not have the capabilities specified in Regulatory Position C.2.2. Additionally, the response dated October 18, 2017, to RAI 9051, Question 16-25 (ADAMS Accession No. ML17291A299), included modifications to LCO 3.4.7 conditions and required actions. The NRC staff reviewed the revised LCO 3.4.7 and found it acceptable because it addresses Regulatory Position C.2.3, in that at least two independent methods have the acceptable quantitative detection capabilities specified in Regulatory Position C.2.2.

Based on the above, the staff determined the proposed two TS LCOs are acceptable in meeting RG 1.45, Regulatory Position C.4.1, relating to the TS requirement for RCPB leakage detection.

However, in reviewing the LBB application for main steam system (MSS) and feedwater system (FWS) line breaks, the NRC staff identified inadequacies in the NuScale TS, as discussed in the following section.

<u>Leak-Before-Break Application and Limiting Condition for Operation</u>

NuScale DCA Part 2, Tier 2, Sections 5.2.5.2, and 9.3.6.2.3, "System Operation," indicate that the leak-detection methods of CNV pressure monitoring, CES sample tank level change monitoring, and CES vacuum discharge radiation monitoring are used for LBB leakage monitoring. NuScale applies LBB for main steam and feedwater piping within the CNV, as described in DCA Part 2, Tier 2, Section 3.6.3.5.

In RAI 8843, Question 05.02.05-04, the staff asked the applicant to clarify whether TS LCO 3.4.5 is applicable to the MSS and FWS lines and, if not, to justify its decision not to have a TS LCO for the LBB application. It should be noted that all previously certified designs and licensed operating plants that apply LBB technology, including MSS leakage, have a TS LCO on the leakage limit.

In the response to the RAI, dated July 26, 2017 (ADAMS Accession No. ML17207A906), the applicant explained that TS LCO 3.4.5 is not applicable to MSS and FWS LBB. Further, NuScale justified its position of not having TS LCO by its assessment of the TS LCO applicability in accordance with 10 CFR 50.36(c)(2)(ii) because MSS and FWS leakages do not meet Criteria 1 and 2 of the referenced regulation. The staff disagreed with NuScale's assessment on meeting Criterion 2 and also found that it did not address Criterion 4.

In addition, the staff found that NuScale's characterization of the LBB leakage limit as solely "an indicator for the need to take further action" is not correct. The LBB leakage limit is related to the critical crack size in the LBB analyses described in DCA Part 2, Tier 2, Section 3.6.3, "Leak-Before-Break Evaluation Procedures." Beyond the critical crack size, the crack growth becomes unstable, and the success of LBB to prevent gross pipe failures (i.e., high-energy pipe breaks) cannot be assured. When the limit is exceeded, pipe breaks could occur. Consequently, the dynamic effects resulting from those potential high-energy pipe breaks should be evaluated to meet the GDC 4 requirement to ensure that nearby SSCs important to safety are protected from the dynamic effects resulting from those postulated high-energy pipe breaks. It should be noted that the NuScale design does not have protection against the dynamic effects of MSS and FWS line breaks. This differs from current PWR designs. The dynamic effects (pipe whips and jet impingement) could lead to the failure of the instrument and failure of the integrity of the fission product barrier. Therefore, the combination of (1) exceeding LBB leakage limit and (2) the design of MSS and FWS line breaks without protection from the dynamic effects resulting from postulated high-energy pipe breaks may not be consistent with GDC 4 requirements.

Based on the above, the NRC staff determined that NuScale's justifications for "not having TS for LBB application" were not acceptable. Therefore, the staff closed RAI 8843, Question 05.02.05-04, and issued follow-up RAIs: RAI 9201, Question 05.02.05-07, as to the 10 CFR 50.36(c)(2)(ii) assessment, and RAI 9213, Question 03.06.03-11, on the LBB limit and compliance with GDC 4 when the limit is exceeded.

In RAI 9201, the staff requested the applicant to justify its position of not having TS LCO for LBB leakage limit by considering the requirements in 10 CFR 50.36(c)(2)(ii) and GDC 4. In the response dated April 13, 2018, to RAI 9201 (ADAMS Accession No. ML18103A097), NuScale provided a probabilistic risk assessment (PRA) from DCA Part 2, Tier 2, Chapter 19, to address Criterion 4 of 10 CFR 50.36(c)(2)(ii) and maintained its position that a TS LCO for LBB leakage limit is not needed based on 10 CFR 50.36(c)(2)(ii). In addition, NuScale referenced NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004, to support its position.

The staff found that the PRA is not adequate because NuScale ignored the dynamic effects from the pipe breaks resulting from exceeding the LBB limit. Exceeding the LBB limit, the high-energy line breaks associated with dynamic effects could occur such that the success of ECCS and module protection system instrumentation could not be ensured. Because the referenced PRA in DCA Part 2, Tier 2, Chapter 19, took credit for the success of the ECCS and module protection system (MPS) without any justification, the referenced PRA analyses are not adequate to address the subject issue because the ECCS and MPS assumptions being used in the PRA may not be valid. Therefore, the NRC staff determined that the applicant has not adequately justified the applicability of its PRA analyses for Criterion 4 of 10 CFR 50.36(c)(2)(ii).

In addition, NuScale stated that its conclusion is consistent with the NRC staff's safety evaluation of a certified design TS LCO 3.7.8, "Main Steam Line Leakage," as described in NUREG-1793, which states that "the main steam line leakage limit…does not satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii)…."

Although NUREG-1793 is a safety evaluation report and does not contain NRC requirements, the staff found that NuScale misinterpreted the conclusion in NUREG-1793 relating to an MSS leakage LCO of a certified design. NUREG-1793 states the following:

> [T]he [plant design] applies leak-before-break technology to the main steam line and the primary coolant system, while current operating PWRs only apply this technology to the primary coolant system. New specifications 3.7.8 for main steam line leakage, is provided to account for these differences. … The main steam line leakage limit does not affect a fission product barrier and is not an initial condition of a DBA. Accordingly, this limit does not satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii), but is included in the TS for defense in depth. <u>Therefore, TS 3.7.8 is acceptable</u>.

It should be noted that the key conclusion of NUREG-1793, as described above, is that TS 3.7.8 is acceptable. To appropriately apply NUREG-1793, there should be a TS LCO for the main steam leakage. When the leakage exceeds a specified limit, the plant is required by the TS to be in mode 3 within 6 hours and in mode 5 within 36 hours. NuScale's position that the LCO is not needed, using NUREG-1793 as a basis, omits a vital element of the resolution achieved through the NUREG.

Due to the disagreement described above, the NRC staff performed an independent reevaluation of the applicability of Criterion 2 in 10 CFR 50.36(c)(2)(ii) for LBB application, as described below.

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria: …
> (B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The NuScale LBB leakage limit is an operating restriction that is an initial condition of the LBB analysis. Beyond the leakage limit, the postulated MSS/FWS pipe break could occur, and the NuScale design does not have protection against the dynamic effects. The MSS/FWS pipe breaks could result in the failure of a fission product barrier (a closed system functioning as a containment boundary), and the resulting dynamic effects could challenge the integrity of a fission product barrier (primary coolant system). Therefore, the NRC staff finds that 10 CFR 50.36(c)(2)(ii), Criterion 2, is met, and an LCO must be established.

Additionally, as discussed in RAI 9201, it is unclear to staff how the lack of a TS LCO for the LBB leakage limit assures compliance with GDC 4. The dynamic effects resulting from the potential pipe breaks should be evaluated to meet the GDC 4 requirement such that nearby SSCs important to safety are protected from the dynamic effects resulting from the postulated high-energy pipe breaks.

The applicant responded to the concern in the response to RAI 9201 as follows:

> If a leak were detected in excess of the LBB limit, the licensing basis for the facility would no longer be met because the feedwater and main steam line piping would no longer satisfy its design basis. Plant procedures will require the facility to address and resolve such a failure in a timely manner consistent with Appendix B of 10 CFR 50. Indication of a leak from the main steam or feedwater piping inside containment would require resolution by placing the unit in a safe condition—in this case depressurizing the piping by shutdown of the unit. No additional control to assure conformance with the licensing and design basis of the plant, nor to correct a situation in which the LBB limit is challenged is required.

The staff reviewed the above responses, taking credit for plant procedures that a future COL applicant will develop, and found them unacceptable to address compliance with GDC 4 for the following reasons:

- Plant procedures do not meet any of the provisions in SRP Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,” for dealing with the dynamic effects of high-energy line breaks or provide alternate means to satisfy GDC 4.

- NuScale DCA Part 2 does not include the protection design and information from MSS and main feedwater pipe break analyses to describe the dynamic effects of the high-energy line pipe breaks for development of the procedures.

- The plant procedures do not explicitly require operators to shut down the plant within a specified time when the limit is exceeded.

- The plant procedures do not require NRC oversight.

In addition, the applicant addressed the concern in the response to RAI 9213, dated January 25, 2018 (ML18026A652) using a proposed COL information item as an alternative method to address staff’s concern regarding compliance with GDC 4. The safety evaluation of the response to RAI 9213, Question 03.06.03-11 is discussed below. The NRC staff determined the applicant’s response to RAI 9201, Question 05.02.05-07, to be an open unresolved item.

By the letter dated December 20, 2018 (ADAMS Accession No. ML183548172), NuScale provided a supplemental response to RAI 9201 and proposed a new LCO 3.7.3, “In-Containment Secondary Piping Leakage,” to address the NRC staff’s concerns. The staff reviewed the LCO and identified the following remaining concerns with respect to the proposed LCO:

- The second paragraph of page B.3.7.3-2 of the proposed LCO 3.7.3 states that, "although the in-containment secondary system piping leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria, this specification has been included…." The NRC staff has reviewed NuScale's justifications for the above statement in the responses to RAI 9201 and RAI 9213 and concluded that, for the LBB application, the secondary system piping leakage limit is required by 10 CFR 50.36(c)(2)(ii), Criterion 2. Even though the proposed LCO 3.7.3 addresses the NRC staff's concern in general, the above statement in page B.3.7.3-2 contradicts the staff's finding and does not provide an acceptable basis for the proposed TS. The NRC staff believes that this statement should be removed, and the RAI responses should be revised accordingly.

- The initial TS surveillance frequency of SR 3.7.3.1 needs to be provided so that the staff and licensee will know the value/starting point for use with the Surveillance Frequency Control Program.

- The proposed leakage limit of "1.5 gallons per hour" is currently under NRC evaluation relating to RAI 9213. At this point, staff found that the data used to develop the bounding analysis curves (BACs) are not the most recent. NuScale is in the process of performing additional seismic analysis. Staff is currently waiting for the new data so it can perform the confirmatory analysis for the new BACs proposed by NuScale.

In conclusion, the staff is tracking **RAI 9201 as Open Item 5.2.5-3** pending resolution of the above identified concerns. The NRC staff is preparing a letter requesting the applicant to address the above staff's concerns. The response to RAI 9213 is being evaluated below.

As discussed above in the sections regarding RAI 8843 and RAI 9201, the staff refers to the safety evaluation on the response to RAI 9213, Question 03.06.03-11, for SER Section 3.6.3.4 to support the LBB leakage detection evaluation of DCA Part 2, Tier 2, Section 3.6.3.5. In the response to RAI 9213, NuScale revised DCA Part 2, Tier 2, Section 3.6.3.5, to state that "the LBB leak detection availability and limits will be included in the owner-controlled requirements manual" and proposed the following new COL Information Item 16.1-2:

> A COL applicant that references the NuScale Power Plant design certification will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.

The NRC staff reviewed the proposed COL item and determined that the applicant did not adequately address the staff's specific concern related to compliance with GDC 4 requirements. The staff's determination is based on the safety evaluation in SER Section 3.6.3.4 discussed below.

GDC 4 requires, in part, that SSCs important to safety be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic effects of postulated high-energy line pipe ruptures, including MSS and FWS line breaks. SRP

Section 3.6.2 provides guidance on the protection of the dynamic effects of postulated high-energy line ruptures. NuScale does not include this protection in its design. Nevertheless, NuScale uses the LBB provision allowed by SRP Section 3.6.2 to address the GDC 4 requirement. If an applicant proposes to use LBB technology to exclude the dynamic effects of postulated pipe ruptures from the design basis of plant SSCs, the staff will review the applicant's design and analyses to determine the acceptability of the application of the LBB provision.

NuScale DCA Part 2, Tier 2, Section 3.6.3, describes LBB application for MSS and FWS lines that the plant's capability detects a leak in the piping components well before the onset of the unstable crack growth. The applicant determined a critical crack size associated with the LBB leakage limit in DCA Part 2, Tier 2, Section 3.6.3. The staff found that beyond the LBB leakage limit, the leaks may become breaks due to the limitation of fracture mechanics in the LBB analysis.

Further, staff found that for LBB beyond the critical crack size (corresponding to a leakage limit), the crack growth becomes unstable and can propagate catastrophically, such that the success of LBB to prevent gross pipe failures cannot be assured. The LBB analyses in NuScale DCA Part 2, Tier 2, Section 3.6.3, do not cover the range beyond this limit. Beyond this limit, the NuScale design does not have an applicable LBB analysis, nor the protection design to assure compliance with GDC 4. As a result, the staff finds that the jet impingement and pipe whips could cause the failure of the primary coolant pressure boundary, reactor protection instrumentation, and ECCS. Therefore, the LBB limit could be a safety-significant parameter. Moreover, the NuScale design does not have a TS LCO to limit the operation beyond the LBB leakage limit that other NRC-certified designs have. Therefore, without shutting down the plant, it is unclear that NuScale plants will continue to meet GDC 4 when the LBB limit is exceeded. The staff is concerned about continued operation when the LBB limit is exceeded. The TS LCO would prevent plant operation in an unanalyzed, unsafe condition and enables the applicant to meet GDC 4 without pipe break analyses beyond the critical crack size.

In the RAI 9213 responses, the applicant did not address the NRC staff's safety concern regarding GDC 4 and mischaracterized the concern as "additional assurance of the reliability and availability of the LBB detection capability." The staff's concern, which was clearly discussed and emphasized in RAI 9201 and RAI 9213, is the unanalyzed pipe breaks beyond the LBB leakage limit, where the LBB analyses in DCA Part 2, Tier 2, Section 3.6.3, are not applicable. The staff needs assurance that GDC 4 is met.

The proposed COL item, as an alternative method to meeting GDC 4, does not explicitly direct the operator to shut down the plant within a specified time when the limit is exceeded. The proposed owner-controlled procedure manual does not require NRC oversight or meet any of the provisions in SRP Section 3.6.2 for dealing with the dynamic effects of high-energy line breaks or propose an alternative method to satisfy GDC 4. Therefore, the NRC staff found that the proposed COL item, requiring an owner-controlled procedure manual, was not acceptable. By letter dated December 20, 2018 (ADAMS Accession No. ML183548172), NuScale provided a supplemental response to RAI 9201 and proposed a new LCO 3.7.3. The proposed LCO does address the safety concern identified in RAI 9213 on compliance with GDC 4. However, the staff reviewed the LCO relating to RAI 9201 and identified several concerns with respect to the proposed LCO, including the specified leakage limit of "1.5 gallons per hour," which is currently under NRC evaluation relating to RAI 9213. The NRC staff believes that NuScale's RAI response incorrectly characterizes the staff's concern as focused on instrumentation reliability. In conclusion, the staff is tracking **RAI 9213 as Open Item 5.2.5-4**.

5.2.5.8 Combined License Information Items

SER Table 5.2.5-1 lists COL information item numbers and descriptions from DCA Part 2, Tier 2, Table 1.8-2.

Table 5.2.5-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 5.2.5

Item No.	Description	DCA Part 2 Tier 2 Section
5.2-7	A COL applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.	5.2.5

The staff found the proposed COL information item acceptable, based on the review in SER Section 5.2.5.4.6.

5.2.5.9 Conclusion

The staff finds it was unable to finalize its conclusions as to acceptability because of the open item related to RCPB leakage detection and LBB leakage detection.

5.3 Reactor Vessel

An NPM consists of a reactor core, two SGs, and a PZR within a single RV within a CNV that surrounds the RV. The NPM includes the piping connecting the RV and CNV.

The RV is a pressure-retaining vessel component of the RCS. The RV forms part of the RCPB and is a barrier to the release of fission products. The RV contains the reactor core, RVIs, SGs, PZR, and reactor coolant volume. The RV is supported laterally and vertically by the CNV. The RV provides support and attachment locations for the control rod drive mechanisms (CRDMs), the CRDM seismic support structure, PZR heater bundles, in-core instrumentation, SG system piping, RCS piping, RSVs, reactor vent valves, and reactor recirculation valves.

5.3.1 Reactor Vessel Materials

5.3.1.1 Introduction

This section addresses material specifications, special processes used for the manufacture and fabrication of components, special methods for NDE, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, material surveillance, and RV fasteners. This section of the DCD should contain pertinent data in sufficient detail to provide assurance that the materials (including weld materials), fabrication methods, and inspection techniques used for the RV and applicable attachments and appurtenances conform to all applicable regulations. SER Section 5.2.3 addresses other RCS materials.

5.3.1.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1.1, “Design Description,” describes the NPM and includes other information associated with this section.

DCA Part 2, Tier 2: DCA Tier 2, Section 5.3.1, “Reactor Vessel Materials,” describes the materials used in the RV, as summarized, in part, below.

DCA Part 2, Tier 2, Section 5.3.1, as supplemented by letters dated June 21, 2017; September 7, 2017; December 12, 2017; March 9, 2018; March 21, 2018; and April 6, 2018 (ADAMS Accession Nos. ML17172A744, ML17250A838, ML17346A519, ML18068A622, ML18080A176, and ML18096B882, respectively), addresses material specifications, special processes used to manufacture and fabricate components, special methods for NDE, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, material surveillance, and RV fasteners.

The RV is fabricated in accordance with ASME Code, Section III, requirements as delineated in DCA Part 2, Tier 2, Section 5.3.1.1, “Material Specifications.” Forged low-alloy steel is used to fabricate the assembly shells that surround the reactor core, PZR, and SGs. Forgings are used to form the various geometries, minimizing the amount of welding. This low-alloy steel is clad with austenitic stainless steel to minimize corrosion. The outside surfaces of the RV are clad with a minimum of one layer while the inside and sealing surfaces are clad with a minimum of two layers. This cladding is performed consistent with the guidance in RG 1.43, Revision 1, issued March 2011. All weld cladding processes are qualified in accordance with ASME Code, Section III, Subarticle NB-4300.

NDE is performed consistent with ASME Code, Section V, as modified by Section III, with several additional requirements. All surfaces to be clad are examined by magnetic particle or liquid penetrant testing, consistent with ASME Code, Section III, Paragraphs NB-2545 or NB-2546, before cladding. Preservice examinations include 100 percent of the pressure boundary welds and are to conform to ASME Code, Section III, Subsubarticle NB-5280, and Section XI, Subarticle IWB-2200, using the examination methods in Section V, as modified by Paragraph NB-5111 of ASME Code, Section III.

