5 REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

This chapter of the final safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 5, "Reactor Coolant System and Connecting Systems," of the NuScale Power, LLC (hereinafter referred to as the applicant), Standard Design Approval Application (SDAA) Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 2 of the SDAA dated April 9, 2025 (Agencywide Document Access and Management System (ADAMS) Accession No. ML25099A250). The precise parameter values, as reviewed by the staff in this safety evaluation (SE), are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this SE to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

5.1 Introduction

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, 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 Integrity of Reactor Coolant Boundary

5.2.1 Compliance with Codes and Code Cases

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

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 Code (ASME BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code)) and the code editions and addenda required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and Standards," in the SDAA for the NuScale US460 Power Plant.

5.2.1.1.2 Summary of Application

FSAR, Chapter 5: FSAR, 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 2017 Edition with no addenda. The applicant referenced FSAR, 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 and Components," 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, 2017 Edition. SER Section 3.9.6 discusses the review of this item.

FSAR, Chapter 14.3: FSAR, Chapter 14.3 discusses the applicant's approach to Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC), including consideration of safety-related piping conformance with the rules of construction of ASME BPV Code, Section III.

SDAA Part 8, "License Conditions: Inspections, Tests, Analyses, and Acceptance Criteria": SDAA Part 8 includes descriptions of several systems that indicate the systems and components will conform to the rules of construction of ASME BPV Code, Section III. This Part also addresses the proposed ITAAC, which are intended to satisfy the requirements of 10 CFR 52.47(b)(1). These ITAAC are evaluated in Section 14.3.3 of this SER.

Inspections, Tests, Analyses, and Acceptance Criteria: SDAA Part 8 addresses the proposed ITAAC, which are intended to satisfy the requirements of 10 CFR 52.47(b)(1). These ITAAC are evaluated in Section 14.3.3 of this SER.

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 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, in ASME BPV Code, respectively, specified Section III

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 5.2.1.1, "Compliance with the Codes and Standards Rule, 10 CFR 50.55a," Revision 4, issued December 2016 (ML16088A127), 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 must 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 the FSAR for compliance with the requirements presented above. FSAR, Section 5.2.1.1, states that the ASME BPV Code of Record for the applicant's design is the 2017 Edition with no addenda, which has been incorporated by reference in 10 CFR 50.55a and is therefore acceptable to the staff. The ASME OM Code of Record referenced in the FSAR description of the IST program is the 2017 Edition, which the staff finds to be acceptable where implemented, as incorporated by reference in 10 CFR 50.55a. The applicant has included a Combined License (COL) Item (3.9-8) to establish Preservice and Inservice Testing Programs consistent with the requirements in the latest edition and Addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a. SER Section 3.9.6 further discusses the review of the applicant's use of the ASME OM Code, including this COL Item.

The staff notes that any proposed change by the COL applicant in the use of the ASME Code editions or addenda specified as the Code of Record for the NuScale reactor design will require NRC approval prior to implementation.

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

FSAR, Section 5.2, "Integrity of Reactor Coolant Boundary," states that the RCPB for each NPM meets 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. Based on its review, the staff finds that RCPB components, as defined in 10 CFR 50.55a, are classified properly as ASME Code Section III, Class 1 (Quality Group A) components, and that the RCPB does not include Quality Group B or Group C components.

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. FSAR, 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 FSAR, 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 stated that Quality Group A, B, and C SSCs met the applicable conditions promulgated in 10 CFR 50.55a(b), in addition to the requirements of the appropriate class from ASME BPV Code, Section III. 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 COL items in FSAR 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 Systems and Components," address the applicant's compliance with an ISI or 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. FSAR, Section 5.2.1.1, does not discuss any proposed alternatives to compliance with 10 CFR 50.55a. 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

SER Section 14.3 discusses the Staff's use of 10 CFR 52.47(b)(1) as guidance for reviewing ITAAC in an SDAA. In 10 CFR 52.47(b)(1), the NRC requires that a Design Certification Application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the [Atomic Energy] Act, and the Commission's rules and 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.

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, 2, 3, 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. These ITAAC are evaluated in Section 14.3.3 of this SER.

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 identified an acceptable ASME BPV Code of Record to apply to the applicant's 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

FSAR, Chapter 5: FSAR, Section 5.2.1.2, "Compliance with Applicable Code Cases," discusses the application of ASME BPV Code Cases to the NuScale US460 Power Plant design. FSAR, Table 5.2-1, "American Society of Mechanical Engineers Code Cases," lists the ASME BPV Code, Section III, Code Cases used by the applicant.

FSAR, Table 5.2-1, lists the following ASME Code Cases:

- ASME Code Case N-4-13, “Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and CS,” issued February 2008

- ASME Code Case N-60-6, “Material for Core Support Structures, Section III, Division 1,” issued December 2011

- ASME Code Case N-759-2, “Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1,” issued January 2008

- ASME Code Case N-774, “Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4,540 kg) and Otherwise Conforming to the Requirements of SA-336/SA-336M for Class 1, 2 and 3 Construction, Section III, Division 1,” issued September 2008

- ASME Code Case N-782, “Use of Code Editions, Addenda and Cases Section III, Division 1,” issued January 2009

- ASME Code Case N-844, “Alternatives to the Requirements of NB-4250(c) Section III, Division 1,” issued February 2014

- ASME Code Case N-845-1, “Qualification Requirements for Bolts and Studs, Section XI, Division 1,” issued April 2016

- ASME Code Case N-849, “In Situ VT-3 Examination of Removable Core Support Structure Without Removal, Section XI, Division 1,” issued September 2014

- ASME Code Case N-885, “Alternative Requirements for Table IWB-2500-1, Examination Category B-N-1, Interior of Reactor Vessel, Category B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels, Category B-N-3, Removable Core Support Structures, Section XI, Division 1,” issued December 2018

- ASME Code Case N-890, “Materials Exempted from G-2110(b) Requirement Section XI, Division 1,” issued October 2018

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

Technical Specifications: There are no TS 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, must be identified and evaluated to determine their adequacy and applicability

- 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.

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 the FSAR for compliance with the requirements presented above. Acceptable ASME Code Cases that may be used for the NuScale US460 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 SDAA. FSAR, Section 5.2.1.2 states that the ASME BPV Code, Section III Code Cases chosen for design, fabrication, and construction are from 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). FSAR, Section 5.2.1.2 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.

The staff reviewed FSAR, 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. 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 has reviewed the FSAR for compliance with the requirements presented above. Acceptable ASME Code Cases that may be used for the NuScale US460 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 SDAA. FSAR, Section 5.2.1.2 states that the ASME BPV Code, Section XI Code Cases used for preservice inspection and ISI are from those listed in the applicable ASME BPV Code edition specified in 10 CFR 50.55a(a)(1)(ii) or Tables 1 and 2 of RG 1.147 pursuant to 10 CFR 50.55a(a)(3)(ii) and subject to the applicable provisions of 10 CFR 50.55a(b). FSAR, Section 5.2.1.2 further states that Code Cases that are used and listed in Table 2 of RG 1.147 also meet the conditions established in the RG. The applicant stated that Section 5.2.4, RCPB Inservice Inspection (ISI) and Testing, and Section 6.6, Inservice Inspection and Testing of Class 2 and 3 Systems and Components, provide a summary discussion of preservice and ISI examinations and procedures.

The staff reviewed FSAR, 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 1.147. The staff finds the reference to conditionally and unconditionally approved Code Cases to be consistent with 10 CFR 50.55a and RG 1.147 and, therefore, acceptable.

The staff observes that the applicant did not mention specific ASME Code Cases approved for use in accordance with RG 1.192. Therefore, no finding is made with respect to this RG.

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 1.147 and RG 1.192.

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 applicant's approved design. FSAR, Table 1.8-1, "Combined License Information Items," 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 the FSAR are acceptable as specified in the applicable NRC RGs and are in conformance with conditions in the applicable RGs. The staff concludes that the information in the FSAR 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 applicants 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 steam generators (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 spring-operated RSVs, the battery-supplied power-operated reactor vent valves (RVVs), the thermal relief valves, and the pressurizer (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. Two 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 is protected against brittle failure. 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 that do not result in a reactor trip or containment isolation. The SEs of these components appear in their respective sections of this report.

5.2.2.2 Summary of Application

FSAR 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 provided information on these devices in FSAR Section 3.9.3.2, “Design and Installation of Pressure Relief Devices”; Section 5.4.1, “Steam Generators”; Section 5.4.5, “Pressurizer,” and Section 6.3, “Emergency Core Cooling System.”

ITAAC: Overpressure protection system ITAAC is discussed in Section 14.3.4.3 of this SER.

Technical Reports: TR-130877-P, Revision 1, “Pressure and Temperature Limits Methodology.”

Technical Specifications:

- TS Section 3.3.1, “Module Protection System Instrumentation”
- TS Section 3.4.3, “RCS Pressure and Temperature (P/T) Limits”
- TS Section 3.4.4, “Reactor Safety Valves (RSVs)”
- TS Section 3.4.10, “Low Temperature Overpressure Protection (LTOP) Valves”
- TS Section 5.6.4, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)”

Initial Test Program:

FSAR Section 14.2.12 provide the individual test descriptions related to overpressure protection.

- FSAR Table 14.2-39, “Test # 39 Reactor Coolant System” (LTOP function verified by Test # 56.02.01)

- FSAR Table 14.2-40, “Test # 40 Emergency Core Cooling System” (LTOP function verified by Test # 56.02.02)

- FSAR Table 14.2-56, “Test # 56 Module Protection System” (LTOP function verified by Test # 56.02.01, 02 and 04)

- FSAR Table 14.2-59, “Test # 59 Safety Display and Indication System” (RSV function verified by Test # 59.01.05)

5.2.2.3 Regulatory Basis

In 10 CFR Part 50, Appendix A the General Design Criteria contains:

- 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 52.137(a) states, in part, the following:

> The 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), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) of 10CFR 50.34(f):

10 CFR 50.34(f) states, in part, 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….

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 operations including AOOs and 120 percent for postulated accidents. In addition, overpressure protection is provided during startup, and shutdown conditions to protect the reactor vessel from brittle fracture.

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, safety-related, and designed to maintain pressure below 110 percent of the design pressure (i.e., 16.69 megapascals (MPa) (2420 pounds per square inch absolute (psia)). Each RSV is designed to provide at least 100 percent of the required relief capacity. The RSV is a spring-operated relief valve designed in accordance with ASME requirements. During power operations, the RSV will begin to relieve to the CNV when the RCS pressure exceeds the RSV set pressure and will reseat below the set pressure within 5%. FSAR Section 5.2.2.8, "Process Instrumentation," states that direct position indication for each RSV is provided in the control room in accordance with the requirement of 10 CFR 50.34(f)(2)(xi).

FSAR Section 5.2.2.6, "Applicable Codes and Classification," states that the RSVs are designed in accordance with ASME BPV Code, Section III, Subarticle NB 3500, and function to satisfy the overpressure protection criteria described in ASME BPV Code, Section III, Article NB 7000. This is discussed in Section 3.9.6.4.6.3, "Reactor Safety Valves," of the SER.