Welding of ferritic steels is conducted in accordance with ASME Code, Section III, Subarticle NB-4300, and Section XI, as elaborated in DCA Part 2, Tier 2, Section 5.2.3.3, “Fabrication and Processing of Ferritic Materials.” Welding of austenitic stainless steel is conducted in accordance with ASME Code, Sections III and XI, as elaborated in DCA Part 2, Tier 2, Section 5.2.3.4, “Fabrication and Processing of Austenitic Stainless Steels.” Electroslag welding is used only for cladding and is performed consistent with RG 1.43, Revision 1.

Fracture toughness properties of the RCPB comply with the requirements of 10 CFR Part 50, Appendix G, and ASME Code, Section III, Subarticle NB-2300. The RV is designed against nonductile fracture in accordance with ASME Code, Section III, Appendix G; ASME Code, Section XI, Appendix G; and 10 CFR Part 50, Appendix G.

A material surveillance program, consistent with ASTM E185-82, “Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” monitors the changes in the fracture toughness properties of the RV, as required by 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

The RV closure studs, nuts, and washers are to be fabricated from SB-637 Alloy 718 to prevent general corrosion during immersion. DCA Part 2, Tier 2, Section 3.13, “Combined License Information Items,” contains further information on fasteners.

ITAAC: DCA Part 2, Tier 1, Table 2.1-4, contains the ITAAC associated with DCA Part 2, Tier 2, Section 5.3.1. Table 2.1-4, Item 2, indicates that inspections will be performed on as-built component data reports to conclude that NPM ASME Code Class 1 and 2 components are constructed in accordance with the requirements in ASME Code, Section III. DCA Part 2, Tier 1, Table 2.1-4, Item 6, indicates that the RV beltline material charpy USE will be confirmed to be 101.6 J (75 ft-lb) or greater. DCA Part 2, Tier 1, Table 2.1-4, Item 12, indicates that an inspection of the RV is performed to ensure that it contains at least four surveillance capsule holders.

Technical Specifications: There are no TS for this area of review.

5.3.1.3 Regulatory Basis

SRP Section 5.3.1, “Reactor Vessel Materials,” contains the relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, as follows, as well as the review interfaces with other SRP sections:

- GDC 1 and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing of SSCs
- GDC 4, as it relates to the environmental compatibility of components
- GDC 14, as it relates to the prevention of rapidly propagating failures of the RCPB
- GDC 31, as it relates to material fracture toughness
- GDC 32, as it relates to the requirements for a materials surveillance program
- 10 CFR 50.55a, as it relates to quality standards for design, as well as determination and monitoring of fracture toughness
- 10 CFR 50.60, “Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,” as it relates to the RCPB fracture toughness and material SRs of 10 CFR Part 50, Appendix G and Appendix H
- 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control
- 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness

- 10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness

For the review of DCA Part 2, Tier 2, Section 5.3.1, the staff has determined that the use of ASME Code, Section III, appropriately supports GDC 1 without supplement.

5.3.1.4 Technical Evaluation

The staff reviewed NuScale DCA Part 2, Tier 2, Section 5.3.1, using SRP Section 5.3.1. The ASME Code of record for NuScale is the 2013 Edition. Subject to the conditions of 10 CFR 50.55a, ASME Code, Section III, Subsection NB, contains the construction requirements for the RV. The applicant noted that the RV is certified and stamped in accordance with ASME Code, Section III, Article NCA-8000.

5.3.1.4.1 Materials Specifications

The materials specifications for the RV are acceptable if they are in accordance with ASME Code, Section III, Article NB-2000. ASME Code, Section III, Paragraph NB-2121, states that pressure-retaining material shall conform to the requirements of one of the specifications for material given in ASME Code, Section II, Part D, Subpart 1, Tables 2A and 2B. ASME Code, Section III, Paragraph NB-2128, states that materials for bolts and studs shall conform to the requirements of one of the specifications listed in ASME Code, Section II, Part D, Subpart 1, Table 4. The ASME Code also states that welding and brazing material shall comply with an SFA specification in ASME Code, Section II, Part C, except as otherwise permitted in ASME Code, Section IX.

DCA Part 2, Tier 2, Section 5.3.1.1, states that DCA Part 2, Tier 2, Table 5.2-4, lists the materials used in the RV. Ferrous materials used to fabricate the RV include SA-508, Grade 3, Class 1 (lower RV flange shell, RV bottom head, and core support blocks); SA-508, Grade 3, Class 2 (RV top head, PZR shell, upper RV flanged transition shell, and upper/lower RV SG shells); and an assortment of Type 304/304L, F304/F304L, 316, 316L components. Nickel-chromium-ion (NiCrFe) alloy 690 is used for PZR pressure taps, thermowell nozzles, safe ends, and the PZR closure flange. The instrumentation and control (I&C) access port cover RV closure flange, and RSV flange threaded fasteners are SB-637, alloy 718. Based on the review of the information described above, the staff determined that the material specifications are acceptable because they meet the requirements of ASME Code, Section III.

DCA Part 2, Tier 2, Table 5.2-4, also specifies the weld materials used in the RV, and DCA Part 2, Tier 2, Section 5.3.1.2, "Special Processes Used for Manufacture and Fabrication of Components," states that welding materials for the RV conform to ASME Code, Section II, and ASME Code, Section III, or satisfy the requirements for other welding materials as permitted in ASME Code, Section IX. The weld materials used to fabricate the RV include SFA 5.4, 5.5, 5.9, 5.11, 5.14, 5.22, 5.23, 5.28, and 5.29. Based on the review of the information described above, the staff determined that the material specifications are acceptable because they meet the requirements of ASME Code, Section III.

The applicant described all RCPB materials as complying with 10 CFR Part 50, Appendix A, GDC 1, 4, 14, 30, and 31; and 10 CFR Part 50, Appendix G. Specifically, the applicant noted that the RV is fabricated according to ASME Code, Section III, Article NB 4000, except the SG tube supports (NG-4000); and the RV supports and CRDM seismic supports, which are fabricated in accordance with ASME Code, Section III, Article NF 4000. Unstabilized austenitic stainless steel that is welded or exposed to sensitizing temperatures is to have a maximum carbon content of 0.03 weight percent, consistent with RG 1.44, Revision 1, issued March 2011. Finally, COL Item 5.3-1 requires applicants to establish measures to control onsite cleaning of the RV during construction, in accordance with RG 1.28. The staff determined that the applicant appropriately conformed to the above because it complies with the pertinent guidance in the SRP and RG 1.44.

5.3.1.4.2 Special Processes Used for Manufacture and Fabrication of Components

The special processes used for the manufacture and fabrication of the RV are acceptable if they are in accordance with ASME Code, Section III. Special processes that do not have ASME Code requirements are reviewed on a case-by-case basis.

DCA Part 2, Tier 2, Section 5.2.3, states that the RCPB materials, including the RV, conform to the fabrication, construction, and testing requirements of ASME Code, Section III, Subsection NB. Cladding of the low-alloy steel components of the RV is to conform to ASME Code, Section III, Article NB-4300, and RG 1.43, Revision 1, as documented through reference to DCA Part 2, Tier 2, Section 5.2.3.

DCA Part 2, Tier 2, Section 5.3.1.2, indicates that measures are taken to prevent sensitization of austenitic stainless steel materials through appropriate use of ASME Code, Section II, heat treatment parameters. The subject materials are to be either water quenched or cooled quickly enough through the sensitization range to avoid sensitization. When means other than water quenching are used, nonsensitization of the base material is to be verified by Practice A or E of ASTM A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels.” Welding practices and material compositions are controlled to manage sensitization. Where unstabilized Type 3XX austenitic stainless steels are subjected to sensitizing temperatures for more than 60 minutes during postweld heat treatment, nonsensitization is to be verified in accordance with Practice A or E of ASTM A262, as recommended by RG 1.44. The staff has reviewed this information and found it to be acceptable because it meets the requirements of ASME Code, Section III, and is consistent with the recommendations of RG 1.44, Revision 1.

5.3.1.4.3 Special Methods for Nondestructive Examination

DCA Part 2, Tier 2, Section 5.3.1.4, “Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels,” states that RV pressure-retaining and integrally attached materials examinations are to meet the requirements specified in ASME Code, Section III. The DCD further details inspection of clad surfaces and aspects of preservice examinations. The staff discusses this topic in more detail in its review of DCA Part 2, Tier 2, Section 5.2.4. The staff reviewed the information in DCA Part 2, Tier 2, Section 5.3.1.4, for accuracy and consistency with DCA Part 2, Tier 2, Section 5.2.4, and found it acceptable, as it is in accordance with ASME Code, Section III.

5.3.1.4.4 Special Controls for Ferritic and Austenitic Stainless Steels

DCA Part 2, Tier 2, Section 5.3.1.4, references DCA Part 2, Tier 2, Section 5.2.3.3, for welding of ferritic steel components in the RV, and DCA Part 2, Tier 2, Section 5.2.3.4, for welding of austenitic stainless steel components in the RV. DCA Part 2, Tier 2, Section 5.3.1.4, also notes that electroslag welding is not used for joining materials, rather only for cladding low-alloy steel in compliance with RG 1.34. The staff reviewed DCA Part 2, Tier 2, Sections 5.2.3.3 and

5.2.3.4, for welding for ferritic and austenitic steels, respectively. The staff reviewed the use of electroslag welding for cladding low-alloy steel and found it acceptable, as it is consistent with the recommendations of RG 1.34.

DCA Part 2, Tier 2, Section 5.3.1.4, references DCA Part 2, Tier 2, Sections 4.5.2.4, "Fabrication and Processing of Austenitic Stainless Steel Components" and 4.5.1.1, "Materials Specifications," concerning tools for abrasive work and use of cold worked austenitic stainless steel. The staff has documented its review of these topics in the associated DCD sections.

5.3.1.4.5 Fracture Toughness

DCA Part 2, Tier 2, Section 5.3.1.5, "Fracture Toughness," describes how the fracture toughness requirements of 10 CFR Part 50, Appendix G, are met for RV beltline materials in the NuScale design. The applicant stated that RCPB pressure-retaining materials will comply with the requirements of 10 CFR Part 50, Appendix G; ASME Code, Section III, Subarticle NB-2300; ASME Code, Section III, Appendix G; and ASME Code, Section XI, Appendix G. The DCD states that RV beltline materials have an initial minimum USE of 101.6 J (75 ft-lb) as determined by Charpy V-notch testing and are evaluated to ensure a minimum end-of-life Charpy V-notch USE of 67.8 J (50 ft-lb). The effects of neutron irradiation are taken into account in accordance with RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, issued May 1988. The staff determined that the information in the DCD is acceptable, as it meets the requirements of 10 CFR Part 50, Appendix G.

SER Section 5.3.2 documents the staff's evaluation of the applicant's predicted Charpy USE for the beltline materials.

5.3.1.4.6 Material Surveillance

GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV. Appendix H to 10 CFR Part 50 states that RVs that are projected to have a peak neutron fluence exceeding 1×10^{17} n/cm^2 (E > 1.0 MeV) must have their beltline materials monitored by a surveillance program complying with ASTM E185. The latest edition of ASTM E185 incorporated by reference into 10 CFR Part 50, Appendix H, is the 1982 Edition.

To meet the requirements of GDC 32, the NuScale design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the RV beltline materials to neutron radiation. In DCA Part 2, Tier 2, Section 5.3.1.6, "Material Surveillance," the applicant detailed the various aspects of the RV materials surveillance program, including material selection, the type and quantity of test specimens, the design of the surveillance capsules, capsule locations, and the withdrawal schedule. NuScale provided four identical surveillance capsule assemblies, of which three are for the initial 40-year operating license. DCA Part 2, Tier 2, Table 5.3-4 presents the type and quantity of specimens contained in each capsule, which are consistent with the requirements of ASTM E185-82.

DCA Part 2, Tier 2, Section 5.3.1.6, states that materials for the surveillance capsule program are those selected in accordance with ASTM E185-82. Test materials are prepared from the actual material used in fabricating the beltline region of the RV and include the base metal, weld metal, and heat-affected zone material. This is acceptable because it is meets the requirements of ASTM E185-82.

DCA Part 2, Tier 2, Section 5.3.1.6, and associated figures describe the design and location of the surveillance capsule assemblies as well as the associated lead factors. DCD Figure 5.3-2 shows a diagram of the capsule locations. The surveillance capsules are located to produce a lead factor (ratio of the neutron flux at the location of the capsule to that at the RV inner surface at the peak neutron fluence location) of approximately 2.5, consistent with the recommendations of ASTM E185-82.

DCA Part 2, Tier 2, Section 5.3.1.6, references TR-0116-20781, Revision 0, “Fluence Calculation Methodology and Results,” dated December 30, 2016 (ADAMS Accession No. ML17005A116) as the basis for projected neutron flux and fluence calculations. The information in the DCD is acceptable because it meets the requirements of ASTM E185-82.

DCA Part 2, Tier 2, Table 5.3-5, “Surveillance Capsule Withdrawal Schedule,” describes the surveillance capsule withdrawal schedule. The schedule is based on a design life of 60 years and calls for a capsule to be withdrawn at 6 effective full-power years (EFPYs), 15 EFPYs, and 32 EFPYs and at an end-of–life (EOL) neutron fluence between 1 and 2 times the RV peak 57-EFPY fluence. The staff notes that the withdrawal schedule for NuScale is an extension of the withdrawal schedule in Table 1 of ASTM E185-82 for a predicted transition temperature shift of less than 56 degrees C (100 degrees F). For a predicted transition temperature shift of less than 56 degrees C (100 degrees F), ASTM E185-82 requires a minimum of three surveillance capsules. However, the NuScale surveillance program provides four primary surveillance capsules because the design life of the NuScale RV (60 years) is longer that the design life defined in ASTM E185-82 (40 years or 32 EFPYs). The staff determined that the NuScale withdrawal schedule is consistent with the one recommended in ASTM E185-82. On this basis, the staff determined that the withdrawal schedule covering the initial license period within the NuScale surveillance capsule program was acceptable.

Based on the review described above, the staff determined that the description of the RV materials surveillance program for NuScale is acceptable because it is in accordance with ASTM E185-82 and thus meets the requirements of 10 CFR Part 50, Appendix H.

5.3.1.4.7 Reactor Vessel Fasteners

DCA Part 2, Tier 2, Section 5.3.1.7, “Reactor Vessel Fasteners,” states that the bolting material for the RV closure head is fabricated from SB-637, alloy 718. The use of austenitic nickel-based alloy bolting is to prevent general corrosion when the bolting is submerged during startup and shutdown processes. The staff evaluates this bolting material as part of its evaluation of DCA Part 2, Tier 2, Section 3.13.

DCA Part 2, Tier 2, Section 5.3.1.7, further discusses the RPV flange connection lock plates. The application details the design, materials, and testing related to these lock plates. The staff reviewed this material and finds it acceptable, based on the management of the impact of the lock plates on the RPV integrity and the requirements for nonstructural attachments in accordance with ASME Code, Section III, Subparagraph NB 1132.1(c)(2), and ASME Code, Section III, Paragraph NB 4435.

5.3.1.4.8 Combined License Information Item

DCA Part 2, Tier 2, Table 1.8-2 lists COL information item numbers and descriptions reproduced below in SER Table 5.3.1-1.

Table 5.3.1-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 5.3.1

Item No.	Description	DCA Part 2 Tier 2 Section
5.3(1)	A COL applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the RV during construction in accordance with RG 1.28.	5.3.1.1
5.3(3)	A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG-0800, Section 5.3.1.	5.3.1.6

The staff finds the above listing to be complete when COL Item 13.4-1 is considered, and that no additional COL information items are needed for this section. The staff notes that COL Item 13.4-1 requires a COL applicant to provide site-specific information, including an implementation schedule for operational programs such as the RV material surveillance program. The list above adequately describes actions related to the RV materials considerations that are necessary for the COL applicant for DCA Part 2, Tier 2, Section 5.3.1.

5.3.1.5 Inspections, Tests, Analyses, and Acceptance Criteria

ITAAC: The applicant gave the ITAAC associated with DCA Part 2, Tier 2, Section 5.3.1, in DCA Part 2, Tier 1, Table 2.1-4. DCA Part 2, Tier 1, Table 2.1-4, Item 2, indicates that as-built components conform to the rules of construction of ASME Code, Section III, and that reports exist that conclude that "The NuScale Power Module ASME Code Class 1 and 2 components conform to the rules of construction of ASME Code Section III." Because the ASME Code data reports will provide the required information that verifies the as-built components will be designed, constructed, inspected, and tested in accordance with ASME Code, Section III, the staff finds ITAAC Item 2 acceptable.

DCA Part 2, Tier 1, Table 2.1-4, Item 6, indicates that the RV beltline material has a Charpy USE of greater than 101.6 J (75 ft-lb). The ASME Code certified material test report associated with the RV beltline material will provide the information to confirm that the material meets this criterion. Because the test report provides the required information to verify the USE, the staff finds ITAAC Item 6 acceptable.

DCA Part 2, Tier 1, Table 2.1-4, Item 12, indicates that an inspection ensures that at least four surveillance capsules are in the RV. Because this ITAAC is consistent with ASTM E185-82 and the requirements of 10 CFR Part 50, Appendix H, the staff finds ITAAC Item 12 acceptable.

5.3.1.6 Conclusion

The staff concludes that the NuScale RV materials and associated manufacturing and fabrication processes, NDE methods, fracture toughness testing, and material surveillance meet the requirements of the ASME Code; 10 CFR 50.55a; and 10 CFR Part 50, Appendices G and H, which provide an acceptable basis for satisfying the requirements of GDC 1, 4, 14, 30, 31, and 32 for this section.

5.3.2 Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock

5.3.2.1 Introduction

Neutron radiation is known to cause embrittlement, or a reduction in ductility, in the RV. This degradation is most severe in the beltline region because of its exposure to the highest average neutron flux at power. This reduction in ductility is typically measured in terms of a change in nil-ductility transition reference temperature (RT_{NDT}) or a change in Charpy USE. To limit radiation embrittlement, controls are placed on the weight percentage of residual elements, such as copper, nickel, and phosphorus, in the materials used to fabricate the RV. An additional requirement placed on beltline materials is that they must maintain Charpy USE at an acceptable level throughout the life of the RV. In addition, pressure-temperature (P-T) limits are imposed on the RCS to provide adequate safety margins against nonductile fracture during normal operation; heatup; cooldown; AOOs; and system hydrostatic, preservice, and inservice leakage tests.

Pressurized thermal shock (PTS) events are potential transients in a pressurized-water RV that can cause severe overcooling of the RV wall, followed by immediate repressurization. The thermal stresses caused when the inside surface of the RV cools rapidly, combined with high-pressure stresses, will increase the potential for fracture if a flaw is present in a low-toughness material. The materials most susceptible to PTS are the materials in the RV beltline where neutron radiation gradually embrittles the material over time.

5.3.2.2 Summary of Application

DCA Part 2, Tier 1: There are no DCA Part 2, Tier 1, entries for this area of review.

DCA Part 2, Tier 2: The applicant described how it addresses P-T limits, PTS, and Charpy USE in DCA Part 2, Tier 2, Section 5.3.2, "Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses," summarized, in part, below.

DCA Part 2, Tier 2, Section 5.2.3, as supplemented by letter dated June 21, 2017 (ADAMS Accession No. ML17172A744), addresses P-T limits, PTS, and Charpy USE data and analyses. RCS P-T limits for normal heatup and criticality conditions, normal cooldown, and inservice leak and hydrostatic test rates shall be established and documented in accordance with the Generic Technical Specifications (GTS) in DCA Part 4, Volume 1, Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits." The detailed methodology for developing the P-T limit curves appears in NuScale TR-1015-18177-P. Generic P-T limit curves based on bounding RV material properties appear in DCD Figures 5.3-3, -4, and -5. A COL applicant that references the NuScale DC will develop P-T limit curves based on plant-specific data.

Bounding PTS and Charpy USE values are determined based on neutron fluence projections over a 60-year design life. A COL applicant that references the NuScale design is to verify the PTS reference temperature (RT_{PTS}) and USE values based on plant-specific material properties and neutron fluence.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: DCA Part 2, Tier 2, Chapter 16, "Technical Specifications," references the GTS associated with SER Section 5.3.2. In addition, GTS 5.6.4 specifies the content of the RCS pressure-temperature limits report (PTLR).

5.3.2.3 Regulatory Basis

SRP Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock," includes the relevant requirements of NRC regulations for this area of review and the associated acceptance criteria, as are summarized below, as well as review interfaces with other SRP sections:

- GDC 1, as it relates to quality standards for design, fabrication, erection, and testing
- GDC 14, as it relates to prevention of rapidly propagating failures of the RCPB
- GDC 31, as it relates to material fracture toughness
- GDC 32, as it relates to the requirements for a materials surveillance program
- 10 CFR 50.55a, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs important to safety
- 10 CFR 50.60, as it relates to RCPB fracture toughness and material SRs of 10 CFR Part 50, Appendix G and Appendix H
- 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," as it relates to fracture toughness criteria for PWRs relevant to PTS events
- 10 CFR Part 50, Appendix G, as it relates to material testing and acceptance criteria for fracture toughness

Acceptance criteria adequate to meet the above requirements include the following:

- RG 1.99, Revision 2, as it relates to RV beltline material properties
- RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," as it relates to the calculation of neutron fluence estimates

5.3.2.4 Technical Evaluation

5.3.2.4.1 Pressure-Temperature Limits

To address the requirements of 10 CFR Part 50, Appendix G, relating to P-T limits, the applicant submitted TR-1015-18177-P, for NRC review and approval. This PTLR follows the guidelines of Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits," dated January 31, 1996, and provides the bounding P-T limits, LTOP system limits, and the complete methodology for their development. The staff has documented its review of the PTLR in detail in a separate SER (ADAMS Accession No. ML19098B507 – separate SER).

DCA Part 2, Tier 2, Section 5.3.2.2, “Operating Procedures,” states that DCA Part 2, Tier 2, Section 13.5, “Plant Procedures,” addresses plant operating procedures to ensure that the P-T limits are not exceeded. COL Item 5.3-2 states that COL applicants referencing the DCD will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy has been demonstrated. The staff reviewed the COL item and finds it acceptable, as it supports conformance with the requirements of 10 CFR Part 50, Appendix G.

5.3.2.4.2 Pressurized Thermal Shock

PTS events are potential transients in a pressurized-water RV that can cause severe overcooling of the RV wall, followed by immediate repressurization. The thermal stresses, caused when the inside surface of the RV cools rapidly, combined with the high-pressure stresses, will increase the potential for fracture if a flaw is present in a low-toughness material. The materials most susceptible to PTS are the materials in the RV beltline where neutron radiation gradually embrittles the material over time.

The PTS Rule, 10 CFR 50.61, established screening criteria to serve as a limiting level of RV material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The rule gives screening criteria in terms of reference temperature, RT_{PTS}. The screening criteria are 132.2 degrees C (270 degrees F) for plates and axial welds, and 148.9 degrees C (300 degrees F) for circumferential welds. The following equation defines the RT_{PTS}:

$$RT_{PTS} = RT_{NDT\,(U)} + \Delta RT_{PTS} + M$$

where:

$RT_{NDT\,(U)}$ = initial reference temperature

ΔRT_{PTS} = mean value in the adjustment in reference temperature caused by irradiation

M = margin to be added to cover uncertainties in the initial reference temperature, copper and nickel contents, neutron fluence and calculational procedures

In DCA Part 2, Tier 2, Section 5.3.2.3, “Pressurized Thermal Shock,” the applicant provided RT_{PTS} values for the limiting materials in DCA Part 2, Tier 2, Table 5.3-8, “Pressurized Thermal Shock Screening Result.” Using the projected neutron fluence values at 60 years, the staff verified that the limiting RT_{PTS} values for all RV forgings and welds satisfied the PTS screening criteria of 10 CFR 50.61. On this basis, the staff determined that the design is acceptable because the RT_{PTS} values for all RV beltline materials meet the requirements of 10 CFR 50.61. In addition, DCA Part 2, Tier 2, COL Item 5.3-2, indicates that a COL applicant that references the NuScale DC will provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61 for RV beltline materials.

5.3.2.4.3 Charpy Upper-Shelf Energy

DCA Part 2, Tier 2, Section 5.3.2.4, “Upper-Shelf Energy,” states that RV beltline materials are evaluated to ensure a minimum Charpy V-notch USE value of 67.8 J (50 ft-lb) as required by

10 CFR Part 50, Appendix G, at 57-EFPY fluence. With an initial required USE of 67.8 J (50 ft-lb), the applicant determined that the EOL USE for the RV materials would be greater than 67.8 J (50 ft-lb). This determination was made consistent with RG 1.99, Revision 2, with the results shown in DCA Part 2, Tier 2, Table 5.3-9, "1/4-T Charpy Upper Shelf Energy per RG 1.99, Rev. 2." Using the applicant's projected neutron fluence and the copper and nickel contents for the base metal and welds, the staff verified that the reduction in USE resulted in EOL USE values that were greater than 67.8 J (50 ft-lb). On this basis, the staff finds that the applicant's USE values meet the requirements of 10 CFR Part 50, Appendix G, and are therefore acceptable.

5.3.2.5 Combined License Information Items

DCA Part 2, Tier 2, Table 1.8-2 lists COL information item numbers and descriptions reproduced below in SER Table 5.3.2-1.

Table 5.3.2-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 5.3.2

Item No.	Description	DCA Part 2 Tier 2 Section
5.3(2)	A COL applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated. These procedures will be based on material properties of the as-built reactor vessels.	5.3.2.2

The staff finds the above list to be complete and that it adequately describes actions necessary for the COL applicant. DCA Part 2, Tier 2, Table 1.8-2, needs no additional COL information items for P-T limits, PTS, or Charpy USE considerations.

5.3.2.6 Conclusion

The staff concludes that the P-T limits for the RCS for operating and testing conditions to ensure adequate safety margins against nonductile and rapidly propagating failure comply with the fracture toughness criteria of 10 CFR Part 50, Appendix G. Further, a material surveillance program, developed in compliance with 10 CFR Part 50, Appendix H, will verify the change in fracture toughness properties of the RV beltline materials during operation. The use of operating limits, as determined by the criteria defined in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur. This constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a, 60, and 61; 10 CFR Part 50, Appendix G; and GDC 1, 14, 31, and 32, with respect to P-T limits.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Introduction

Although the staff reviews most of the features and topics addressed in this section in other sections of this SER, principally 5.3.1 and 5.3.2, the integrity of the RV is of such importance that a special review of all factors relating to the integrity of the reactor is warranted to ensure complete and accurate review. The staff reviews the information in each area of the application for completeness and consistency with requirements to ensure RV integrity. The only information unique to DCA Part 2, Tier 2, Section 5.3.3 pertains to shipment and installation.

5.3.3.2 Summary of Application

DCA Part 2, Tier 1: There are no DCA Part 2, Tier 1, entries specific to this area of review.

DCA Part 2 Tier 2: DCA Part 2, Tier 2, Section 5.3.3, “Reactor Vessel Integrity,” describes RV integrity, as summarized, in part, below.

DCA Part 2, Tier 2, Section 5.3.3, provides references to the appropriate DCA Part 2, Tier 2, sections for design, materials of construction, fabrication methods, inspection requirements, operating conditions, inservice surveillance, and threaded fasteners. The only nonreferenced content concerns shipment and installation. The applicant stated that packaging, shipping, handling, and storing the RV are to be in accordance with ASME Code, Section III, Subparagraph NC A 4134.13, and will meet the requirements for Level C items in accordance with ASME NQA-1-2008/1A-2009 Addenda, Subpart 2.2. The use of a nonchloride, noncorrosive desiccant maintains a dry environment for all RV surfaces. Humidity indicators will be used during shipping. Both the primary and secondary sides of the RV will be maintained under positive pressure during shipment with the internal atmosphere of the SG tubes evacuated and filled with nitrogen. “Appropriate foreign material exclusion measures” will be taken for shipment.

ITAAC: There are no ITAAC entries specific to this area of review.

Technical Specifications: There are no TS specific to this area of review.

5.3.3.3 Regulatory Basis

SRP Section 5.3.3, “Reactor Vessel Integrity,” includes the relevant requirements of the NRC regulations for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- GDC 1 and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing of SSCs
- GDC 4, as it relates to the compatibility of components with environmental conditions
- GDC 14, as it relates to prevention of rapidly propagating failures of the RCPB
- GDC 31, as it relates to material fracture toughness
- GDC 32, as it relates to the requirements for a materials surveillance program
- 10 CFR 50.55a, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs important to safety
- 10 CFR 50.60, as it relates to RCPB fracture toughness and material SRs of 10 CFR Part 50, Appendix G and Appendix H

- 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control

- 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness

- 10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness

5.3.3.4 Technical Evaluation

The staff reviewed most areas separately in accordance with other SRP sections. SRP Section 5.3.3 provides the references that form the bases for this evaluation. The staff reviewed the information pertaining to design, materials of construction, fabrication methods, inspection requirements, operating conditions, inservice surveillance, and threaded fasteners in Sections 5.3.1 and 5.3.2, above. Those sections of this SER associated with the sections of DCA Part 2, Tier 2, referenced in DCA Part 2, Tier 2, Section 5.3.3, contain a detailed discussion of the staff's findings.

With regard to shipment and installation, the integrity of the RV is maintained by ensuring that the as-built characteristics of the vessel are not degraded by improper handling. DCA Part 2, Tier 2, Section 5.3.3.5, "Shipment and Installation," states that the requirements of ASME NQA-1, Subpart 2.2, Level C, are followed for the packaging and shipment of the RV. A dry environment is maintained for all RV surfaces by installation of a nonchloride, noncorrosive desiccant. Humidity indicators are to be used during shipment to monitor humidity. Both the primary and secondary sides of the RV are held under positive pressure during shipment. The internal atmosphere of both sides of the SG tubes are evacuated to eliminate residual moisture. Foreign material exclusion measures are taken commensurate with ASME NQA-1, Subpart 2.2, Level C. The information in the DCD is acceptable to the staff because proper cleanliness and freedom from contamination during all stages of shipping, storage, and installation of the RV are appropriately ensured through adherence to the recommendations in ASME NQA-1, Subpart 2.2, Level C.

5.3.3.5 Combined License Information Items

There are no COL information items for this section.

5.3.3.6 Conclusion

For DCA Part 2, Tier 2, Sections 5.3.1 and 5.3.2, and the PTLR referenced in DCA Part 2, Tier 2, Section 5.3.2, the staff concludes that the structural integrity of the NuScale RV is acceptable because it meets the requirements of GDC 1, 4, 14, 30, 31, and 32; 10 CFR Part 50, Appendices G and H; and 10 CFR 50.55a. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the NuScale plant conform to the applicable NRC regulations and the ASME Code. The NuScale design meets the fracture toughness requirements of the regulations and ASME Code, Section III, including requirements for surveillance of RV material properties throughout its service life. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with the regulations and the ASME Code.

5.4 <u>Reactor Coolant System Components and Subsystem Design</u>

5.4.1 Steam Generator Materials and Design

5.4.1.1 Introduction

The SGs transfer heat from the RCS to the secondary system to produce the steam required for turbine operation. For the NuScale design, the SG tubing is a large part of the RCPB because the reactor coolant circulates outside the SG tubes (primary side or shell side) with steam production occurring inside the SG tubes (secondary side).

5.4.1.2 Summary of Application

By application dated December 31, 2016 (ADAMS Accession No. ML17013A229), as supplemented by letters dated August 3, 2017 (ADAMS Accession Nos. ML17215A977 and ML17215A978); January 31, 2018 (ADAMS Accession No. ML18032A391); February 12, 2018 (ADAMS Accession No. ML18043B174); April 2, 2018 (ADAMS Accession No. ML18092B091); April 3, 2018 (ADAMS Accession No. ML18093B542); August 2, 2018 (ADAMS Accession No. ML18215A261); October 17, 2018 (ADAMS Accession No. ML18290B059); October 22, 2018 (ADAMS Accession No. ML18295A787); January 7, 2019 (ADAMS Accession No. ML19007A091), and April 2, 2019 (ADAMS Accession No. ML19098A236), the applicant submitted information related to the NuScale SMR SG program.

The staff has confirmed that Revision 2 of the DCD (ADAMS Accession No. ML18311A006) incorporates changes made before October 22, 2018. The staff will confirm that the final version of the DCD incorporates acceptable changes to the DCD submitted in the applicant's letters dated October 22, 2018, January 7, 2019, and April 2, 2019. The staff is tracking the changes as **Confirmatory Items 5.4.1-1, 5.4.1-2, and 5.4.1-3**.

DCA Part 2, Tier 1: In DCA Part 2, Tier 1, Section 2.1, the applicant described the components of the RCS, including the SGs. The Tier 1 information includes design information and ITAAC related to verification that the SGs meet ASME Code design requirements.

DCA Part 2, Tier 2: In DCA Part 2, Tier 2, Sections 5.2 and 5.4.1, the applicant described the SGs, including design, performance evaluation, materials selection, materials fabrication and processing, tests and inspections, compatibility with the primary and secondary water, flow-induced vibration (FIV), and design features for accessing the secondary side.

DCA Part 2, Tier 2, Section 5.4.1.1, "Design Basis," states that the NuScale SGs provide two safety-related functions—they form a portion of the RCPB and transfer decay heat to the DHRS. DCA Part 2, Tier 2, Section 5.4.1, states that the SG system consists of the feedwater piping from the containment system (CNTS) to the feed plenum access port, thermal relief valve, feed plenum access port and access cover, SGs, steam plenum cap, steam plenum access port and access cover, and main steam piping from the steam plenum access port to the CNTS. The SGs in the NuScale design are inside the RV, and the RV provides the SG shell and forms the SG's outer pressure boundary. Each NPM has two once-through SGs with helical tube columns that form an intertwined tube bundle around the upper riser assembly. Seamless helical tubes with no intermediate welds terminate at tubesheets in two feed and two steam plenums in each SG. DCA Part 2, Tier 2, Section 5.4.1, states that the feedwater plenum is within the feed plenum access port and the main steam plenum is within the RV integral steam plenum shell.

DCA Part 2, Tier 2, Section 5.4.1.2, "System Design," states that each helical tube has bends at each end that transition from the helix to a straight configuration at the entry to the tubesheets. The SG tubes are secured to the tubesheets by expansion fits and are welded to the tubesheets on the secondary side. The tube material is thermally treated (TT), nickel-based alloy 690 (alloy 690 TT, ASME SB-163), and each NPM has 1,380 helical tubes (690 tubes per SG). Reference to the SB-163 specification was included I Rev. 1 of DCA Part 2, Tier 2, Section 5.4.1.5, but not in Rev. 2. NuScale Power (ADAMS Accession No. ML19098A236), dated April 2, 2019, confirmed that reference to the SB-163 specification will be included in Rev. 3. This is **Confirmatory Item 5.4.1-3.** Each NPM has eight sets of SG supports (lower and upper SG supports) and SG tube supports (SG tube support assemblies that span the full height of the SG tube bundle, which include stamped tabs and middle and outer spacers) to provide vertical, lateral, vibration, and seismic support. To limit unstable flow oscillations, the design includes flow restrictors that extend into the tubes and are mounted on a plate in each feed plenum. To provide overpressure protection for the secondary-side of the SGs and the steam and feedwater piping inside containment during shutdown conditions, the design includes a single spring-operated, balanced-bellows thermal relief valve located on each feedwater line that vents directly into the containment.

The reactor coolant flows upward through the core and lower and upper riser assemblies. The reactor coolant exits the upper riser assembly and is redirected downwards into the SG region between the RV wall and the upper riser assembly. The reactor coolant on the outside of the SG tubes transfers heat to the secondary water on the inside of the SG tubes to produce steam. On the secondary side, feedwater flows from each feed plenum through the bottom of the SG tube columns, passes up and around the outside of the upper riser assembly, and is converted to steam. Steam plenums collect the steam from the top of the SG tube columns and direct it through the steam nozzles to the main steam and power conversion systems.

The applicant discussed FIV in DCA Part 2, Tier 2, Section 5.4.1.3, "Performance Evaluation." The staff also reviewed TR-0716-50439, Revision 1, "Comprehensive Vibration Assessment Program (CVAP) Technical Report," issued January 20, 2018. The staff reviewed this information under NuScale DSRS Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," and Section 3.9.2 of this SER contains the staff's evaluation.

ITAAC: As noted above in this section, DCA Part 2, Tier 1, lists the ITAAC related to the SGs.

Technical Specifications: There are no TS specifically related to SG materials and design. DCA Part 2, Tier 2, Chapter 16, contains the NuScale TS and bases related to the SG program. Specifically, in TS Sections 3.4.5, "RCS Operational LEAKAGE;" 3.4.9, "Steam Generator (SG) Tube Integrity"; 5.5.4, "Steam Generator (SG) Program"; and 5.6.5, "Steam Generator Tube Inspection Report"; and Bases Sections B 3.4.5, "RCS Operational LEAKAGE," and B 3.4.9, "Steam Generator (SG) Tube Integrity." The purpose of these TS and bases is to maintain SG tube structural and leakage integrity, and SER Section 5.4.2 discusses them further.

Technical Reports: As noted above in this section, the staff reviewed NuScale's CVAP TR, TR-0716-50439, Revision 1.

5.4.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 1 requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. If generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented to provide adequate assurance that these SSCs will perform their safety functions and that records will be maintained.

- GDC 4 requires, in part, that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents.

- GDC 14 relates to the RCPB being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 15 requires that the RCS and associated auxiliary control and protection systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

- GDC 30 requires, in part, that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

- GDC 31 requires, in part, that the RCPB be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner, thereby minimizing the probability of rapidly propagating fracture.

- 10 CFR 50.36, "Technical Specifications," applies to the SG program in the TS.

- 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e), generally require certain groupings of components, including those comprising the pressure boundaries, to meet the requirements of ASME Code, Section III.

- Appendix B to 10 CFR Part 50 applies to the SG materials. Of particular note is Criterion XIII, which requires, in part, that measures be established to control the cleaning of material and equipment in accordance with work and inspection procedures to prevent damage or deterioration.

- Appendix G to 10 CFR Part 50 requires that RCPB pressure-retaining components that are made of ferritic materials meet ASME Code requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs.

- 10 CFR 52.47(b)(1) requires that a DCA include 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 plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the AEA, and NRC regulations.