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.

The RSVs sizing calculation uses a methodology based on identifying the mass flow rate needed through one RSV to maintain RV pressure below 110% of design pressure during the turbine trip transient. In its analysis, the applicant used three general methods of adding conservatism to the RSV capacity: (1) increase the heat addition to the RCS, (2) decrease the heat lost from the RCS, 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 **{{ }}**, which includes a **{{ }}** bias for instrument uncertainty. This higher reactor power generates a higher volumetric surge rate into the pressurizer to produce the most severe transient in accordance with SRP Section 5.2.2, Acceptance Criterion 3.B.i.

In addition, the analysis follows SRP Section 5.2.2, Acceptance Criterion 3.B.iii, associated with crediting the second safety grade signal to initiate reactor scram, by deactivating the first reactor trip signal. In the analysis the high pressurizer pressure signal is reached before the RSV opening setpoint is reached, however, this signal is deactivated. The reactor then trips on the second signal for high steam pressure. Delaying the reactor trip allows the pressurizer pressure to build higher, increasing the required RSV flow capacity.

In the analysis, the feedwater temperature was set to the upper bound of the expected operating range; thus, the elevated temperature reduced the ability for the secondary side to remove heat. The SG pressure is set to the lower bound of the expected operating range, which delays the reactor trip signal on high steam pressure, allowing the pressurizer pressure to increase higher prior to reactor trip.

Also, the average coolant temperature was set to the upper bound of the expected operating range. This hotter RCS coolant resulted in an increased coefficient of thermal expansion, which increased the surge rate into the pressurizer.

Finally, the heatup and pressurization during the transient tends to reduce core power because of reactivity feedback associated with an increase in the moderator temperature. However, the reactivity feedback was not credited in this analysis to ensure that core power was not reduced. This resulted in a greater rate of coolant expansion.

Therefore, the staff finds that the analysis and methodology used to determine the RSV flow capacity is acceptable because it complies with SRP Section 5.2.2, Acceptance Criterion 3.B, and therefore provides sufficient margin to account for uncertainties in the design and operation of the plant. In addition, the staff finds the RSV design satisfies the single failure criterion, as defined in SRP Section 5.2.2, Acceptance Criterion 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, which is the spurious actuation of the pressurizer heaters while operating below the LTOP enabling temperature of 143 degrees Celsius (°C)(290 degrees Fahrenheit (°F)). The applicant's analysis determined that the spurious actuation of the PZR heaters with a heat input of 800 kilowatts from the PZR heaters, with an additional input 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 the overpressurization is due to 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. 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 inservice testing and preventive maintenance program. FSAR Section 5.2.2.6, "Applicable Codes and Classification," states that the RVVs are designed in accordance with ASME BPV Code, Section III, Subarticle NB 3500, and function to satisfy the overpressure protection criteria described in ASME BPV Code, Section III, Article NB-7000. This is discussed in Section 3.9.6.4.6.1, "Emergency Core Cooling System Valves," of the SER.

Two 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. FSAR Section 5.2.2.8, states that direct position indication for each RVV is provided in the control room pursuant to the requirement of 10 CFR 50.34(f)(2)(xi).

5.2.2.5 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 FSAR, Table 1.8-1.

Table 5.2.2-1: NuScale COL Information Items for FSAR, Section 5.2.2

Item No.	Description	FSAR Section
COL Item 5.2-1	An applicant that references the NuScale Power Plant US460 standard design 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 RCPB and secondary system design contains adequate overpressure protection features, including low temperature overpressure protection features.	5.2.2

5.2.2.6 Conclusion

The staff reviewed the overpressure protection system design for full power and low temperature conditions, as described in FSAR Section 5.2.2, and TR-130877-P. The staff concludes that GDC 15 is satisfied because the RSVs' overpressure protection is sufficient to ensure that the 110 percent of the RCPB design pressure is not exceeded during normal power operations. GDC 31 is satisfied because RVVs' overpressure protection is adequate to ensure LTOP pressure limits are not exceeded during LTOP operations, such that the RCPB, when stressed, behaves in a nonbrittle way and a rapidly propagating fracture does not occur. In addition, the staff finds that each RSV and RVV provides direct position indication in the control room, in accordance with the requirement of 10 CFR 50.34(f)(2)(xi). Therefore, the staff concludes that RCPB overpressure protection design is acceptable as described in this section.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Introduction

The applicant, in the FSAR, Chapter 5, provided a description of the materials used in the RCPB in Section 5.2.3, "Reactor Coolant Pressure Boundary Materials." 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 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

FSAR, Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” provides a description of materials that make up the RCPB. There are no specific technical specification (TS) requirements associated with area of review. There are no technical reports (TRs) for this area of review. ITAAC related to this area of review are evaluated in Section 14 of this SER.

Material Specifications

The application identified the materials for the Class 1 components and supports that comprise the RCPB, including the Reactor Pressure Vessel (RPV), and SGs and are listed in Table 5.2-3. The material in Table 5.2-3 lists the grade or type, as applicable, of the ferritic low-alloy steels, austenitic stainless steels, and nickel-base alloys specified for the RCPB. The applicant stated that the materials associated with the RPV will be further discussed in Section 5.3, Reactor Vessel.

The applicant stated that the RCPB materials, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC Section III, Subsection NB requirements and the materials selected for fabrication of the RCPB comply with the requirements of ASME BPVC Section II.

Reactor Coolant Chemistry

The applicant stated that the RCS water chemistry is controlled to minimize corrosion of RCS surfaces and minimizes corrosion product transport during normal operation. These controls ensure the integrity of the RCPB materials, the integrity of the fuel cladding, fuel performance by limiting cladding corrosion, and the minimization of radiation fields. Accordingly, the plant maintains alkaline-reducing water chemistry during power operation. Routine sampling and analysis of the coolant verifies its chemical composition. The RCS water chemistry is controlled based on the Electric Power Research Institute (EPRI) TR 3002000505 “Pressurized Water Reactor Primary Water Chemistry Guidelines,” Revision 7, issued April 2014. The EPRI Guidelines minimize negative impacts of chemistry on material integrity, fuel rod corrosion, fuel design performance, and radiation fields, and are 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 (as permitted by the fuel manufacturer chemistry guidelines) to reduce radiation levels and reduce primary water stress-corrosion cracking (PWSCC) initiation rates.

Compatibility of Construction Materials with the Reactor Coolant

The applicant described controls for ferritic steel materials to justify the material’s sufficiency for a 60-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.

The applicant stated that the inside and outside of carbon and low-alloy steels have austenitic stainless-steel cladding, except for surfaces cladded with nickel chromium-iron (Ni-Cr-Fe) surfaces covered with stainless-steel sleeves or inserts, or the inside surfaces of SG tubesheet bores. The Ni-Cr-Fe cladding is deposited with Alloy 52/152. Weld overly cladding procedures will be qualified in accordance with ASME BPVC Section III, Subarticle NB-4300 and Section IX.

The applicant stated that the use of cobalt based alloys is minimized, and limits are established to minimize cobalt intrusion into the reactor coolant. Hard surfacing and wear resistant parts in the CRDMs use cobalt-based alloys. Low cobalt or cobalt-free alloys may be used for hard facing and wear resistant parts in contact with the reactor coolant if their wear and corrosion resistance are qualified to meet design requirements.

Fabrication and Processing of Ferritic Materials

The applicant stated that the fracture toughness properties of the ferritic RCPB components comply with the requirements of 10 CFR Part 50, Appendix G, "Fracture toughness requirements," and ASME BPVC Section III, Subarticle NB-2300. Discussion of the fracture toughness requirements of the RPV materials is in Section 5.3, Reactor Vessel.

Welding Control – Ferritic Materials

The applicant stated that welding procedures shall be qualified in accordance with the applicable requirements of ASME BPVC Section III, Subarticle NB-4300 and Section IX for welding of ferritic materials used for components of the RCPB. Preheat and interpass temperature controls for ferritic steels will meet the requirements of ASME Code, Section III, Division 1, Nonmandatory Appendix D. Controls on preheat for low-alloy steel forgings will be in accordance with RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

Procedure qualification records and welding procedure specifications used to support welding of low-alloy steel welds in the RCPB will be in accordance with ASME BPVC Section III, Subarticle NB-4300 and Section IX. Welding and welding operator qualifications will be in accordance with ASME BPVC Section III, Subarticle NB-4300 and ASME Section IX. Welding of ferritic steel under limited accessibility conditions will be controlled in accordance with RG 1.71, "Welder Qualification for Areas of Limited Accessibility."

The applicant stated that post-weld heat treatment (PWHT) temperature of the RPV low-alloy steel material is 1100°F to 1175°F.

The applicant stated that ferritic low-alloy and carbon steels used in the pressure retaining applications will 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 inside and outside surfaces of carbon and low-alloy steels have austenitic stainless-steel cladding, except for surfaces cladded with Ni-Cr-Fe, surfaces covered with stainless steel sleeves or inserts, or the inside surface of SG tubesheet bores. The applicant stated that the final thickness of corrosion-resistant weld overlay is 0.125 inch minimum on both the inside and outside surfaces except for sealing surfaces or surfaces requiring additional weld-buildup. The Ni-Cr-Fe cladding is deposited with Alloy 52/152.

Fabrication and Processing of Austenitic Stainless Steel

The applicant described controls for austenitic stainless-steel materials to justify the material's sufficiency for a 60-year period of operation. Controls for austenitic stainless-steel components primarily focused on preventing sensitization of unstabilized stainless-steels in the base materials and heat-affected zone of weldments.

The applicant stated that unstabilized austenitic stainless-steel base materials will be solution annealed, and water quenched (rapidly cooled), in accordance with the guidance in RG 1.44, “Control of the Processing and Use of Stainless Steel.” FSAR Section 5.3.1.2, "Special Processes Used for Manufacture and Fabrication of Components," states that austenitic stainless-steels heated above 800°F for PWHT or other purposes, other than locally by welding operations, for more than 60 minutes will be tested for nonsensitization in accordance with American Society for Testing and Materials (ASTM) A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless-Steels,” Practice A or E. Additionally, 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.

The applicant stated that unstabilized austenitic stainless-steel base metal and weld metal subjected to sensitizing temperatures after solution heat treatment have a carbon content no more than 0.03 weight percent. In addition, austenitic stainless-steel weld materials for the RCPB are analyzed for delta ferrite content and limited to 5 FN to 20 FN, except for E316, E316L,ER316, and ER316L where the ferrite content is limited to 5 FN to 16 FN, which exceeds the requirements found in RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” and ASME BPVC Section III, Paragraph NB-2433 requirements.

The applicant stated that fabricators of the RCPB components will avoid using cold worked austenitic stainless-steel to the extent practicable. If cold working is performed, then the yield strength of the base material shall not exceed 90,000 pounds per square inch (psi) as determined by the 0.2 percent offset method.