The guidance in NuScale DSRS Section 5.4.2.1, “Steam Generator Materials,” lists the acceptance criteria adequate to meet the above requirements, as summarized below, as well as review interfaces with other NuScale DSRS sections:

- ASME Code, Section II, Section III, and Section XI
- regulatory guidance related to the welding of SG components:
  - RG 1.31
  - RG 1.34
  - RG 1.43
  - RG 1.50
  - RG 1.71
- RG 1.36, as it relates to nonmetallic thermal insulation for austenitic stainless steel
- RG 1.28, as it relates to onsite cleaning and cleanliness controls
- RG 1.44, as it relates to the use of sensitized stainless steel
- RG 1.84, as it relates to ASME Code Case acceptability

5.4.1.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Section 5.4.1, in accordance with NuScale DSRS Section 5.4.2.1 to ensure that the integrity of the SG materials is maintained and that the SG materials meet the relevant requirements of GDC 1, 4, 14, 15, 30, and 31 and 10 CFR Part 50, Appendix B. These requirements are met through compliance with appropriate requirements of the ASME Code and conformance to guidance in RGs by specifying design features shown to preserve SG tube integrity and by specifying water chemistry practices that limit degradation of SG materials. The staff also reviewed supplemental information that the applicant provided in the letters previously identified in SER Section 5.4.1.2. SER Sections 5.4.2 and 3.9.2 discuss the staff’s review of the SG program and FIV, respectively.

5.4.1.4.1 Selection, Processing, Testing, and Inspection of Materials

The applicant described the SG materials proposed for the NPM in DCA Part 2, Tier 2, Section 5.2.3; Section 5.4.1.5, “Steam Generator Materials;” Table 5.2-4; and Table 5.4-3, “Steam Generator Piping, Tube and Piping Supports, and Flow Restrictor Materials.” The materials proposed are ferritic low-alloy steels, austenitic stainless steels, and nickel-based alloys. The staff reviewed the materials proposed in terms of their adequacy, suitability, and compliance with ASME Code, Sections II and III. As discussed in NuScale DSRS Section 5.4.2.1, for purposes of compliance with GDC 1 and 30, the materials used for the SGs are acceptable if they are selected, fabricated, tested, and inspected (during fabrication and manufacturing) in accordance with the ASME Code.

NuScale defined the quality groups in DCA Part 2, Tier 2, Section 3.2.2, “System Quality Group Classification,” and identified the design criteria in DCA Part 2, Tier 2, Table 3.2-1. DCA Part 2, Tier 2, Section 5.4.1.5, states that the SG system materials forming the RCPB are specified in accordance with ASME Code, Section II, and meet the requirements in ASME Code, Section III, Article NB-2000. DCA Part 2, Tier 2, Section 5.4.1.5, also states that the RCPB materials used in the SGs are classified Quality Group A and are designed, fabricated, constructed, tested, and inspected as ASME Code Class 1, and that the steam and feedwater piping from the CNTS to the SGs, thermal relief valves, steam plenum access ports, and plenum access covers are classified Quality Group B and are designed, fabricated, constructed, tested, and inspected as ASME Code Class 2. DCA Part 2, Tier 2, Section 5.4.1.5, further states that the welding of the

RCPB and shell side portions of the SG system is performed using procedures qualified in accordance with the applicable requirements of ASME Code, Section III, Subarticles NB-4300 and NC-4300, respectively, and Section IX. DCA Part 2, Tier 2, Section 5.4.1.2, states that the SG system and connected components up to the feedwater isolation valves (FWIVs) and main steam isolation valves (MSIVs) are seismic Category I components.

DCA Part 2, Tier 2, Section 3.9.3.1.2, "Load Combinations and Stress Limits," states that the SG supports and SG tube supports are designed as internal structures and are seismic Category I components. DCA Part 2, Tier 2, Section 5.4.1.5, states that the SG tubes are fabricated with alloy 690 TT, in accordance with ASME Code, Section II, Specification SB 163. Reference to the SB-163 specification was included I Rev. 1 of DCA Part 2, Tier 2, Section 5.4.1.5, but not in Rev. 2. NuScale Power (ADAMS Accession No. ML19098A236), dated April 2, 2019, confirmed that reference to the SB-163 specification will be included in Rev. 3. This is **Confirmatory Item 5.4.1-3.** The staff finds alloy 690 TT appropriate because it is listed in ASME Code, Section II, and is, therefore, permitted by 10 CFR 50.55a; additionally, since its first use for SG tubing in the United States in 1989, it has resisted degradation by corrosion mechanisms. DCA Part 2, Tier 2, Table 5.4 2, "Steam Generator Design Data," notes that the tubes have a tube wall outer diameter of 0.625 inch (15.9 mm) and a tube wall thickness of 0.050 inch (1.27mm). The staff notes that, for operating plants with alloy 690 TT SG tubes, the tube wall outer diameter ranges from 15.9 mm (0.625 inch) to 22.2 mm (0.875 inch), and the tube wall thickness ranges from 0.97 mm (0.038 inch) to 1.27 mm (0.050 inch) (NUREG 1841, "U.S. Operating Experience with Thermally Treated Alloy 690 Steam Generator Tubes," issued August 2007 (ADAMS Accession No. ML072330588)). DCA Part 2, Tier 2, Section 5.4.1.6, "Steam Generator Program," states that the greater wall thickness is based on incorporation of a substantial degradation allowance. DCA Part 2, Tier 2, Section 5.4.1.2, states that the lifetime degradation allowance is 0.010 inch (0.25 mm). SER Section 5.4.1.4.5 further discusses this degradation allowance.

DCA Part 2, Tier 2, Section 5.4.1.2, states that the SG supports are welded to the inner surface of the RV. In accordance with ASME Code, Section III, Subparagraph NB-1132.2(d), the welds connecting the SG supports to the RV are considered part of the RV; therefore, the welds are classified as ASME Code Class 1 and conform to ASME Code, Subsection NB. DCA Part 2, Tier 2, Section 5.2.3, addresses the design, fabrication, inspection, testing, and quality assurance of the welds connecting the SG supports to the RV. The staff also notes that DCA Part 2, Tier 2, Table 5.2-6, notes the ISI examination category and examination method for these welds as B-N-2 and VT-2, respectively, which are consistent with ASME Code, Section XI, Table IWB-2500-1. ASME Code, Section XI, Subsection IWB, includes the ISI requirements for ASME Code Class 1 components.

Pending resolution of **Confirmatory Items 5.4.1-1, 5.4.1-2, and 5.4.1-3**, the staff concludes that the materials used for the SGs are acceptable because they will be selected, fabricated, tested, and inspected in accordance with the ASME Code and, therefore, comply with GDC 1 and 30.

DCA Part 2, Tier 2, Table 5.2 4, notes that the SG supports and the SG tube supports are Type 304/304L stainless steel. DCA Part 2, Tier 2, Table 5.2 4, also notes that the integral steam and feed plenum access ports, including the main steam and feedwater supply nozzles that are part of the integral steam plenum and feed plenum access ports, are SA 508, Grade 3, Class 2, low-alloy steel; the integral steam plenum cap is alloy 690; the threaded inserts for the integral steam plenum and feed plenum access ports are Type 304/304L; and the threaded fasteners, nuts, and washers for the integral steam plenum and feed plenum access ports are nickel-based alloy 718. The inside and outside surfaces of the integral steam plenum and feed plenum access ports, including the steam and feed tubesheets, are clad with austenitic stainless steel. The inside surfaces are clad with at least two layers, with the first layer of Type 309L and subsequent layers of Type 308L, and the outside surfaces are clad with at least one layer of Type 309L. DCA Part 2, Tier 2, Table 5.4 2, notes that the steam tubesheet thickness is 4.0 inches (10 cm) without clad and 4.625 inches (11.75 cm) with clad; and the feed tubesheet thickness is 6.0 inches (15 cm) without clad and 6.625 inches (16.83 cm) with clad. The cladding on the outside surface of the steam and feed tubesheets is 0.375 inch (9.53 mm) and the cladding on the inside surface of the steam and feed tubesheets is 0.250 inch (6.35 mm) (total steam and feed tubesheet cladding thickness of 15.9 mm (0.625 inch) (0.375 + 0.250)). DCA Part 2, Tier 2, Table 5.4 3, notes that the integral steam plenum and feed plenum access port covers are Type 304/304L. DCA Part 2, Tier 2, Table 5.4 3, notes that the SG Class 2 piping and SG piping supports are Type 304/304L and the SG piping reducers and elbows are Grade F304/F304L. DCA Part 2, Tier 2, Section 5.4.1.2, notes that the pressure-retaining materials of the thermal relief valves are specified in accordance with the materials identified in DCA Part 2, Tier 2, Table 6.1 3. DCA Part 2, Tier 2, Table 5.4 3, also notes that the flow restrictors, flow restrictor bolts, and flow restrictor mounting plates are Type 304; the flow restrictor mounting plate spacers are alloy 690; and the flow restrictor stud bolts, nuts, and washers are alloy 718. DCA Part 2, Tier 2, Section 5.4.1.5, states that the SG weld filler metals are in accordance with ASME Code, Section II, Part C. DCA Part 2, Tier 2, Tables 5.2 4 and 5.4 3, note that the weld filler material for the nickel-based alloys used in the SGs is alloy 52/52M/152. The staff finds these materials acceptable because they are listed in ASME Code, Section II, and, therefore, are permitted by 10 CFR 50.55a.

The applicant proposed to use precipitation-hardened, nickel-based alloy 718 as the bolting material. This material is listed in ASME Code, Section II, and is, therefore, permitted by 10 CFR 50.55a. SER Section 3.13 further discusses the staff's review of threaded fasteners.

Pending resolution of **Confirmatory Items 5.4.1-1, 5.4.1-2, and 5.4.1-3**, the staff determined that the SG materials meet the requirements of GDC 1, 4, 14, 15, 30, and 31; 10 CFR Part 50, Appendices B and G; and 10 CFR 50.55a, as they relate to selection. The staff based its conclusion on determining that the materials selection, fabrication, testing, and inspection meet the ASME Code requirements.

5.4.1.4.2 Steam Generator Design

The staff reviewed the adequacy of the design and fabrication process proposed for the NuScale SGs to determine whether crevice areas are limited, residual stresses are limited in the tube bends and tubesheet crevice region, corrosion-resistant materials are used, corrosion allowances are specified, and suitable bolting materials are used. As previously discussed in SER Section 5.4.1.4.1, the SG components forming part of the RCPB comply with ASME Code Class 1, and the steam and feedwater piping from the CNTS to the SGs, thermal relief valves, steam plenum access ports, and plenum access covers comply with ASME Code Class 2. Compliance with ASME Code Class 1 and Class 2 design includes consideration of an additional thickness to allow for corrosion. Because the potential for degradation depends partly on the materials and water chemistry, this SER further discusses provisions for limiting degradation.

DCA Part 2, Tier 1, Section 2.1.1, states that the SG supports the RCS by supplying part of the RCPB. DCA Part 2, Tier 2, Section 5.4.1, states that the SG tubes, tube-to-tubesheet welds, and tubesheets provide part of the RCPB. Therefore, the tube-to-tubesheet welds must satisfy the design requirements of the ASME Code Class to comply with GDC 1, 14, 15, 30, and 31 and 10 CFR 50.55a. The SG tubes are secured to the feed and steam plenum tubesheets by expansion fit and are welded to the tubesheets on the secondary side. DCA Part 2, Tier 2, Section 5.4.1.5, states that the SG tubes, including weld materials, conform to the fabrication, construction, and testing requirements of ASME Code, Section III, Subsection NB. This is acceptable to meet the requirements of GDC 1, 14, 15, 30, and 31 and 10 CFR 50.55a, as they relate to the primary-to-secondary pressure boundary formed by the tube-to-tubesheet welds.

Operating experience has shown that crevices around SG tubes in the tubesheet region have caused corrosion. DCA Part 2, Tier 2, Section 5.4.1.2, states that full-length expansion of the tube within the steam and feed plenum tubesheets prevents tube-to-tubesheet crevices. The staff determined that the NuScale design meets the acceptance criteria in NuScale DSRS Section 5.4.2.1, which states that full-depth expansion of the tube through the tubesheet region limits the crevice between the tube and tubesheet.

DCA Part 2, Tier 2, Table 5.4 2, notes that the minimum SG tube transition bend radius is greater than or equal to 6.250 inches (15.88 cm). The applicant did not propose to thermally treat the helical coil SG tubing to reduce residual stresses. Although NuScale DSRS Section 5.4.2.1 states that short-radius U bends should be thermally treated to reduce residual stresses, the staff finds this acceptable because industry guidance in EPRI TR 016743, "Guidelines for Procurement of Alloy 690 Steam Generator Tubing," issued April 1999, specifies thermal treatment for bends with a radius less than 15.88 cm (6.250 inches) for this tube diameter.

The acceptance criteria in NuScale DSRS Section 5.4.2.1 state that the SG design is acceptable with respect to the tube support structures if the support structures are fabricated from corrosion-resistant material and the design provides for flow along the tubes. The NuScale SG design has eight sets of austenitic stainless steel (Type 304/304L) SG supports and SG tube supports that provide vertical, lateral, vibration, and seismic support. The SG tube supports are located between each column of tubes and span the full height of the tube bundle; they are anchored at the top by the upper SG supports that are welded to the RV and the integral steam plenum and are anchored at the bottom by connection to the lower SG supports that are welded to the RV shell below the SG tube bundle. DCA Part 2, Tier 2, Figure 5.4-6, "Steam Generator Tube Supports and Steam Generator Supports," shows that the SG tube supports consist of top, middle, and bottom sections that are welded together. DCA Part 2, Tier 2, Section 5.4.1.2, states that the SG tube supports include stamped tabs that envelope part of the circumference of the tubes. DCA Part 2, Tier 2, Section 5.4.1.2, states that outer and middle spacers are welded into the pockets in the back of the tube supports and that the spacers "allow for the tabs from each adjacent column to nest with each other to create a continuous support path though the columns."

DCA Part 2, Tier 2, Section 5.4.1.2, states that the potential for crevices between the tube and support is limited because the circumferential support is not continuous. DCA Part 2, Tier 2, Section 5.4.1.2, also states that the SG support material limits the generation and buildup of corrosion products and that the geometry of the SG supports minimizes crevices and facilitates flow. The staff determined that the design of the SG supports and SG tube supports meets the guidance in NuScale DSRS Section 5.4.2.1 because it uses corrosion-resistant material and promotes flow along the tube surface. The CVAP evaluates the SG supports and SG tube supports. The staff evaluates the CVAP in SER Section 3.9.2.

The SG system includes the feedwater piping from the CNTS to the feed plenum access port and the main steam piping from the steam plenum access port to the CNTS. DCA Part 2, Tier 2, Table 5.4-3, notes that the SG piping supports are Type 304/304L and the SG piping reducers and elbows are Grade F304/F304L. DCA Part 2, Tier 2, Section 5.4.1.5, states that the SG piping conforms to fabrication, construction, and testing requirements of ASME Code, Section III, Subsection NC. In addition, DCA Part 2, Tier 2, Section 5.4.1.5, states that the materials selected for fabrication of the SG piping conform to the material specifications of ASME Code, Section II, Article NC-2000, and that the materials meet the requirements of ASME Code, Section III. DCA Part 2, Tier 2, Section 5.4.1.5, states that the structural supports for the SG piping, including weld materials, conform to the fabrication, construction, and testing requirements of ASME Code, Section III, Subsection NF.

DCA Part 2, Tier 2, Section 5.4.1.2, describes the inlet flow restrictors used in the NuScale SG design. The flow restriction device at the inlet to each tube is designed to limit unstable flow oscillations. The flow restrictors extend into the tubes and are mounted on a plate in each feed plenum that is attached to the secondary-side face of the tubesheets with stud bolts that are welded to the tubesheets at each mounting location. Spacers located at each mounting plate attachment point hold the flow restrictor mounting plate off the surface of the tubesheets. A flow restrictor bolt holds the flow restrictor subcomponent. DCA Part 2, Tier 2, Section 5.4.1.5, states that the inlet flow restrictors are nonstructural attachments and are designed, fabricated, constructed, tested, and inspected in accordance with ASME Code, Section III, Subsection NC. DCA Part 2, Tier 2, Section 5.4.1.2, also states that the flow restrictors can be removed to support SG tube inspection, cleaning, and plugging. The flow restrictors, flow restrictor bolts, and flow restrictor mounting plates are Type 304; the flow restrictor mounting plate spacers are alloy 690; and the flow restrictor stud bolts, nuts, and washers are alloy 718. The staff reviewed NuScale's proposed flow restrictor design as part of its SG audit, and the audit report (ADAMS Accession No. ML18051A722) summarizes the staff's observations. The CVAP evaluated the inlet flow restrictors, and SER Section 3.9.2 further discusses the staff's evaluation of the CVAP. This section does not review flow stability; however, SER Section 15.9 further discusses the topic, addressed in TR-0516-49417-P, Revision 0, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," issued September 28, 2017.

The NuScale design includes a single spring-operated, balanced-bellows thermal relief valve located on each feedwater line that vents directly into the containment to provide overpressure protection during shutdown conditions. DCA Part 2, Tier 2, Section 5.2.2.2, "Design Evaluation," states that the thermal relief valves are not credited for safety-related overpressure protection during operation. DCA Part 2, Tier 2, Section 5.4.1.2, states that the thermal relief valves are Quality Group B and designed, fabricated, constructed, tested, and inspected as Class 2, in accordance with ASME Code, Section III, and are seismic Category I components. DCA Part 2, Tier 2, Section 5.4.1.2, also notes that the pressure-retaining materials of the thermal relief valves are specified in accordance with the materials identified in DCA Part 2, Tier 2, Table 6.1-3.

DCA Part 2, Tier 2, Section 5.4.1.2, states that the NuScale SG design includes provisions to reduce the potential for tube damage caused by loose parts wear. DCA Part 2, Tier 2, Section 5.4.1.2, also states that the internal and external sides of the tubesheets can be accessed for inspection and removal of foreign objects. As discussed in SER Section 5.4.1.4.1, the design nominal wall thickness includes a lifetime degradation allowance of 0.25 mm (0.010 inch), as further described in SER Section 5.4.1.4.5.

Although this section does not review the topics of water hammer, thermal stratification, and flow-accelerated corrosion (FAC), the staff made the following observations with regard to these topics and the SGs. Although much of the staff guidance in BTP 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators," is not applicable to the NuScale SG design (i.e., not a top feed or preheat design and no auxiliary FWS), as noted in DCA Part 2, Tier 2, Table 1.9-3, "Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)," the applicant followed the tests and test procedures guidance. DCA Part 2, Tier 2, Section 3.6.3.1.4, "Water Hammer/Steam Hammer," states that the FWS and SG contain design features and operating procedures that minimize the potential for and effect of water hammer. DCA Part 2, Tier 2, Section 3.6.3.1.4, further states that these features are designed to minimize or eliminate the potential for water hammer in the SG FWS. DCA Part 2, Tier 2, Section 3.12.5.8.3, "Feedwater Line Stratification," states that the thermal stratification load is minimized because the SG feedwater nozzles on the feedwater inlet plenums and the adjacent feedwater lines are either vertical or angled downward from the horizontal. DCA Part 2, Tier 2, Section 10.3.5.1.1, "Chemistry Control Objectives and Basis," states that the "secondary system components and piping exposed to wet steam, flashing liquid flow, or turbulent single phase flow where significant loss of material could occur use corrosion, erosion, and flow accelerated corrosion (FAC) resistant materials." SER Chapters 3 and 10 further discuss water hammer, thermal stratification, and FAC.