Fabrication and Processing of Nickel-Based Alloy Materials

The applicant stated that Alloy 690 base metal is used in RCPB components and structures along with Alloy 52/152 cladding and weld metals. The applicant stated that the Alloy 690 base metal used for SG tubes is used in the thermally treated condition to optimize resistance to intergranular corrosion.

Cleaning and Contamination Protection Procedures

The applicant stated that procedures provide cleanliness controls during the various phases of manufacture and installation, including final flushing. The suppliers implement a written cleanliness control plan before and during manufacturing and assembly of components, which continue until components are sealed for shipment.

For all RCPB components, the applicant committed to meeting the requirements in NQA-1, Subpart 2.1. This scope 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 Cleanliness Class B for all interior surfaces of the RCPB and Cleanliness Class C for all exterior surfaces. In addition, the applicant committed to meeting the handling, storage, and shipping requirements in ASME NQA-1, Subpart 2.2.

Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Based Alloys

The applicant stated that the nickel-based alloy components in the RCS are protected from PWSCC by using Alloy 690/152/52 in the nickel-based alloy applications, controlling chemistry, mechanical properties, and thermo-mechanical processing requirements to produce an optimum microstructure for resistance to intergranular corrosion, and limiting the sulfur content of nickel-based alloy base metal in contact with the RCS primary fluid to a maximum 0.02 weight percent.

Threaded Fasteners

The applicant stated that the threaded fasteners used in the RPV main closure flange, PZR heater bundle closures, RCS piping flanges, RVV flanges, RRV flanges, and RSV flanges are nickel-based Alloy 718. The threaded fastener materials conform to the applicable requirements of ASME BPVC Sections II and III and are selected for their compatibility with the borated water environment in the RCS and reactor pool water.

The applicant stated that Section 3.13, "Threaded Fasteners," provides further descriptions of the design of threaded fasteners for the RPV and pressure retaining components including design requirements for the use of Alloy 718 for the mitigation of SCC.

5.2.3.3 Regulatory Basis

The SRP 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:

- 10 CFR 50.55a. In accordance with Section III, Subsection NB of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPVC), the design, fabrication, construction, testing, and inspection of the RPV and pressure retaining components associated with the RCPB meet the applicable conditions promulgated in 10 CFR 50.55a(b). Section 5.2.1, "Compliance with Codes and Standards," provides additional details.

- GDC 1, "Quality Standards and Records," and GDC 30, "Quality of Reactor Coolant Pressure Boundary," 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, "Fracture Prevention of Reactor Coolant Pressure Boundary," as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB.

- Criterion XIII of 10 CFR Part 50, Appendix B. The Quality Assurance Program requires procedures for the control of on-site cleaning of RPV and the RCPB during construction.

- Appendix G to 10 CFR Part 50. The RPV ferritic pressure retaining and integrally attached materials meet applicable fracture toughness acceptance criteria. The design supports an exemption from the requirements of 10 CFR 50.60 which invokes compliance with 10 CFR Part 50, Appendix G. This is discussed further in Section 5.3.1.6 of the SDAA.

5.2.3.4 *Technical Evaluation*

Material Specifications

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," as required by GDC 30. The staff reviewed Table 5.2-3 of the application which includes materials and specifications associated with the RPV attachments and appurtenances. The table lists the grades or type, as applicable, of the ferritic low-alloy steels, austenitic stainless steels, and nickel-based alloys specified for the RCPB. The applicant stated that the discussion of materials associated with the RPV is in Section 5.3, Reactor Vessel.

The applicant has selected RCPB base 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 FSAR, Section 5.2.1.1, all RCPB components will meet the regulatory requirements found in 10 CFR 50.55a, including 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.

Reactor Coolant Chemistry

The RCS water chemistry is controlled to minimize corrosion of RCS surfaces and minimizes corrosion product transport during normal operation. These controls ensure the integrity of RCPB materials, the integrity of the fuel cladding, fuel performance by limiting cladding corrosion and the minimization of radiation fields. The applicant stated that the plant maintains alkaline-reducing water chemistry during power operation. Routine sampling and analysis of the coolant to verify its chemical composition is also performed.

Control of the quality of the chemicals and the makeup water added to the reactor coolant limits potential contamination. The staff reviewed the applicant's reactor coolant chemistry and determined it to be acceptable because it accomplishes the following. The applicant's reactor coolant chemistry parameters and impurity limitations are monitored during power operations and conform to the limits specified in the EPRI pressurized water reactor Primary Water Chemistry Guidelines, fuel vendor primary chemistry guidelines, and RG 1.44 limits as provided in Table 5.2.4 of the SDAA. In addition, zinc is added to the primary system (as permitted by the fuel manufacturer chemistry guidelines) to reduce radiation levels in plant maintenance areas and reduces PWSCC initiation rates.

The applicant specifies the performance frequency of sampling of water chemistry based on plant operating conditions and the EPRI water chemistry guidelines. Industry guidelines as described in EPRI Technical Report 3002000505, Pressurized Water Reactor Primary Water Chemistry Guidelines inform the water chemistry program. The program includes periodic monitoring and control of chemical additives and reactor coolant impurities as listed in Table 5.2-4 of the SDAA. Detailed procedures implement the program requirements for sampling and analysis frequencies and corrective actions for control of reactor water chemistry. Operating experience has shown that these guidelines are sufficient in monitoring and correcting water chemistry before material degradation can occur.

Compatibility of Materials with the Reactor Coolant

The staff reviewed the information in the SDAA 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 applicant stated that the RCPB ferritic low-alloy steels used in pressure retaining applications have austenitic stainless-steel cladding or Ni-Cr-Fe cladding on surfaces that are exposed to the reactor coolant. Low-alloy steel forgings have an average grain size of Number 5 or finer in accordance with the ASTM standards. The grain size of 5 is a “fine grain” microstructure which is consistent with the guidance of RG 1.43, “Control of Stainless-Steel Weld Cladding of Low-Alloy Steel Components.” The cladding of ferritic base materials receives a PWHT as required by ASME BPVC Section III, Subarticle NB-4622.

The applicant stated that the inside and outside surfaces of carbon and low-alloy steel have austenitic stainless-steel cladding, except for surfaces cladded with Ni-Cr-Fe, surfaces covered with stainless steel sleeves or inserts, or the inside surfaces of SG tubesheet bores. The final thickness of corrosion-resistant weld overlay is 0.125 inch minimum on both the inside and outside surfaces except for sealing surfaces or surfaces requiring additional weld-buildup. Ni-Cr-Fe cladding will be deposited with Alloy 52/152. Cladding procedures will be qualified in accordance with the applicable requirements of ASME BPVC Section III, Subarticle NB-4300 and Section IX. The minimum overlay thickness of 0.125 inch is sufficient for the functions that the cladding performs and is consistent with industry practice.

Fabrication and Processing of Ferritic Materials

The staff reviewed the SDAA information to assure that (1) brittle failure of ferritic materials is prevented, (2) the use of RGs is consistent with the SRP, and (3) PWHTs for ferritic materials are defined and meet the ASME Code.

The applicant stated that RCPB components do not contain ferritic steel tubular products.

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, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” and RG 1.71, “Welder Qualification for Areas of Limited Accessibility.” These RGs ensure that sufficient preheat will be used for low-alloy steel and that welding qualifications will consider limited accessibility conditions.

The applicant committed to using the requirements of ASME Code, Section III, Division 1, Nonmandatory Appendix D which will control the preheat and interpass temperatures for ferritic steels. The applicant stated that the PWHT of low-alloy steel RV materials will be 1100°F to 1175°F. Alternative PWHT time and temperatures specified in Subparagraph NB-4622.4(c) of ASME BPVC, Section III, Subsection NB will not be used.

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 cladding will meet the requirements of ASME Code, Section IX. The staff finds that 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 control on the fabrication and processing of ferritic materials, and the information in the SDAA meets the acceptance criteria in SRP 5.2.3.

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

The staff reviewed the information in the SDAA to assure that (1) corrosion is adequately considered in the design of the applicant’s plant, (2) the applicant provided welding process controls to prevent hot cracking and SCC, (3) thermal embrittlement of cast austenitic stainless-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 stated they use a three-step approach to SCC prevention of unstabilized austenitic stainless-steels. The applicant uses the guidelines of RG 1.44 to avoid sensitization and intergranular attack by (1) controlling material chemistry to prevent sensitization, (2) place limitations on cold work, and (3) control the reactor coolant chemistry. The applicant committed to the EPRI Primary Water Chemistry Guidelines which ensure that the potential for corrosion is minimized.

The applicant stated that they control oxygen, chlorides and fluorides in the reactor coolant during normal operation to further minimize the probability of SCC of unstabilized austenitic stainless steels. The use of hydrogen in the reactor coolant inhibits the presence of oxygen during operation. Gaseous argon may also be added to the reactor coolant, if required, to support primary to secondary leakage controls. The effectiveness of these controls has been demonstrated by test and operating experience.

Controls for 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 limited to 5 FN to 20 FN, except for E316, E316L, ER316, and ER316L where the ferrite content is limited to 5 FN to 16 FN which is more restrictive than the requirements in RG 1.31, “Control of Ferrite Content in Stainless-Steel Weld Metal,” and ASME Section III, Paragraph NB-2433. Weld filler metal with a delta ferrite content consistent with the RG is a sufficient means to prevent hot cracking and meets the SRP acceptance criteria.

The applicant committed to having welders and welding operators qualified in accordance with ASME BPVC Section IX and RG 1.71. Welding procedures will be qualified in accordance with ASME BPVC Section III, Subarticle NB-4300 and ASME BPVC Section IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding will be in accordance with ASME requirements.

The applicant stated that all austenitic stainless-steel materials are procured in the solution annealed and rapidly quenched state, as required by the 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 applicant also stated that 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 staff guidance in RG 1.44, "Control of the Processing and Use of Stainless-Steel."

The applicant stated that fabricators of RCPB components avoid, to the extent practicable, use of cold worked austenitic stainless-steel. The applicant stated that fabricators of RCPB components do not use cold worked austenitic stainless-steel with a material yield strength greater than 90,000psi, as determined by the 0.2 percent offset method. The controls on cold working are identical to SRP Section 5.2.3. RG 1.44 cites the SRP guidance on cold work as adequate to limit stress concentration in the material.

The applicant stated that the external surfaces of the upper RPV have austenitic stainless-steel cladding. External surfaces of the RCPB have no exposed ferritic materials, maintain compatibility with a borated water environment, and are resistant to general corrosion.

The applicant stated that for Austenitic Stainless Steel Tubular products, preservice nondestructive examinations performed on Class 1 tubular products to detect unacceptable defects comply with ASME BPVC Section III, Subparagraph NB-5280 and ASME BPVC Section XI examination requirements. For Class 1 piping welds requiring UT preservice examination, the surface finish and marking requirements of ASME BPVC Section III, Subparagraph NB-4424.2 are applicable.

Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Based Alloys

The licensee stated that PWSCC is avoided in nickel-based alloy components in the RCS by

- Using Alloy 690/152/52 in nickel-based alloy applications
- Controlling chemistry, mechanical properties, and thermo-mechanical processing requirements to produce an optimum microstructure for resistance to intergranular corrosion for nickel-based alloy base metal.
- Limiting the sulfur content of nickel-based alloy base metal in contact with RCS primary fluid to a maximum 0.02 weight percent.