The applicant described how the overall SG design in DCA Part 2, Tier 2, Section 5.4.1.3, addresses FIV of the tube bundle. The staff evaluated this as part of its FIV review in SER Section 3.9.2.

Pending resolution of **Confirmatory Items 5.4.1-1, 5.4.1-2, and 5.4.1-3**, the staff determined that the SG design meets the acceptance criteria in NuScale DSRS Section 5.4.2.1, as they relate to limiting the potential for degradation of the tubes and other secondary-side components. These criteria, in conjunction with the acceptance criteria for interacting reviews and appropriately performed ISIs, provide assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture will be extremely low, (2) the design conditions of the RCPB are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure, consistent with the requirements of GDC 14, 15, and 31.

5.4.1.4.3 Fabrication and Processing of Ferritic Materials

To comply with GDC 14, 15, and 31, the fracture toughness of the ferritic materials forming the primary and secondary pressure boundaries of the SGs must resist rapidly propagating failure and ensure that the design conditions will not be exceeded during operation. The pressure-retaining ferritic materials selected for use in SGs are acceptable with respect to fracture toughness if they (1) comply with Appendix G to 10 CFR Part 50 and 10 CFR 50.55a(c), (d), and (e), and (2) follow the provisions of Appendix G to ASME Code, Section III. For Class 1 and Class 2 SG components, the regulations cited above require the use of ASME Code, Section III. ASME Code, Section III, Subarticle NB-2300, Subarticle NC-2300, and Appendix G, address fracture toughness requirements for Class 1 and Class 2 components. DCA Part 2, Tier 2, Section 5.2.3.3.1, "Fracture Toughness," states that the fracture toughness properties of the RCPB components comply with the requirements of Appendix G of 10 CFR Part 50 and ASME Code, Section III, Subarticle NB-2300. Therefore, the staff has determined that the SG ferritic material complies with the requirements related to fracture toughness. SER Section 5.2.3 further discusses the staff's review of the fracture toughness of the RCPB materials.

To comply with GDC 1 and 30, the welding of the ferritic steel for the primary and secondary pressure boundary of the SGs must meet the requirements of 10 CFR 50.55a(c), (d), and (e). Ferritic steel pressure-boundary welding must also meet the requirements of ASME Code, Section III, Appendix D, Subsubarticle D-1210, and adhere to the guidance in RGs 1.43, 1.50, and 1.71. DCA Part 2, Tier 2, Section 5.2.3.3.2, "Welding Control—Ferritic Materials," describes how the NuScale design meets these welding requirements of ferritic materials used for RCPB components. DCA Part 2, Tier 2, Section 5.4.1.5, states that DCA Part 2, Tier 2, Section 5.2.3, describes welding controls related to ASME Code Class 1. Therefore, the staff has determined that the design complies with the requirements of 10 CFR Part 50 related to welding SG ferritic materials. SER Section 5.2.3 further discusses the staff's review of welding RCPB materials.

5.4.1.4.4 Fabrication and Processing of Austenitic Stainless Steel

To comply with GDC 1, 14, 15, 30, and 31, the use of austenitic stainless steel in SG pressure-boundary applications must include limiting the susceptibility to SCC and performing welding according to quality standards. The requirements of GDC 4 and 10 CFR Part 50, Appendix B, Criterion XIII, are met through compliance with the applicable provisions of the ASME Code and with the guidance in RGs 1.31, 1.34, 1.36, 1.44, and 1.71.

DCA Part 2, Tier 2, Section 5.2.3.4, describes how the NuScale design meets these requirements for use of austenitic stainless steel in pressure-boundary applications. DCA Part 2, Tier 2, Table 1.9-2, "Conformance with Regulatory Guides," indicates that, because the NuScale design does not use nonmetallic thermal insulation on RCPB or CNV components, RG 1.36 does not apply to DCA Part 2, Tier 2, Chapter 5, Sections 5.2 and 5.4. Therefore, the staff has determined that the design complies with the requirements of 10 CFR Part 50 related to fabricating and processing SG austenitic stainless steels. SER Section 5.2.3 further discusses the staff's review of fabrication and processing requirements for austenitic stainless steel used in other RCPB applications.

5.4.1.4.5 Compatibility of Materials with the Primary and Secondary Coolant and Cleanliness Control

The SG components that form the RCPB and the supporting structural components must be compatible with the reactor coolant and secondary coolant to meet the requirements of GDC 4.

DCA Part 2, Tier 2, Section 5.2.3.2.1, describes the control of the primary water chemistry, and DCA Part 2, Tier 2, Section 10.3.5.1, "Chemistry Control Program," describes the secondary water quality control program. Control of the primary water chemistry is based on the EPRI "Pressurized Water Reactor Primary Water Chemistry Guidelines," while control of the secondary water chemistry is based on the EPRI "Pressurized Water Reactor Secondary Water Chemistry Guidelines," and NEI 97 06, Revision 3, "Steam Generator Program Guidelines," issued January 2011. In addition, the proposed secondary water chemistry program conforms to the latest revision of the Standard Technical Specifications (STS). The staff has determined that the EPRI guidelines are acceptable for primary and secondary water chemistry control for the NuScale design. SER Sections 5.2.3.2, 9.3.4, and 10.4.6 further discuss the staff's review of primary and secondary water chemistry control.

As noted in SER Section 5.4.1.4.2, the design nominal wall thickness includes a lifetime degradation allowance of 0.25 mm (0.010 inch). DCA Part 2, Tier 2, Section 5.4.1.3.1, “Allowable Tube Wall Thinning under Accident Conditions,” states that the degradation allowance accounts for general corrosion, erosion, and wear. DCA Part 2, Tier 2, Section 5.4.1.3.1, also states that the “reduced wall thickness provides a safety margin (sufficient wall thickness) for a maximum stress less than the allowable stress limit, as defined by the ASME Code Level D Service condition” and that the 0.25 mm (0.010 inch) degradation allowance provides a reasonable bounding condition. Based on the description in DCA Part 2, Section 5.4.1, the staff determined that the steam generators are designed with the intent to prevent all forms of tube degradation. They are also designed for the tubes to maintain structural margins even when thinned by degradation. Because the design includes margin for degradation, and because operating experience has revealed no corrosion-related degradation of Alloy 690 tubes with the proposed primary and secondary water chemistries, the staff determined the proposed degradation allowance is acceptable.

DCA Part 2, Tier 2, Section 5.4.1.5, states that DCA Part 2, Tier 2, Section 5.2.3.4.2, “Cleaning and Contamination Protection Procedures,” contains the cleaning and cleanliness controls for the SGs. DCA Part 2, Tier 2, Table 1.9-2 notes that DCA Part 2, Tier 2, Section 5.4.1 is in conformance with ASME NQA 1-2008 and the NQA 1a-2009 addenda, as endorsed in RG 1.28, Revision 4, issued June 2010. Because RG 1.28, Revision 4, contains the staff’s guidance on cleaning and cleanliness controls, the design meets the requirements of 10 CFR Part 50, Appendix B, Criterion XIII, with respect to the cleaning and cleanliness controls for the SG design.

The staff determined that the materials selected for the SGs are compatible with the primary and secondary coolant, and the primary and secondary coolant chemistry controls will limit the susceptibility of the SGs to corrosion. In addition, the proposed secondary water chemistry program conforms to the latest revision of the STS. Therefore, the staff determined that the GDC 4 requirements are met.

5.4.1.4.6 Provisions for Accessing the Secondary Side (Primary Side for the NuScale Design) of the Steam Generator

The design for accessibility is considered acceptable if it provides adequate secondary-side (primary side for the NuScale design) access for tools to inspect and remove corrosion products and foreign objects that may affect tube integrity. The staff reviewed the information in DCA Part 2, Tier 2, Section 5.4.1.2. The NuScale design includes four steam plenum inspection ports that provide openings for access to the top of the tube bundle, steam tubesheets, steam plenums, upper SG supports, and upper SG tube supports. The design also includes four feed plenum inspection ports that provide openings for access to the bottom of the tube bundle, feed tubesheets, feed plenums, lower SG supports, and lower SG tube supports, and flow restrictor assemblies. DCA Part 2, Tier 2, Section 5.4.1.2, states that the internal and external sides of the tubesheets can be accessed for inspection and removal of foreign objects. The staff also reviewed the accessibility of the primary side for the NuScale design as part of its SG audit; the audit report (ADAMS Accession No. ML18051A722) summarizes the staff’s observations.

The staff determined that this level of primary-side access is acceptable because tools may be inserted to inspect and remove corrosion products, contaminants that may lead to corrosion, and foreign objects (including loose parts) that may affect tube integrity. Therefore, the requirements of GDC 14 and 15 are met with respect to secondary side access.

5.4.1.5 Steam Generator-Related Inspections, Tests, Analyses, and Acceptance Criteria

DCA Part 2, Tier 2, Section 14.3.5, "Tier 1 Chapter 2, Unit-Specific Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria," evaluates ITAAC associated with this section.

5.4.1.6 Combined License Information Items

There are no COL information items for this review topic.

5.4.1.7 Conclusion

Pending resolution of **Confirmatory Items 5.4.1-1, 5.4.1-2, and 5.4.1-3**, on the basis of its review of DCA Part 2, Tier 2, Section 5.4.1, the staff concludes that the NuScale SG materials satisfy the acceptance criteria for materials selection, design, fabrication, compatibility with the service environments, and secondary-side accessibility. The staff further concludes that these materials as specified are acceptable and meet the requirements of GDC 1, 4, 14, 15, 30, and 31 and 10 CFR Part 50, Appendices B and G.

5.4.2 Steam Generator Program

5.4.2.1 Introduction

The SG program is intended to ensure that SG tube structural and leakage integrity are maintained during operation and postulated accident conditions.

5.4.2.2 Summary of Application

By application dated December 31, 2016 (ADAMS Accession No. ML17013A229), as supplemented by letters dated August 3, 2017 (ADAMS Accession Nos. ML17215A977 and ML17215A978); January 31, 2018 (ADAMS Accession No. ML18032A391); February 12, 2018 (ADAMS Accession No. ML18043B174); April 2, 2018 (ADAMS Accession No. ML18092B091); April 3, 2018 (ADAMS Accession No. ML18093B542); August 2, 2018 (ADAMS Accession No. ML18215A261); October 17, 2018 (ADAMS Accession No. ML18290B059); October 22, 2018 (ADAMS Accession No. ML18295A787); and January 7, 2019 (ADAMS Accession No. ML19007A091), the applicant submitted information related to the NuScale SMR SG program.

The staff confirmed that Revision 2 of the DCD (ADAMS Accession No. ML18311A006) incorporates the changes to the DCD made before October 22, 2018. The staff will confirm that the final revision of the DCD incorporates acceptable changes to the DCD submitted in the applicant's letters dated October 22, 2018, January 7, 2019, and April 2, 2019. The staff is tracking the changes as **Confirmatory Items 5.4.1-1, 5.4.1-2, and 5.4.1-3**.

DCA Part 2, Tier 1: There are no Tier 1 entries for this review topic.

DCA Part 2, Tier 2: The applicant described the proposed SG program in DCA Part 2, Tier 2, Sections 5.4.1 and 5.4.1.6. DCA Part 2, Tier 2, Section 5.4.1.6, states that the NuScale SG program will provide monitoring and management of tube degradation and degradation precursors and will permit preventative and corrective actions to be taken in a timely manner. The program is based on NEI 97-06 and the STS. The major program elements are assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.

ITAAC: There are no ITAAC for this review topic.

Technical Specifications: The applicant provided the NuScale TS and bases related to the SG program in DCA Part 2, Tier 2, Chapter 16; specifically, in TS Sections 3.4.5, 3.4.9, 5.5.4, and 5.6.5 and Bases Sections B 3.4.5 and B 3.4.9. The purpose of these TS and bases is to maintain SG tube structural and leakage integrity.

Technical Reports: There are no TRs for this review topic.

5.4.2.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 32 requires, in part, that the designs of all components that are part of the RCPB permit periodic inspection and testing of critical areas and features to assess their structural and leak tight integrity.

- 10 CFR 50.55a(g) requires that ISI programs meet the applicable inspection requirements in ASME Code, Section XI.

- 10 CFR 50.36 applies to the SG program in the TS.

- 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

- Appendix B to 10 CFR Part 50 applies to quality assurance in the implementation of the SG program.

Acceptance criteria adequate to meet the above requirements include the following:

- NEI 97-06

- NUREG-1430, NUREG-1431, and NUREG-1432, “Standard Technical Specifications”

- NUREG-2194, “Standard Technical Specifications, Westinghouse Advanced Passive 1000 (AP1000) Plants,” issued April 2016

- TS Task Force (TSTF) 510, “Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection”

- RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” as it relates to determining the plugging criteria for degraded SG tubes

- BTP 5-1, “Monitoring of Secondary Side Water Chemistry in PWR Steam Generators,” as it relates to monitoring secondary-side water chemistry

5.4.2.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Sections 5.4.1 and 5.4.1.6, in accordance with NuScale DSRS Section 5.4.2.2, “Steam Generator Program,” to ensure that the SG, as part of the RCPB, is designed to permit periodic inspection and testing of the tubes and other critical areas and that it includes features to assess the structural and leakage integrity of the tubes, as required by GDC 32. The staff reviewed TS Sections 3.4.5, 3.4.9, 5.5.4, 5.6.5, and Bases Sections B 3.4.5 and B 3.4.9, which are related to the SG program. The staff also reviewed supplemental information that the applicant provided in the letters identified in SER Section 5.4.2.2.

DCA Part 2, Tier 2, Section 5.4.1.1, states that the design of the primary and secondary sides of the SGs permit implementation of an SG program that provides reasonable assurance the structural and leakage integrity of the SG tubes are maintained. DCA Part 2, Tier 2, Section 5.4.1.2, states that, when the NPM is disassembled for refueling, periodic inspection and testing of critical areas and features to assess their structural and pressure boundary integrity can be performed as a result of the SG design. The internal surface of the SG tubes can be inspected over their entire length with NDE methods and techniques capable of detecting the types of degradation that may occur over the life of the tubes. NuScale acknowledged, in its response to RAI 9273, Question 05.04.02.01-1e (ADAMS Accession No. ML18093B542), that a specific plant location may need to qualify an eddy current system for inspection of the NuScale SGs. The staff reviewed the feasibility of SG tube inspection for the NuScale design as part of its SG audit; the audit report (ADAMS Accession No. ML18051A722) summarizes the staff’s observations. DCA Part 2, Tier 2, Section 5.4.1.6.1, “Degradation Assessment,” notes that the applicant’s degradation assessment identified wear as the most likely degradation mechanism and noted under-deposit pitting and intergranular attack as potential secondary-side degradation mechanisms. The staff reviewed the applicant’s degradation assessment as part of its SG audit, and the audit report (ADAMS Accession No. ML18051A722) summarizes the staff’s observations. The internal and external sides of the tubesheets can be accessed for inspection and removal of foreign objects. The staff finds the NuScale SG design meets the acceptance criteria in NuScale DSRS Section 5.4.2.2 because all of the SG tubes can be accessed for full-length inspection, the SG can be examined using NDE techniques capable of detecting the types of degradation that may occur over the life of the SG tubes, and the primary and secondary sides of the SG can be accessed for inspection and foreign object removal.

DCA Part 2, Tier 2, Section 5.4.1.4, “Tests and Inspections,” states that the SG program is based on NEI 97-06 and is documented in TS Section 5.5.4. The SG program also follows applicable EPRI guidance; implements applicable portions of ASME Code, Section XI; and specifically addresses 10 CFR 50.55a(b)(2)(iii). DCA Part 2, Tier 2, Section 5.4.1.6, states that 10 CFR Part 50, Appendix B, applies to implementing the SG program. The staff finds this acceptable because these bases for the SG program conform to NuScale DSRS Section 5.4.2.2.

As discussed in SER Section 5.4.1.4.5, the water chemistry programs are controlled in accordance with industry guidelines that conform to the acceptance criteria in NuScale DSRS

Section 5.4.2.2. SER Sections 5.2.3.2, 9.3.4, and 10.3.5 further discuss the staff's review of the control of the primary and secondary water chemistry.

DCA Part 2, Tier 2, Section 5.4.1.6, states that the plant TS describe the SG program, which is part of the overall ISI program. In accordance with NuScale DSRS Section 5.4.2.2, the staff reviewed the technical requirements of the SG program in NuScale's proposed GTS. Specifically, the staff reviewed the proposed criteria for assessing the as-found condition of a SG tube following a tube inspection, SG tube integrity performance criteria (tube structural integrity, accident-induced primary-to-secondary leakage limits, and operational primary-to-secondary leakage limits), SG tube plugging criteria, criteria for selecting SG tube inspection intervals and the tubes to be inspected, and provisions for monitoring operational primary-to-secondary leakage. The staff used NUREG-1431, Revision 4, issued April 2012, and TSTF-510, Revision 2, dated March 1, 2011, as the basis for its review of NuScale's GTS associated with the SG program. Revision 4 of NUREG-1431 is for Westinghouse plants, which is acceptable because, with respect to the SG program, all four versions of the STS for PWRs are the same (Westinghouse plants, Combustion Engineering plants, Babcock and Wilcox plants, and AP1000 plants).

The staff observed differences between NuScale's GTS and the STS because of the design of the NPM and the SGs. Some of the differences are found throughout the GTS, and the staff reviews them further in SER Chapter 16. These differences include the following:

- operating mode definitions
- completion times for required actions
- frequency of SRs
- definition of a "unit" as an NPM

Some nonstandard TS wording is related to the NuScale design. NuScale's proposed GTS use SG tube "failure" instead of SG tube "burst" that is used in the STS. The staff finds this acceptable because burst is not the expected failure mechanism as the SG tubes in the NuScale design have the higher pressure on the outside. The staff finds it acceptable that NuScale's proposed GTS Subsection 5.5.4.d.1 uses "initial startup and SG replacement" instead of "SG installation" as stated in corresponding STS Subsection 5.5.9.d.1 because the NuScale SGs are fabricated integral to the upper module. NuScale's proposed GTS do not discuss wastage and denting of SG tubes. The staff finds this acceptable because these degradation mechanisms have not been observed in alloy 690 TT SG tubes in operating plants and are not anticipated in the NuScale SGs.

Although NUREG-1431 and TSTF-510 allow the use of "tube repair criteria" and "plugged [or repaired]," NuScale's proposed GTS use "tube plugging criteria" and "plugged," respectively. TSTF-510 differs from the STS because it changed the standard terminology from "repair criteria" to "plugging criteria," while still including repair as a bracketed option. For example, TS Subsection 5.5.9.c in TSTF-510 is "Provisions for SG tube plugging [or repair] criteria." NuScale proposed using the term "plugging criteria," without including a bracketed alternative plugging or repair option or the embedded notes about alternatives to the proposed plugging criterion. The staff finds this acceptable because "plugging criteria" is consistent with the STS, and plant TS are not required to propose alternatives, as provided under STS Subsection 5.5.9.c.