The applicant stated that there have been no signs of PWSCC in Alloy 690 and its weld metal in operating PWRs and that Alloy 690 resists PWSCC initiation. The staff concurs that Alloy 690 and its weld metals provides reasonable assurance of the high resistance to PWSCC and the industry best practices are consistent with the SRP acceptance criteria and are therefore acceptable.

Cleaning of Components and Systems

Cleaning of RCPB components complies with ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications.” The applicant stated that procedures provide cleanliness controls during the various phases of manufacture and installation, including final flushing. The suppliers implement a written cleanliness control plan before and during manufacturing and assembly of components, which continues until components are sealed for shipment. The applicant also stated that the controls minimize the introduction of potentially harmful contaminants including chlorides, fluorides, and low melting point alloys on the surface of austenitic stainless-steel components. The applicant committed to removal of cleaning solutions, processing equipment, degreasing agents, and other foreign materials at any stage of processing before elevated temperature treatments shall be performed in accordance with RG 1.44. Acid pickling is avoided on stainless steel.

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 Cleanliness Class B for all interior surfaces of the RCPB and Cleanliness Class C for all exterior surfaces. The staff finds the cleanliness classes chosen by the applicant to be consistent with the guidance in NQA-1. In addition, the applicant committed to meeting the handling, storage, and shipping requirements in ASME NQA-1, Subpart 2.2.

5.2.3.5 *Combined License Information Items*

SER Table 5.2.3-1 lists COL information item numbers and descriptions from FSAR, Table 1.8-1.

Table 5.2.3-1 NuScale COL Information Items for FSAR, Section 5.2.3.2

Item No.	Description	FSAR Section
5.2-2	An applicant that references the NuScale Power Plant US460 standard design 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 Electric Power Research Institute Pressurized Water Reactor Primary Water Chemistry Guidelines.	5.2.3.2
5.2-3	An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Boric Acid Control Program that includes: Inspection elements to ensure the integrity of the RCPB components for subsequent service, monitoring of the containment atmosphere for evidence of reactor coolant system leakage, the	5.2.3.2

	type of visual or other nondestructive inspections to be performed, and the required inspection frequency.	

5.2.3.6 *Conclusion*

The staff concludes that the information provided by the applicant demonstrates that the RCPB materials of the design will meet the requirements of 10 CFR 50.55a; 10 CFR Part 50, Appendix A, GDC 1, 4, 14, 30 and 31; 10 CFR Part 50, Appendix B; and 10 CFR Part 50, Appendix G to address the topics noted in SER Section 5.2.3.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

5.2.4.1 Introduction

The applicant in the FSAR, Chapter 5 provided a description of the preservice inspection (PSI), ISI, and IST of ASME BPVC Class 1 pressure-retaining components used in Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing." The applicant stated that ISI and IST of ASME BPVC Class 1 pressure retaining components associated with the RCPB are in accordance with ASME Code, Section XI pursuant to 10 CFR 50.55a(g), including ASME BPVC, Section XI mandatory appendices.

5.2.4.2 Summary of Application

The application dated December 31, 2022, contains the following section for review:

FSAR: The applicant has provided description of its ISI program for Class 1 RCPB components in the FSAR, Section 5.2.4 as summarized below.

FSAR, Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," provides a description of the PSI, ISI and IST of ASME BPVC Class1 pressure-retaining components.

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.

The applicant stated that the initial ISI Program will incorporate the latest edition and addenda of the ASME BPVC approved in 10 CFR 50.55a(a) before initial fuel load, as specified in 10 CFR 50.55a(b). Inservice examination 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 BPVC incorporated by reference in 10 CFR 50.55a(a), subject to the conditions listed in 10 CFR 50.55a(b).

There are no TRs for this area of review. There are no specific TS requirements associated with this area of review. ITAAC related to this area of review are evaluated in Section 14 of this SER.

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.

5.2.4.4 Technical Evaluation

The staff reviewed the FSAR, Section 5.2.4 in accordance with SRP Section 5.2.4. The SDAA, Section 5.2.4 details the proposed requirements for the ISI and IST of the Class 1 components including the RPV 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 ISI program will incorporate the latest edition and addenda of ASME Code, Section XI approved in 10 CFR 50.55a(a).

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(a) 18 months before the start of the 120-month inspection interval (or 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).

The specific edition and addenda of the ASME BPVC used to determine the requirements for the inspection and testing plan for the initial and subsequent inspection intervals is provided in the ISI. The ASME BPVC includes requirements for system pressure tests and functional tests for active components. The applicant stated that the requirements for system pressure tests are in accordance with ASME Code, Section XI, Articles IWA-5000 and IWB-5000. These tests verify the pressure boundary integrity in conjunction with ISI. The applicant stated that Class 2 and 3 component examinations will be discussed in Section 6.6 of the SDAA.

5.2.4.4.1 Inservice Inspection and Testing Program

The applicant stated that the ISI and IST program are composed of the following:

- The component inspection program which includes non-destructive examination (NDE) of major components, piping system and support systems.

- The valve IST program, which monitors and detects degradation of selected valves.

- The hydrostatic testing program.

The RCPB is accessible and permits periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity pursuant to GDC 32, “Inspection of reactor coolant pressure boundary.” The NuScale US460 SDAA design allows inspection, testing and maintenance of the components located inside the RCPB and NPM. The equipment to be inspected is in an accessible position to minimize time and radiation exposure during refueling and maintenance outages. In addition, the applicant stated that plant technicians access components without being placed at risk for excessive dose or situations where excessive plates, shields, covers, or piping must be moved or removed to access components.