The TS include an accident-induced leakage (AIL) performance criterion to ensure the primary to-secondary leakage caused by a design-basis accident is within the accident analysis assumptions. Together with the operational leakage performance criterion and the SG program, the AIL performance criterion ensures the dose from SG tube leakage remains below the onsite and offsite dose limits. To be consistent with the STS, the AIL performance criterion must be less than the leakage rate assumed in the accident analyses and must be no more than 1 gpm (4 liters per minute (l/m)). As noted in NEI 97 06, the primary-to-secondary leakage rate assumed in accident analyses can vary. NuScale described its accident analyses and the primary-to-secondary leakage values assumed in its accident analyses in DCA Part 2, Tier 2, Chapter 15, and in its August 2, 2018, letter (ADAMS Accession No. ML18215A261), respectively. NuScale selected 150 gallons per day (0.568 m3/day) for the AIL performance criterion, which is equivalent to its operational leakage performance criterion and, in most cases, the leakage values assumed in its accident analyses. For currently operating plants, the AIL performance criterion is typically a higher value than the operational leakage performance criterion based on analyses of cracks, showing that the leakage rate at accident conditions is higher than at operating conditions. However, NuScale stated that an AIL performance criterion greater than the operational leakage criterion is not required for its design, based on structural integrity performance evaluations. The staff finds the specified AIL performance criterion acceptable because the plant will be required to maintain AIL at a value that will not exceed the leakage rate assumed in the accident analyses and will not exceed 1 gpm (4 l/m). These requirements are consistent with the corresponding AIL performance criteria in the STS.

The STS allows selection of the SG tube inspection interval based on SG tube material. The inspection requirements in the STS are based on well-established behavior of the predominant tube materials in SGs with longstanding designs. Although the industry is prepared to observe new degradation modes and continues to refine the examination techniques, the modes of degradation and examination techniques have been stable, along with the ability to detect and manage service degradation and flaws from other sources (such as manufacturing). In the STS, the longest inspection intervals apply to SGs with alloy 690 TT tubing. NuScale proposed the longest SG tube inspection interval in its GTS because the SG tube material is alloy 690 TT, the most likely degradation mechanism is wear, water chemistry is controlled in accordance with industry guidelines, and design elements may reduce degradation and corrosion product accumulation. NuScale noted that existing examination techniques may have to be modified for its SGs for PSI and ISI of the tubes. However, NuScale also noted that there is a high certainty that its techniques will be capable of detecting and characterizing wear, the most likely degradation mechanism, because of the known design geometry (wear most likely to occur at tube-to-tube support locations). NuScale stated that the longest SG tube inspection interval from the STS is a maximum inspection interval and, depending upon SG performance, which is monitored by the SG program, the inspection interval may need to be adjusted. The staff finds the SG tube inspection interval in NuScale's proposed GTS acceptable, based on operating experience for SGs with alloy 690 TT tubes, the use of industry best practices in water chemistry, and the overall requirements of the SG program. Operating experience for SGs with alloy 690 TT tubes has revealed no instances of corrosion-related degradation of the tubes, and this is expected to be the case for the NuScale SGs based on the materials, temperature, and water chemistry. Therefore, the staff agrees that wear is the most likely form of degradation to expect. Nonetheless, a licensee's SG program will be required to have examination techniques capable of detecting flaws of any type that may satisfy the SG tube plugging criterion. Qualification of NDE procedures and personnel and demonstration of the capabilities of the NDE technique are the responsibility of the licensee and will have to be completed before PSI of the tubes. The staff also agrees that the SGs will be monitored through the SG program and that the length of the inspection interval may need to be adjusted, depending on the performance of the SGs through implementation of the SG program. For example, although the

NDE technique may have good detection and characterization capabilities, a lack of operating experience may limit the ability to apply propagation rate information to the operational assessment.

The staff determined that NuScale's proposed GTS are appropriately consistent with the STS and TSTF-510, considering the differences in the NuScale design. SER Section 16.1 further discusses the staff's review of NuScale's proposed TS and bases.

DCA Part 2 and the TS include an SG tube plugging criterion based on maintaining the structural and leakage integrity of the tubes. The SG tube plugging criterion establishes a minimum acceptable SG tube wall thickness that accounts for flaw growth and uncertainty in measuring the size of a flaw. SG tubes with flaws exceeding the plugging criterion will be removed from service by plugging. Tubes that are plugged can also be stabilized, if necessary, to prevent effects on other tubes. DCA Part 2, Tier 2, Section 5.4.1.2, states that individual SG tubes may be plugged and, if necessary, stabilized to prevent adverse interaction with nonplugged tubes. DCA Part 2, Tier 2, Section 5.4.1.6.1, states that the NuScale SG tube plugging criterion is 40-percent through-wall defect and is consistent with the existing PWR SG fleet. DCA Part 2, Tier 2, Section 5.4.1.4, states that "tubes with flaws that exceed 40 percent of the nominal tube wall thickness shall be plugged. Tubes with flaws that could potentially compromise tube integrity prior to the performance of the first inservice inspection, and tubes with indications that could affect future inspectability of the tube, shall also be plugged." DCA Part 2, Tier 2, Section 5.4.1.2, notes that each of the two SGs include a 10-percent tube plugging margin with respect to heat transfer.

The STS use a value for the SG tube plugging criterion that is meant to be bounding, based on an analysis of the loads and degradation applicable to the two basic types of SG designs for operating reactors—recirculating (U-tube) and once-through. The SG tube plugging criterion addresses operating loads, accident loads, uncertainty in NDE, and flaw growth between inspections. The limiting loads are under normal operating conditions, so the SG tube plugging criterion is intended to maintain structural integrity with a safety factor of 3 times normal operating pressure differential (NOPD).

Because the loads and degradation mechanisms applicable to the currently operating SGs are not directly applicable to the NuScale design, a design-specific determination was necessary. A significant difference is that, under normal operating conditions, the tubes in the NuScale design have the higher primary pressure on the outside rather than on the inside. Therefore, collapse is the primary mode of failure rather than burst.

NuScale submitted the 40-percent plugging criterion as a bracketed value in Subsection 5.5.4.c of its proposed GTS. This means that a COL applicant can either use the 40-percent plugging criterion by demonstrating it is applicable to its plant or propose and justify an alternative SG tube plugging criterion for its plant.

NuScale's determination of the generic SG tube plugging criterion value followed RG 1.121 in accordance with the guidance in NuScale DSRS Section 5.4.2.1. RG 1.121 describes an acceptable methodology for determining the SG tube plugging criterion. DCA Part 2, Tier 2, Table 1.9 2 notes conformance with RG 1.121, and DCA Part 2, Tier 2, Section 5.4.1.1, states that the SG program is based on NEI 97-06 and RG 1.121 and is described in TS Section 5.5.4. NuScale's determination used a finite element analysis (FEA) considering the applicable loads, degradation mechanisms, and NDE uncertainty. NuScale used industry guidance to determine the uncertainty from eddy current testing and flaw sizing uncertainty. NuScale's FEA models considered ovality (only for wear flaws), cracks (through-wall axial and circumferential), and wear (flat and tapered with 50 percent and 60 percent, respectively, through-wall depths). NuScale concluded that 50 percent through-wall was adequate to maintain structural integrity and specified the 40 percent SG tube plugging criterion to provide margin for flaw growth and NDE uncertainty. This was based on wear flaws equivalent to the length of a tube support tab (1.27 cm (0.500 inch)).

The staff evaluated NuScale's proposed SG tube plugging criterion using an independent FEA modeling approach informed by results of collapse tests on Alloy 600. The staff's SG audit provided design details, assumptions, and calculation inputs for the staff's evaluation of the proposed plugging criterion. An audit report summarizes the staff's observations (ADAMS Accession No. ML18051A722.) The independent FEA models considered flat-bottomed wear and axial cracks, including the effect of ovality. The independent FEA models also considered 1.52 cm (0.600 inch) wear flaws to account for some relative motion and some variations on the tube-to-tube support tab wear configuration. The independent FEA models did not consider wall thickness variations and NDE uncertainty. The independent analyses demonstrated that there is margin on meeting the 3 times NOPD criterion for the 1.52 cm (0.600 inch), 40 percent deep wear flaw. The independent analyses also demonstrated that, to maintain the safety factor of 3 for NOPD, the maximum allowable length of a 40 percent deep wear flaw would be approximately 2 cm (0.7 inch). The independent analyses further demonstrated that, to maintain the safety factor of 3 for NOPD for a wear flaw of any length, the maximum allowable depth of the flaw would be approximately 24 percent. This indicates the plugging criterion for tube collapse depends on the length of the flaw.

In summary, the staff's independent evaluation indicates that an SG tube plugging criterion of 40-percent through-wall with an axial length close to the tab width is a reasonable preliminary value for the type of degradation considered most likely (tube-to-tube support tab wear). Enclosing the SG tube plugging criterion value in brackets in NuScale's proposed GTS requires a COL applicant to confirm that this value is appropriate or propose and justify a different value.

DCA Part 2, Tier 2, Section 5.4.1.1, states that the SG system components are designed such that ASME Code, Section XI, ISI requirements can be performed, including the ASME Code, Section III, PSI requirements. DCA Part 2, Tier 2, Section 5.4.1.4, states that the PSI of the SG tubes will be performed after tube installation and shop or field primary-side hydrostatic testing and before initial power operation. DCA Part 2, Tier 2, Section 5.4.1.4, also states that a full-length examination of 100 percent of the tubing in each SG will be performed with a volumetric technique capable of detecting the types of preservice flaws that may be present in the tubes and shall permit comparisons to the results of the ISIs expected to be performed to satisfy plant TS for SG tube inspections. DCA Part 2, Tier 2, Section 5.4.1.4, also states that the length of the tube extends from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet; the tube-to-tubesheet welds are not part of the tube. The staff finds that the PSI of the NuScale SG design meets the acceptance criteria in NuScale DSRS Section 5.4.2.2 because all SG tubes will be inspected along their full length after fabrication and before being placed into service with NDE techniques capable of detecting the degradation that may occur over the life of the SG, which can be compared to subsequent ISIs.

The NuScale SGs are designed to be accessible for inspection. All tubes can be inspected from the inside using NDE techniques capable of detecting the types of degradation that may occur over the life of the tubes. The licensee is responsible for qualification of NDE procedures and personnel and demonstration of the capabilities of the NDE technique. The SGs are also designed with access to the shell side for inspection, cleaning, and evaluation of conditions such as loose parts. On this basis, the staff determined that the design of the NuScale SGs is acceptable as it relates to providing access to allow ISIs.

5.4.2.5 Combined License Information Items

Table 5.4.2-1 lists COL information numbers and descriptions related to the SG program and the SG PSI and ISI operational programs, from DCA Part 2, Tier 2, Table 1.8-2.

Table 5.4.2-1 NuScale SG Related COL Information Items in DCA Part 2, Tier 2

Item No.	Description	DCA Part 2 Tier 2 Section
COL Item 5.4-1	A COL applicant that references the NuScale Power Plant design certification will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on the latest revision of Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and applicable Electric Power Research Institute steam generator guidelines at the time of the COL application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.	5.4.1.6
COL Item 13.4-1	COL applicant that references the NuScale Power Plant design certification will provide site specific information, including implementation schedule, for the following operational programs: • Inservice inspection programs (refer to Section 5.2, Section 5.4, and Section 6.6) • Inservice testing programs (refer to Section 3.9 and Section 5.2) • Environmental qualification program (refer to Section 3.11) • Pre-service inspection program (refer to Section 5.2 and Section 5.4) • Reactor vessel material surveillance program (refer to Section 5.3)	13.4

Item No.	Description	DCA Part 2 Tier 2 Section
	• Pre-service testing program (refer to Section 3.9.6, Section 5.2, and Section 6.6) • Containment leakage rate testing program (refer to Section 6.2) • Fire protection program (refer to Section 9.5) • Process and effluent monitoring and sampling program (refer to Section 11.5) • Radiation protection program (refer to Section 12.5) • Non-licensed plant staff training program (refer to Section 13.2) • Reactor operator training program (refer to Section 13.2) • Reactor operator requalification program (refer to Section 13.2) • Emergency planning (refer to Section 13.3) • Process control program (PCP) (refer to Section 11.4) • Security (refer to Section 13.6) • Quality assurance program (refer to Section 17.5) • Maintenance rule (refer to Section 17.6) • Motor-operated valve testing (refer to Section 3.9) • Initial test program (refer to Section 14.2)	

COL Information Item 5.4-1 requires a COL applicant referencing the NuScale DC to develop and implement an SG program based on the latest revision of NEI 97-06. COL Information Item 5.4-1 also includes the specific elements that a COL applicant should include in the SG program. COL Information Item 13.4-1 requires a COL applicant to prepare SG PSI and ISI operational programs, including implementation schedules. The staff concludes that the COL information items related to the SG program are acceptable because they meet the guidance in NuScale DSRS Section 5.4.2.2, the requirements in 10 CFR 50.55a, and the guidance in NEI 97-06 for individual licensees to develop an SG program and SG PSI and ISI operational programs.

5.4.2.6 Conclusion

Pending resolution of **Confirmatory Item 5.4.1-1**, on the basis of its review of DCA Part 2, Tier 2, Sections 5.4.1 and 5.4.1.6, the staff concludes that the NuScale SG program, as described in the DCD, satisfies the acceptance criteria for accessibility for periodic inspection and testing of critical areas for structural and leakage integrity. The staff also concludes that the proposed TS are appropriately consistent with the STS, considering the differences in the NuScale design. The tube-plugging criterion was determined using the methodology specified in RG 1.121 and is a reasonable preliminary value for the type of degradation considered most likely (tube-to-tube support tab wear). Therefore, the staff concludes that the SG program is acceptable and meets the requirements of GDC 32; 10 CFR 50.55a; 10 CFR 50.36; 10 CFR 50.65; and Criteria IX, XI, and XVI in Appendix B to 10 CFR Part 50.

5.4.3 Reactor Coolant System Piping and Check Valves

5.4.3.1 Introduction

RCS piping and connected piping that penetrate the RCS form the RCPB and include the PZR spray supply, RCS injection, RCS discharge, and RV high-point degasification piping.

Although the SRP does not contain a section related to RCS piping and check valves, the review scope of SRP Section 5.4, "Reactor Coolant System Component and Subsystem Design," includes this topic. Detailed information about the RCS piping and check valves, including the staff's evaluation and conclusion on RCS piping and check valves design features and performance requirements, appears in SER Sections 3.9.1, 3.9.2, 3.9.3, 3.10, 3.12, 5.2.3, 5.2.4, 5.2.5, 5.4.3, 6.1, 6.3, 6.6, and 10.3.6.

5.4.3.2 Technical Evaluation and Conclusion

The above SER sections discuss the staff's technical evaluation and conclusion on RCS piping and check valves design features as well as performance requirements.

5.4.4 Decay Heat Removal System

5.4.4.1 Introduction

In traditional light-water reactor designs, residual heat removal (RHR) systems are used to cool the RCS following a shutdown. In the NuScale design, safety-related RHR following accidents is accomplished using passive systems, while the cooldown following a routine shutdown is performed by using normal feedwater and secondary side systems followed by the containment flood and drain system. The primary focus of review for this section is the safety-related means of removing heat, the DHRS. This system performs a similar function to the auxiliary FWS in a traditional PWR but uses natural circulation to drive fluid through a heat exchanger submerged in the reactor building pool (which acts as the ultimate heat sink) to remove heat rather than pumping feedwater into the SGs. This function is important for non-LOCA events when normal secondary side cooling is unavailable to bring the plant to safe-shutdown conditions.

5.4.4.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, describes the NPM and associated systems, one of which is the DHRS. DCA Part 2, Tier 1, Table 2.1-1, describes the power module piping systems, which include the DHRS lines from the SGs to the condenser and the lines leading from the condenser back into containment. DCA Part 2, Tier 1, Table 2.1-2, provides an inventory of the mechanical equipment; for the DHRS, this is limited to the four actuation valves and the two condensers. DCA Part 2, Tier 1, Table 2.1-3, similarly inventories the electrical equipment, which includes the DHRS actuation valves; these valves are also considered for equipment qualification in DCA Part 2, Tier 1, Table 2.8-1, which also lists the instrumentation required to monitor the DHRS. DCA Part 2, Tier 1, Table 2.1-4 and Table 2.8-2, contain the ITAAC for the DHRS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 5.4.3, "Decay Heat Removal System," describes the functions of the DHRS. The DHRS is safety related and designed to remove heat from the power module such that the RCS temperature reaches 420 degrees F (215.6 degrees C) within 36 hours after an initiating event, using only one train of DHRS. The DHRS has two trains, each comprising piping from an SG (shared with the MSS) that branches off upstream of the MSIV to two DHRS actuation valves in parallel. These valves lead to an orifice before the DHRS condenser, which is submerged in the reactor pool. Downstream of the condenser, the DHRS piping rejoins with the feedwater line downstream of the main feedwater isolation valve. DCA Part 2, Tier 2, Chapter 5, Figure 5.4-10, "Decay Heat Removal System Simplified Diagram," displays this layout in a simplified diagram.

The DHRS is normally isolated and actuates on a collection of signals specified in DCA Part 2 Tier 2, Chapter 7, Table 7.1-4, "Engineered Safety Feature Actuation System Functions." This signal, in addition to actuating the DHRS valves by removing power from the trip solenoids, also changes the state of the following valves:

- MSIVs
- main steam isolation bypass valves
- secondary MSIVs
- secondary main steam isolation bypass valves
- FWIVs
- feedwater regulation valves

This configuration results in a closed loop associated with the secondary side of the reactor module for the DHRS to remove heat from the primary side via the SGs to the reactor pool via the DHRS condenser.

DCA Part 2, Tier 2, Table 5.4 5, "Decay Heat Removal System Design Data," documents the design parameters for the DHRS. DCA Part 2, Tier 2, Table 5.4 8, "Failure Modes and Effects Analysis - Decay Heat Removal System" includes a failure modes and effects analysis (FMEA), documenting the system capability and response to a set of component failures. DCA Part 2, Tier 2, Figures 5.4 11 through 5.4 16 show the thermal-hydraulic performance of the DHRS under various conditions and inventory configurations.

ITAAC: The DHRS is part of the NPM, described in DCA Part 2, Tier 1, Section 2.1. DCA Part 2, Tier 1, Section 2.1, provides a high-level description of the systems comprising the power module, including the DHRS and its specific components. The specific ITAAC associated with the DHRS are Items 15 and 20 of DCA Part 2, Tier 1, Table 2.1-4, in addition to components that form part of the DHRS that are subject to the ASME Code construction ITAAC in DCA Part 2, Tier 1, Table 2.1-4, as well as Item 8 in DCA Part 2, Tier 1, Table 2.8-2.

Technical Specifications: TS for the DHRS appear in TS Section 3.5.2, "Decay Heat Removal System (DHRS)." DHRS operability is also determined from the operability of the MSIVs and FWIVs, which have operating limits stipulated in TS Sections 3.7.1, "Main Steam Isolation Valves (MSIVs)" and 3.7.2, "Feedwater Isolation," and the reactor pool temperature, which has operating limits specified in TS Section 3.5.3, "Ultimate Heat Sink" directly affects system performance.

Technical Reports: There are no TRs associated with DCA Part 2, Tier 2, Section 5.4.3.

5.4.4.3 Regulatory Basis

SRP Section 5.4.7, "Residual Heat Removal (RHR) System," includes the relevant requirements of NRC regulations for this area of review and the associated acceptance criteria, as summarized below, as well as review interfaces with other SRP sections:

- GDC 2, as it relates to the requirement to protect the DHRS from natural phenomena
- GDC 4, as it relates to the requirement to ensure the DHRS is compatible with the environmental effects of normal operation and accident conditions, and to be protected against dynamic effects
- GDC 5, "Sharing of Structures, Systems, and Components," as it relates to the requirement that sharing of SSCs shall not significantly impair their ability to perform their safety functions
- GDC 14, as it relates to the requirement that the RCPB (for the DHRS, the SGs) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture
- GDC 19, "Control Room"
- GDC 34, "Residual Heat Removal," as it relates to the ability of the DHRS to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded
- GDC 44, "Cooling Water," as it relates to the ability of the reactor pool to act as the ultimate heat sink to remove heat from the reactor module or DHRS
- GDC 45, "Inspection of Cooling Water System"
- GDC 46, "Testing of Cooling Water System"
- GDC 54, "Piping Systems Penetrating Containment," as it relates to the detection, isolation, and containment capabilities of portions of the DHRS that penetrate containment
- GDC 57, "Closed System Isolation Valves," as it relates to closed system isolation valves on piping systems penetrating primary reactor containment
- 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," as it relates to the design provisions for automatic initiation of the auxiliary FWS (for NuScale, this function would be performed with the DHRS) in an ATWS event

- 10 CFR 50.63, "Loss of All Alternating Current Power," as it relates to the design provisions for withstanding and recovering from a station blackout, including an acceptable degree of independence from the alternating current (ac) power system and the capability for removal of decay heat at an appropriate rate for an appropriate duration

Note that, although the SRP guidance uses the section numbering Section 5.4.7 and the application numbers the section as 5.4.3, the two sections reflect the same set of functional requirements.

The NRC staff notes that NuScale has proposed a principal design criterion (PDC) rather than a GDC for criteria 34 and 44. The PDC proposed by NuScale are functionally identical to the GDC, with the exception of the discussion related to electric power. SER Chapter 8 discusses NuScale's reliance on electric power and the related exemption to GDC 17, "Electric Power Systems," as well as the staff's evaluation, dated December 13, 2017, of TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems" (ADAMS Accession No. ML17340A524).

The guidance in SRP Section 5.4.7, in conjunction with DSRS Section 5.4.7, "Decay Heat Removal (DHR) System Responsibilities," lists the acceptance criteria adequate to meet the above requirements and includes review interfaces with other SRP sections. In addition, the following guidance document provides acceptance criteria that confirm that the above requirements have been adequately addressed:

- BTP 5-4, "Design Requirements of the Residual Heat Removal System," both for the SRP and the related NuScale-specific DSRS

5.4.4.4 Technical Evaluation

The DHRS is one of a number of areas for which the NuScale design uses a passive system to satisfy safety requirements. The DHRS is only credited for events in which the RCS is intact and normal, secondary-side cooling (which is not safety related) is not available; the main condenser and FWS provide routine RHR for the NuScale design. The DHRS is a separate system for each reactor module and branches off from the main feedwater and MSS to a condenser for each DHRS train that sits on either side of a power module submerged in the shared reactor pool.

The NuScale design complies with GDC 2 through the location of the DHRS and the fabrication standards associated with the system. The majority of the DHRS is located either inside containment or submerged in the reactor pool, with the portion that is not submerged contained under the bioshield structure that covers each module bay. The DHRS components are seismic Category I and are designed to withstand the effects of a design-basis seismic event. The applicant credited the reactor building and pool structure (which are shared among all the modules) as protection from other natural phenomena; SER Chapter 3 further discusses the reactor building, including the NRC staff evaluation. Additionally, the reactor pool and building are the only components related to the DHRS shared among the modules. SER Section 9.2 contains the staff's evaluation of the reactor pool. The sharing of the reactor building and pool, as discussed in these other SER sections, will not significantly impair the ability of the DHRS to perform its function; therefore, the NuScale design conforms with GDC 5 as it applies to the DHRS.

The NRC staff audited the DHRS equipment design specifications with regard to equipment qualification (ADAMS Accession No. ML18025B026). In its design specifications, the applicant outlined the limiting parameters for the equipment in the DHRS. Much of the piping, the passive condenser, and restriction orifice are located below the reactor pool water level and are normally submerged. These components are qualified for the pool environment, which can sustain temperatures of 100 degrees C (212 degrees F) for extended periods of time in long-term cooling scenarios (when all the modules are rejecting decay heat to the pool). Additional piping and the DHRS actuation valves are in the vapor space under the module bioshield; DCA Part 2, Tier 2, Table 3C 1, “Environmental Qualification Zones - Reactor Building” denotes this as environmental qualification (EQ) Zone G. DCA Part 2, Tier 2, Figure 3C 3, “Bounding Envelope for Average Vapor Temperature at Top of Module (Zone G)” gives the operating curve for the components under the bioshield. SER Section 3.11, includes the staff’s evaluation of the bounding curve. The staff determined that the limiting parameters specified for DHRS components were below those specified in DCA Part 2, Tier 2, Table 3C 7, “Design Basis Event Environmental Conditions” and Figure 3C 3. Therefore, because of the considerations mentioned above, the staff finds the DHRS conforms to the requirements associated with GDC 4.

As applied to the DHRS, the requirements of GDC 14, associated with maintaining the RCPB, focus on the SG. The probability of a tube failure is minimized in part by the CVCS, which is a nonsafety-related system discussed in DCA Part 2, Tier 2, Chapter 9, Section 9.3.4, “Chemical and Volume Control System,” and evaluated by the staff in the corresponding SER section. In the event of a tube failure, the DHRS is designed to the same system pressure as the RCS, therefore acting as a functional backup to retain RCS integrity during a tube rupture. Although such an event would likely render the affected train of DHRS inoperable, the opposite train would remain available to remove decay heat as required. Based on the above considerations, the staff finds the DHRS meets the requirements associated with GDC 14, as applied to the DHRS boundaries.

The staff reviewed the DHRS to determine whether safety-related class 1E power was required; the applicant’s proposed design does not rely on safety-related power, as stated in TR-0815-16497, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” dated October 29, 2015 (ADAMS Accession No. ML15306A065). Because the actuation valves open on deenergization (the only powered component of the system required to change state) and the system is then driven by natural, passive forces, as discussed below, the system does not rely on power to operate. For this reason, in the context of the technical discussion below, the staff determined the DHRS can fulfill its design bases without any need for safety-related power.

As discussed in DCA Part 2, Tier 2, Section 5.4.3, the design has two DHRS trains, each consisting of piping from an SG and branching off before the MSIV to two DHRS actuation valves in parallel. These two actuation valves are the only portion of the DHRS required to change state to begin system operation, although the steam and feedwater systems must also isolate to achieve a closed loop. Because the actuate-to-open portion of the DHRS has two valves in parallel, the design satisfies the single-failure provision for the DHRS for any given train. However, the isolation function contains only a single safety-related valve, so a single failure can render an individual train inoperable if only safety-related components are credited. A single functional train of DHRS is sufficient to cool down the power module for all events for which the DHRS is credited. The DHRS is dependent on the MSIVs and FWIVs, but a single failure can disable only a single train of the DHRS. For an SG tube rupture (SGTR) event, which renders a single train of the DHRS inoperable, the applicant analyzed the limiting condition in DCA Tier 2, Section 15.6.3, “Steam Generator Tube Failure (Thermal Hydraulic)” which demonstrates that a failure of an MSIV is more limiting from a release perspective (rather than from the perspective of most limiting on DHRS cooldown performance). SER Section 5.6 further evaluates the SGTR event. In addition, the applicant has classified the DHRS piping outside as robust piping not subject to break considerations in a similar fashion to piping between the containment and the exterior isolation valve (SER Section 3.6 further discusses this characterization).

The primary function of the DHRS is RHR for non-LOCA events and thereby satisfies a portion of the requirements associated with GDC 34. To evaluate the ability of the DHRS to meet GDC 34, the staff reviewed the system description, design layout, heat removal capacity, performance capabilities, instrumentation, and all associated documentation, as detailed below.

As part of review of the DHRS, the NRC staff audited NuScale design documentation related to the DHRS (ADAMS Accession No. ML18025B2026). The limiting transient for the performance of the DHRS is the feedwater line break, and SER Section 15.2.8 and the DCA highlight representative performance parameters for that event. DCA Part 2, Tier 2, Table 5.4-5, details the performance characteristics of the system, including information on the condenser performance and assumed fouling factor in the analyses documented in DCA Part 2, Tier 2, Chapter 5, Figures 5.4-11 through 5.4-15.

DCA Part 2, Tier 2, Table 5.4-8, provides a comprehensive FMEA for the DHRS components. The FMEA lists all the components that actuate, both in the DHRS and associated consequential components in the steam and feedwater system. Additionally, the FMEA provides failure consequences for passive components in the DHRS loop. The FMEA concludes that a single failure will not compromise the design function of the DHRS; the staff agrees with this assessment.

In the response to RAI 8817 (ADAMS Accession No. ML17352B242), the applicant provided a justification for the proposed demonstration of DHRS performance. The response references prototypic testing, NRELAP5 analyses, ITAAC, and the PRA insights. Although the staff agrees that each of these considerations plays a role in determining the adequacy of the DHRS, they are not sufficient to provide reasonable assurance that the as-built system performance will be bounded by the analytical model. In particular, the assumption that underlies many of the aspects of the response—that if the DHRS actuation valves open, the system will perform adequately—is not sufficient to demonstrate the thermal performance of the DHRS. In a supplemental response to RAI 9570 (ADAMS Accession No. ML18320A276), a followup to RAI 8817, the applicant proposed a first-plant-only test using the module heatup system and compared it to an analysis performed using NRELAP to demonstrate the as-built DHRS performance is within the bounds calculated by a conservative analysis using the licensing-basis analysis tool of record. This test acts to fulfill the as-built justification of performance in accordance with the analytical model. The staff is tracking this revision to the DCA as **Confirmatory Item 5.4.4-3**.

The staff has reviewed the ITAAC related to the DHRS against the requirements of 10 CFR 52.47(b)(1). DCA Part 2, Tier 1, Table 2.1-4, Items 15 and 20, and Table 2.8-2, Item 8, specifically address the performance of the DHRS. The ASME report ITAAC, DCA Tier 1, Table 2.1-4, Items 1 through 5, are also applicable to the DHRS but are evaluated in SER Chapter 3. The staff used the acceptance criteria defined in SRP Sections 14.3 and 14.3.11, “Containment Systems—Inspections, Tests, Analyses, and Acceptance Criteria,” and the ITAAC-related guidance in RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” Sections C.1.14.3 and C.II.1.2.11 to evaluate the ITAAC. The staff finds that the three ITAAC items mentioned above are necessary and sufficient to verify the DCA Tier 1 design commitments for the operation of the components in the DHRS, as this set of ITAAC, if satisfied, demonstrates the structural and functional performance requirements of the system are met. The staff finds reasonable assurance that, if the proposed ITAAC are performed and the acceptance criteria are met, the as-built top-level design parameters described in DCA Tier 1 would be in conformity with the certified design with respect to the parameter values used in the safety analyses corresponding to the DHRS. Based on this review and a review of the selection methodology and criteria for the development of the Tier 1 information in the DCA, the staff concludes that the top-level functional design for the DHRS is appropriately described in DCA Tier 1, and the Tier 1 information is acceptable. Consequently, the staff finds that the NuScale DHRS meets the requirements of 10 CFR 52.47(b)(1).

Based on the information in DCA Tier 2, including the piping and instrumentation diagram in DCA Part 2, Tier 2, Figure 5.4-10, system description, and audited material (ADAMS Accession No. ML18025B026, and the proposed testing, the staff determined there is sufficient information to provide assurance the system will operate in accordance with the analysis assumptions made by the applicant.

In DCA Tier 2, Section 5.4.3.3.2, “System Noncondensible Gas”, the applicant notes that the DHRS (and associated secondary system components) does not include safety related high point venting capability. In light of the reduced margins for operating a natural convection (rather than forced circulation) system, the staff reviewed the assumptions related to noncondensable gases within the DHRS in detail. This was reinforced in the thermal-hydraulic performance analyses audited by the staff, which demonstrated that DHRS performance was sensitive to both low inventory and very high inventory. Abnormally large amounts of noncondensable gases could cause low loop inventories. Additionally, sensitivity studies performed by the applicant demonstrated that the presence of noncondensable gases in the DHRS resulted in degraded heat transfer performance. The assumed amount of noncondensable gases was based on a limiting amount above the level sensors (discussed further below) plus a conservatively high amount dissolved in the loop fluid. This assumed amount of noncondensable gases is present in the DHRS for the low- and high-inventory cases documented in DCA Part 2, Tier 2, Figures 5.4-12 through 5.4-15, “Primary Coolant Temperature Cooldown with Decay Heat Removal System One Train Operation: Low System Inventory 4, Hours,” Primary Coolant Temperature Cooldown with Decay Heat Removal System One Train Operation: Low System Inventory - 36 Hours,” Primary Coolant Temperature Cooldown with Decay Heat Removal System One Train Operation: High System Inventory - 4 Hours,” and “Primary Coolant Temperature Cooldown with Decay Heat Removal System One Train Operation: High System Inventory - 36 Hours,” respectively. The staff believes these are reasonably conservative assumptions based on the design, and, in conjunction with the ASME design report required as part of ITAAC Item 02.01.01 to verify piping designs conform to ASME Code, Section III, the design provides reasonable assurance that the DHRS will function in the presence of a limiting amount of noncondensable gases.

DCA Part 2, Tier 2, Chapter 5, Figures 5.4-11 through 5.4-15, demonstrate DHRS performance out to 36 hours with a temperature of less than 420 degrees F (215.6 degrees C), the acceptance criterion adopted by the applicant as endorsed by the NRC for advanced passive plants in SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. Because the NuScale design does not use any active, safety-related systems to cool down in the event of a non-LOCA transient, it would be difficult to meet the traditional expectation of a cold shutdown associated with a plant using active safety systems. Instead, for passive plant designs, the industry proposed an acceptance criterion for shutdown of an RCS temperature of 420 degrees F (215.6 degrees C), at 36 hours following a transient. The NRC allowed for this alternative criteria in SECY-94-084 for advanced passive LWRs (also used for the design certification of the AP1000), and NuScale has employed similar reasoning for its acceptance criteria, as was noted in the staff requirements memorandum to SECY-94-084, dated June 30, 1994.

The figures presented by the applicant assessed three cases: a nominal, two-train cooldown case; a single-train low-inventory case; and a single-train high-inventory case. Of the three presented, the high-inventory case (Figure 5.4-15) is the most limiting from an RCS temperature perspective, but all maintain acceptable core parameters. These events do not directly parallel design-basis events as set forth in DCA Part 2, Tier 2, Chapter 15, and in its response to RAI 8745, Question 05.04.07 3 (ADAMS Accession No. ML17179A864), the applicant stated that the limiting event for the DHRS is the feedwater line break inside the containment, as stated in DCA Part 2, Tier 2, Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside of Containment." As a result of the break, one of the DHRS trains is disabled, and the remaining DHRS train provides core cooling without challenging the transient acceptance criteria.

The applicant calculated the performance curves using NRELAP5 version 1.2. The staff reviewed the applicability of the code for performing analyses of this type and found it acceptable as documented in the staff safety evaluation for the Topical Report (TR)-0516-49416-P, "Non-Loss-of-Coolant Accident Evaluation Model" (ADAMS Accession No. MLXXXXXXXXX). The staff is tracking this as **Open Item 15.0.2-4**, pending the completion of the staff's review of the non-LOCA topical report.

During the audit (ADAMS Accession No. ML18025B026) the staff reviewed the inputs to the calculations performed for DCA Part 2, Tier 2, Section 5.4.3, and found that NuScale used a value of approximately 100 degrees F, rather than the TS maximum of 140 degrees F (60 degrees C, in DCA Revision 2), as the initial condition for the reactor pool temperature. As a bounding analysis, NuScale performed sensitivity studies demonstrating the effect of a substantially increased reactor pool temperature (212 degrees F), as documented in DCA revision 2, Part 2, Tier 2, Figure 5.4-16. The system performance remains within the specified design limits.

For the DHRS transients discussed above, between the arbitrary limiting transients shown in DCA Part 2, Tier 2, Figures 5.4-11 through 5.4-15, and the feedwater line break, core cooling is maintained such that neither specified fuel design limits nor the reactor coolant boundary design conditions are exceeded, and the core remains covered. Therefore, when considering the staff's evaluation of the DHRS and the thermal performance analyses documented above, the staff finds the DHRS meets the GDC 34 requirements with respect to events in which the RCS is not breached. The ECCS, in conjunction with the containment heat removal system, acts to meet RHR requirements for other transients, or in the longer term, as discussed below.

DCA Part 2, Tier 2, Chapter 5, Section 5.4.3.2.2, "Instrumentation and Controls," discusses instrumentation for the DHRS. The system is required to have adequate instrumentation to provide reasonable assurance that the system will operate as intended under all conditions and that, once actuated, operators in the control room have sufficient indication to assess the performance of system function. Level instrumentation, in the form of two transmitters on each steam pipe just before the DHRS actuation valves, indicates that the system is filled. In the detailed system specifications audited by the staff (ADAMS Accession No. ML18025B026), the DHRS had no vent pathways available except for those on the secondary side, which are normally isolated or unavailable from the DHRS during normal operation.

TS SR 3.5.2.1 requires that the DHRS be filled to be operable. As discussed above, this consideration is important in assessing the available inventory in the DHRS as assumed in the safety analyses. The only other SR directly associated with the DHRS relates to the actuation time of the DHRS valves—these valves are required to be operable in order for the DHRS to be operable. As discussed above, the MSIVs and feedwater regulating valves must also be operable for the DHRS to be operable; these components have TS associated with them in TS Sections 3.7.1 and 3.7.2, respectively.

Further instrumentation for the DHRS consists of DHRS actuation valve position and nitrogen accumulator pressure for valve operability, condensate temperature and pressure indication in the condenser header, and steam pressure indication. Each DHRS line has two sensors upstream of the actuation valves and level transmitters for steam pressure, and two temperature sensors and three pressure sensors in the header of the condenser (as shown in DCA Part 2, Tier 2, Figure 5.4-10). No flow rate instrumentation exists—instead, a combination of the temperature and pressure sensors in the condenser and the RCS pressure and temperature instrumentation indicate DHRS performance. This I&C inventory provides the operators in the control room adequate information to assess the operability and infer the performance of the DHRS and therefore meets the requirements associated with GDC 19, as applied to the DHRS.

Ultimately, in the absence of all nonsafety-related systems, the DHRS will yield the decay heat removal function to the ECCS after 24 hours. The ECCS actuates after 24 hours of low ac electrical distribution system voltage, as would be the case for a type of station-blackout event. DCA Part 2, Tier 2, Figures 8.4 1 through 8.4 4, demonstrate system performance under this type of scenario, and SER Section 8.4 further discusses this scenario. For the performance of the DHRS before the 24 hour point, the station-blackout scenario is bounded by the single train cases shown in DCA Part 2, Tier 2, Section 5.4.3, and the feedwater line break inside the containment, as discussed in DCA Part 2, Tier 2, Section 15.2.8.

Water hammer is an important consideration in two-phase piping systems. The applicant provided an evaluation for the potential mechanisms for water hammer in the system in DCA Part 2, Section 5.4.3.3.1, “Water Hammer”, and has determined that mechanisms that could cause water hammer do not apply to the DHRS. Because of the nature of the system—a natural convection, low flow rate system relying on steam produced from the SGs coupled to condensation in the condenser to drive flow—the staff agrees that water hammer presents an unlikely scenario during the design-basis conditions for the system, except during the establishment of flow when the system is actuated. The applicant further noted that one water hammer mechanism, counter flow, is precluded because of the slope designed into the DHRS piping. Staff review indicated water hammer during flow establishment was not a significant effect during design basis operating conditions due to the design considerations of low differential pressure and relatively low condensation rates caused by the design temperatures of the DHRS piping runs.

An important note related to the DHRS is that the system is only functional when the power module is assembled. The DHRS is not credited nor available for use during refueling or shutdown when the module does not have a closed RCS available. As such, the evaluation for this section concerns decay heat removal for events beginning in mode 3 or higher. For refueling operations, the containment is flooded using the containment flood and drain system, and cooling is achieved in a similar fashion to the long-term cooling case discussed above by transferring heat from the containment to the reactor building pool.

NuScale credited the DHRS for heat removal for events in which the reactor returned to a low power level later in the transient sequence (evaluated in SER Section 15.0.6). DCA Part 2, Tier 2, Section 5.4.3.1, "Design Basis," states that the design basis for the DHRS is to remove post-reactor trip residual and core decay heat from operating conditions, as well as to provide further context for scenarios in which the reactor is not shut down but is stable, stating that the design is also capable of cooling the RCS during a return-to-power scenario such that reactor pressure boundary and fuel design limits are not exceeded. DCA Tier 2, Figure 15.0 11 , "Return to Power – Average Reactor Coolant System Temperature (Peak Power Case)," documents the DHRS performance in maintaining RCS temperature, a primary figure of merit for the adequacy of the DHRS for these design-basis events. The figure demonstrates the DHRS is capable of adequate heat removal for the transient of interest. The applicant made these changes in the response dated May 17, 2018, to RAI 9442, Question 05.04.07 6 (ADAMS Accession No. ML18137A407), and the staff is tracking the changes as **Confirmatory Item 5.4.4-4**.

SER Section 9.2, which includes the NRC staff's findings related to PDC 44 and GDC 45 and 46, discusses the ultimate heat sink. DCA Part 2, Tier 2, Section 9.2 .5, "Ultimate Heat Sink," gives the parameters for the reactor pool, which serves as the shared ultimate heat sink for all the modules. The DHRS, as described in the DCA and evaluated here, can remove heat from the reactor module and transfer it to the reactor pool, the ultimate heat sink.

SER Section 6.2.4, which includes the NRC staff's findings related to GDC 54 and 57, discusses the containment isolation system.

5.4.4.5 Combined License Information Items

There are no COL information items related to DCA Part 2, Tier 2, Section 5.4.3.

5.4.4.6 Conclusion

The staff finds that NuScale has fully addressed the required information related to the DHRS. However, because of the open item related to the non-LOCA TR, the staff was unable to finalize its conclusions as to acceptability.

5.4.5 Reactor Coolant System High-Point Vents Exemption

5.4.5.1 Introduction

DCA Part 2, Tier 2, Chapter 5, Section 5.4.4, "Reactor Coolant System High-Point Vents," addresses the high-point vents for the RCS used in the NuScale design. In 10 CFR 50.46a, "Acceptance Criteria for Reactor Coolant System Venting Systems," and 10 CFR 50.34(f)(2)(vi), the NRC requires high-point vents to ensure adequate core cooling. As part of its DCA, NuScale requested an exemption from 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi), which require the high-point venting capability for the RCS and other systems needed for core cooling. The staff's technical evaluation of the exemption follows.

5.4.5.2 Regulatory Basis

In 10 CFR 52.47(a), the NRC states, in part, the following:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:...
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f)....

In part, 10 CFR 50.34(f) states the following:

> (f) In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section….

In part, 10 CFR 50.34(f)(2)(vi) states the following:

> (vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling…

In part, 10 CFR 50.46a states the following:

> Each nuclear power reactor must be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems….

In 10 CFR 52.7, "Specific Exemptions," the NRC states the following:

> The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part. The Commission's consideration will be governed by § 50.12 of this chapter, unless other criteria are provided for in this part, in which case the Commission's consideration will be governed by the criteria in this part. Only if those criteria are not met will the Commission's consideration be governed by § 50.12 of this chapter. The Commission's consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts.

In part, 10 CFR 50.12(a) states the following:

> The two conditions which must be met for granting an exemption are:

(1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

(2) The Commission will not consider granting an exemption unless special circumstances are present. (Circumstances are enumerated in 10 CFR 50.12(a)(2)).

5.4.5.3 Summary of Application

NuScale provided the following justification for the exemption:

> The underlying purpose of 10 CFR 50.46a, requiring high-point vents for the RCS and RV, is to preclude an accumulation of noncondensible gases that may inhibit core cooling. As stated in 68 *FR* [*Federal Register*] 54123:
>
> > This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.
>
> Similarly, as stated in NUREG-0737 for TMI Item II.B.1, the purpose of 10 CFR 50.34(f)(2)(vi) is to prevent the accumulation of noncondensible gases that may inhibit core cooling during natural circulation.
>
> The NuScale design supports natural circulation core cooling without reliance on the RCS and RV high point venting specified by 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). In the NuScale design, natural circulation is not inhibited by the accumulation of noncondensible gases and core cooling is not dependent on pump operation. Therefore, the underlying purpose of the requirements is met without high point vents.

NuScale provided the following technical basis for the exemption:

> The NuScale design includes an RCS that is integral to the RV; the core, steam generator, and pressurizer are contained in the vessel. Therefore, the high point of the RCS flow loop and pressurizer is the high point of the RV. The accumulation of noncondensible gases in the RCS and pressurizer steam space is minimized during normal operation via the high point degasification line.
>
> As described in DCA Part 2 Tier 2, Section 5.4.4, the ECCS includes three reactor vent valves (RVVs) located on the top of the RV that discharge to the CNV upon ECCS actuation, thereby venting any non-condensable gases accumulated in the pressurizer space. Thus, during ECCS operation the ability of the ECCS to maintain adequate core cooling is not impeded. The RCS does not include separate post-accident high point vent capability. During DHRS

> cooling events, accumulation of non-condensable gases in the pressurizer will not impact the ability of the DHRS to maintain core cooling because the pressurizer volume is not in the DHRS cooling flow path. Accumulation of non-condensable gas in the RV during DHRS operation does not affect the RV level because the liquid phase is incompressible. Liquid circulation in the RV during DHRS operation is therefore not significantly affected by accumulation of non-condensable gas in the RV. Noncondensible gas accumulation within the secondary system was calculated and considered in the DHRS performance analysis, summarized in DCA Part 2 Tier 2, Section 5.4.3, and determined not to impede DHRS operation.

According to the applicant, no other systems are necessary to maintain adequate core cooling that require high-point venting. As described in DCA Part 2, Tier 2, Section 6.2, "Containment Systems," accumulated noncondensable gases vented to the CNV during ECCS operation will not challenge adequate core cooling. Therefore, the underlying purpose of the 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) requirements to preclude the accumulation of noncondensable gases that may prevent adequate core cooling is met without reliance on the features required by this rule.

5.4.5.4 Technical Evaluation

The regulation in 10 CFR 50.34(f)(2)(vi) was one of a number of requirements imposed by the Commission after the events at TMI. The NRC intended that 10 CFR 50.34(f)(2)(vi) would allow operators an avenue to vent noncondensable gases that could impede the circulation of coolant. The event at TMI demonstrated that the potential existed for noncondensable gases to interrupt flow under certain accident scenarios and further impair the ability of other safety systems to perform their intended function. In NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," issued May 1980, the staff stated the intent of the requirement as follows:

> The vents are to provide the ability to deal effectively with the unexpected presence of non-condensable gases in the RV and primary coolant system, particularly in quantities that could interfere with coolant flow and distribution, by establishing a safe vent path.

The underlying purpose of the rule, therefore, was to provide a means to permit venting of noncondensable gases that could interfere with coolant circulation, whether natural circulation or pumped.

Based on the staff review, the NuScale design accomplishes the ECCS function in a substantially different fashion from a traditional PWR. Ultimately, core cooling is accomplished by opening the RCS to the containment to create a recirculation flow via steaming from the three reactor vent valves in the steam space of the PZR and allowing the water that condenses in containment to reenter the RV via the two reactor recirculation valves lower along the vessel. The design fulfills long-term cooling through the transfer of the heat by convection and conduction from the RV through the containment to the reactor pool.

Staff evaluated the design basis events, and determined that in any scenario in which noncondensable gases do accumulate inside the RCS, the actuation of the ECCS acts as an effective vent for noncondensable gases. Upon the point of ECCS actuation, any accumulated noncondensable gases would be relocated to the now-shared RV and containment vapor space, where they would not impair flow (though they could affect heat transfer, as evaluated below).

Because the ECCS function is tightly coupled to the capability of the containment to transfer heat to the ultimate heat sink, the NRC staff reviewed the impact of additional noncondensable gases on the long-term cooling and containment analyses. As part of Technical Report, TR-0916-51299-P, “Long-Term Cooling Methodology” dated January 9, 2017, (ADAMS Accession No. ML17009A490) the TR on the long-term cooling methodology, the applicant performed sensitivity studies, including the effects of additional noncondensable gases. In initializing the calculation, the applicant added more noncondensables in an amount substantially larger than could be present in containment initially to account for noncondensable gases dissolved in the RCS and present in the PZR. As a result, the staff issued RAI 9020, Question 05.04.12-1. In the response dated September 26, 2017 (ADAMS Accession No. ML17269A293), the applicant described in further detail a sensitivity calculation performed for a conservative initial air inventory inside the containment. Further refinement of those assumptions indicated that the inventory of noncondensable gases inside the containment would be substantially smaller than that assumed in the analysis. These sensitivities demonstrating the minimum cooldown rate, which included varying other parameters in addition to the noncondensable gas concentration, had substantial effects on many of the figures of merit, but all remained below acceptable values for the design.

Although the applicant considered the effects in the later stages of the transient as part of TR-0916-51299-P, the staff was unable to determine whether the LOCA analyses demonstrating the peak containment pressure appropriately considered the effects of noncondensable gases. The long-term cooling analysis did not use the same model as that used in the LOCA evaluation analysis for peak containment pressure. The staff is tracking this as an **Open Item 15.0.2-2**, pending the completion of the staff’s review of the LOCA topical report.

The NRC staff also reviewed the potential impacts of noncondensables to the DHRS, which provides core cooling for non-LOCA events. The DHRS is not connected to the RCS, and therefore any high-point vents for the RCS would not provide relief for noncondensable gases in the DHRS. Because of the nature of the design and the loop layout, a concentration of noncondensable gases sufficient to impede natural circulation through the SGs does not represent a design-basis event, as there are no local high points in the SG region that exist to impede flow. The applicant considered the noncondensable gas concentrations in the DHRS as part of the system design. The design provides level sensors in the DHRS to sense up to a limiting amount of noncondensable gases considered in the analysis. SER Section 5.4.4 further evaluates these considerations.

Additionally, the NuScale design does include provisions for venting in the form of a nonsafety-related high-point degasification line connected to the steam space of the PZR. This allows for routine degasification during normal operation, as required, though this system cannot be credited because of its nonsafety-related classification to “minimize…[the] accumulation of noncondensible gases in the RCS and PZR steam space” in any accident scenarios. The staff based its finding of acceptability on the nature of the RCS and ECCS designs themselves and the fact that the presence of noncondensable gases in the RCS does not inhibit operation of the ECCS function.

The staff concludes that the requested exemption will not impact the consequences of a design-basis event, nor will it provide for a new, unanalyzed event. The applicant has considered the impact of the system performance in the presence of a limiting amount of noncondensable gas. In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46a is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security.

Accordingly, the NRC staff determined that the applicant has met the underlying purpose of the rules, to provide a means to permit venting of noncondensable gases that could interfere with coolant circulation. Therefore, the NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the underlying purpose of the rules is not necessary to achieve compliance with the intent of 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46a. As discussed above, because of the nature of the NuScale design, the design achieves compliance with the rules through the functional arrangement of the RCS and ECCS in combination with the design considerations that prevent a limiting amount of noncondensable gases from impairing system performance in the event of a transient.

5.4.5.5 Conclusion

The staff finds that NuScale has fully addressed the required information related to the exemption request. However, because of the open item related to the LOCA TR, the staff was unable to finalize its conclusions as to acceptability.

5.4.6 Pressurizer Exemptions

5.4.6.1 Introduction

RCS pressure is controlled by the PZR, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the PZR, or it is condensed by the PZR spray to limit pressure variations caused by contraction or expansion of the reactor coolant.

The PZR comprises the upper region of the RV, separated from the region of naturally circulating reactor coolant by a baffle plate. The baffle plate provides a low-resistance flowpath between the PZR and the RCS to rapidly communicate pressure changes between the two regions. The principal function of the PZR is to provide a surge volume of saturated water and steam that regulates RCS pressure by maintaining a saturated steam-water interface during heatup, startup, normal, transient, and faulted operating conditions.

Although the SRP does not contain a Section 5.4.10, the review scope of SRP Section 5.4 includes the PZR. SER Sections 3.9.1, 3.9.2, 3.9.3, 5.2.2, 5.2.3, 5.2.4, and 15.6.6 contain detailed information about the PZR, including the staff's evaluation and conclusion about PZR design features and performance requirements.

5.4.6.2 Technical Evaluation and Conclusion

The SER sections listed above discuss the staff's technical evaluation and conclusion on PZR design features and performance requirements.

5.4.6.3 Pressurizer Component Exemptions

5.4.6.3.1 Introduction

In NuScale DCA Part 7, “Exemptions,” the applicant requested an exemption from 10 CFR 50.34(f)(2)(xiii) and from a portion of 10 CFR 50.34(f)(2)(xx), which require, in part, the provision of emergency power for PZR heaters and PZR level indication, respectively. The applicant stated that the NuScale design does not rely on PZR heaters or PZR level indication to achieve and maintain natural circulation during a loss of electrical power and therefore meets the underlying purpose of 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx).

5.4.6.3.2 Regulatory Basis

In part, 10 CFR 52.47(a) states the following:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:....
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f)....

In part, 10 CFR 50.34(f) states the following:

> (f) In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section….

In part, 10 CFR 50.34(f)(2)(xiii) states the following:

> (xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available.

In part, 10 CFR 50.34(f)(2)(xx) states the following:

> (xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources.

In 10 CFR 52.7, the NRC states the following:

> The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part. The Commission's consideration will be governed by § 50.12 of this chapter, unless other criteria are provided for in this part, in which case the Commission's consideration will be governed by the criteria in this part. Only if those criteria are not met will the Commission's consideration be governed by § 50.12 of this chapter. The Commission's consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts.

In part, 10 CFR 50.12(a) states the following:

> The two conditions which must be met for granting an exemption are:
>
> (1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.
> (2) The Commission will not consider granting an exemption unless special circumstances are present. (Circumstances are enumerated in 10 CFR 50.12(a)(2)).

5.4.6.3.3 Summary of Application

For the requested exemptions from 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx), the applicant provided a summary of the exemption request, justification for the request, and technical and regulatory bases. The applicant quoted the following excerpt from NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," issued July 1979, Recommendation 2.1.1, "Emergency Power Supply Requirements for the PZR Heaters, Power-Operated Relief and Block Valves, and PZR Level Indicators in PWRs," which NUREG-0737 references as the basis of 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx):

> In some designs, loss of pressurizer heaters due to a loss of offsite power requires the use of the high-pressure emergency core cooling system to maintain reactor pressure and volume control for natural circulation cooling. Similarly, in some designs the inability to close the power-operated relief valve upon loss of offsite power could result in additional challenges to the high-pressure emergency core cooling system. Finally, proper functioning of the pressurizer level instrumentation is necessary to maintain satisfactory pressure control for natural circulation cooling using the pressurizer heaters.

The applicant stated that the underlying purpose of the requirements is to enable and maintain natural circulation cooling in a loss-of-offsite-power (LOOP) condition. The applicant explained that the DHRS, a passive system requiring no electrical power, achieves and maintains natural circulation without the need for PZR level indication or PZR heaters. The applicant also stated that the NuScale design does not include PZR relief valves or PZR block valves, rendering the portions of 10 CFR 50.34(f)(2)(xx) related to those components technically irrelevant. As a result, the applicant concluded that the NuScale design meets the underlying purpose of the relevant requirements.

5.4.6.3.4 Technical Evaluation

Traditional PWRs use forced circulation for core cooling and rely upon PZR heaters to maintain natural circulation following a LOOP and resulting loss of forced circulation. Following the events at TMI, the NRC imposed several new requirements, including 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx). According to NUREG-0578, the underlying purpose of 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx) was to ensure natural circulation core cooling ability while decreasing reliance on the ECCS to establish and maintain natural circulation following a LOOP. Providing emergency power to the PZR level indication and PZR heaters would increase their reliability to maintain pressure control in the RCS during natural circulation while decreasing demands on the ECCS.

Unlike traditional PWRs, the NuScale design uses natural circulation during normal operation and does not credit PZR heater operation to maintain RCS pressure and natural circulation following a LOOP. The transient analyses in DCA Part 2, Tier 2, Section 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries," show that the NuScale design requires no PZR heater operation, and therefore no need for PZR level indication, following a LOOP. Power is lost to the PZR heaters upon the LOOP. The DHRS removes heat from the RCS via the SGs and discharges it to the reactor pool, thereby maintaining natural circulation in the RCS. The DHRS is a passive system that does not require electrical power for actuation or operation. Furthermore, it does not actuate the ECCS. SER Section 15.2.6 provides the staff's evaluation of the LOOP transient analyses.

The staff concludes that the requested exemptions will not affect power operation or the consequences of a design-basis event, nor will they create a new accident. In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemptions to 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx) are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Furthermore, in accordance with 10 CFR 50.12(a)(2), the staff finds that a special circumstance is present, specifically, provision (ii), that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Because the NuScale design maintains natural circulation following a LOOP without the need for PZR heaters or PZR level indication and without ECCS operation, the design meets the underlying purpose of 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx). The staff also notes that the portions of 10 CFR 50.34(f)(2)(xx) related to providing emergency power to PZR relief valves and PZR block valves are not applicable because the NuScale design does not include those components. For these reasons, the staff recommends granting the requested exemptions to 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx).

5.4.6.3.5 Conclusion

For the reasons set forth in the evaluation above, the staff finds that the requested exemptions to 10 CFR 50.34(f)(2)(xiii) and 10 CFR 50.34(f)(2)(xx) meet the requirements of 10 CFR 50.12(a) and recommends granting the requested exemptions.