The staff finds that the SDAA sufficiently explains that the RCPB components will be designed to allow for inspection and testing of important areas and features pursuant to GDC 32.

The applicant stated that the inspection requirements and conditions of 10 CFR 50.55a, as stated in ASME Code, Section XI apply to Class 1 pressure-containing components and their supports. The applicant stated that the RCPB components subject to inspection as Class 1 components are Quality Group A and comply with the ASME BPVC as described in Section 5.2.1, “Compliance with Codes and Code Cases.” The applicant provided Figure 6.6-1 in the SDAA which identified the ASME Code, Section III, Class 1 boundary for the RCS piping and SG system. The applicant also stated that the ECCS valve actuators and actuator lines form a portion of the ASME Code, Section III, Class 1 boundary and are subject to ASME Code, Section XI testing. The staff finds this to be acceptable because it meets the applicable requirements of ASME Code, Section XI.

The applicant provided the following insights to the development of the inspection program:

- Identification of the appropriate ISI or IST requirements for the design (code version, overall inspections and test required).

- Identification of SSC, the subset inspections or test elements associated with SSC and those SSC that are subject to inspection and testing.

- Identification of appropriate ISI and IST requirements for each structure, system and component.

- Assessment of each inspection and test element.

- Development of a comprehensive ISI and IST plan.

The applicant’s ISI schedule and requirements for Class 1 systems and components are in accordance with the 2017 Edition of ASME Code, Section XI, Division 1.

The applicant’s ISI plan will define the examination program for the 10-year inspection interval. The ISI plan for Class 1 systems and components is developed in accordance with the 2017 Edition of ASME Code, Section XI, Division 1, Articles IWA-2400 and IWB-2400.

When performing surface examinations, the applicant stated that liquid penetrant or magnetic particle techniques will be used. Ultrasonic, radiographic or eddy current techniques will be used when performing volumetric examinations. Visual inspection techniques will be used when determining the surface conditions of components and any evidence of leakage for applicable components. The applicant stated that specific techniques, procedures and equipment, including any special techniques and equipment, are in accordance with the requirements of ASME Code, Section XI and conform to the ISI program. The staff finds this acceptable because it meets the applicable requirements and techniques described in ASME Code, Section XI.

The licensee stated that the visual, surface and volumetric examination techniques and procedures conform to the requirements of Article IWA-2200, and applicable portions of Table IWB-2500-1 of the 2017 Edition of ASME Code, Section XI, Division 1. The methods, procedures, and requirements for qualification of personnel performing ultrasonic examination conform to the requirements of the 2017 Edition of ASME BPVC, Section XI, Article IWA-2300. Qualification of personnel performing visual, liquid penetrant, magnetic particle, eddy current or radiographic examinations as a part of the PSI or ISI program are in accordance with the requirements of Article IWA-2300 of the 2017 Edition of ASME Code, Section XI, Division 1. The staff finds this acceptable because it meets the applicable requirements of ASME Code, Section XI.

The PSI program includes the examination categories in accordance with the 2017 Edition of ASME Code, Section XI, Division 1, Article IWB-2200.

The baseline examinations and data that are collected in accordance with related procedures will be provided in a report with tabulated results. The report will describe the scope of the inspection, procedures and equipment utilized, names and qualifications of personnel, and examination results including instrument calibration criteria in sufficient detail to provide reasonable assurance of repeatability for each examination.

Evaluation of examination results for Class 1 components is in accordance with Articles IWA-3000 and IWB-3000 of the 2017 ASME Code, Section XI. The repair of unacceptable indications conforms to the requirements of Article IWA-4000. Article IWB-3000 provides the criteria of establishing the need for repair or replacement. The staff finds this acceptable because it follows the requirements addressed in ASME Code, Section XI.

The applicant stated that Table IWB-2500-1 (B-P) and Articles IWA-5000 and IWB-5000 of the 2017 ASME Code, Section XI, provide the requirements for system leakage tests followed by a VT-2 examinations for RPV Class 1 pressure retaining boundary. Leakage monitoring continuously occurs from the Class 1 boundary into the CNV. This VT-2 exam is in accordance with ASME Code, Section XI, Paragraph IWA-5241(c). The applicant stated that Section 5.2.5 of the SDAA provides further details of RCPB Leakage Detection. The staff finds this acceptable because the applicant’s methodology for performing pressure testing of the Class 1 boundary and components meets the requirements of the ASME Code.

The applicant stated that the body to bonnet seals on the ECCS trip/reset actuator valve form a portion of the RCPB and require testing to RCS operating pressure before going into operation. The valve is in the reactor pool and therefore, there is no means to perform the required ASME Code, Section XI, Table IWB-2500-1 (B-P) VT-2 examination during the system pressure test. Therefore, a seal test is performed to meet the requirements of ASME Code, Section XI, Table IWB-2500-1 (B-P). The staff finds this acceptable because it meets the requirements of ASME Code, Section XI.

The design of the PZR spray lines, RPV high point degasification line and CVCS injection and discharge lines require the exterior nozzle-to-safe end welds and safe end-to-CITF welds to have a surface examination. The nozzle-to-safe end welds examination conform to the guidance in IWB-2500-1, Category B-F and safe end-to-CITF welds examination conform to the guidance IWB-2500-1, Category B-J of ASME Code, Section XI. The staff finds this acceptable because it meets the requirements of ASME Code, Section XI.

The licensee stated that the ASME Class 1 boundary valves (i.e., CIVs) are outside the NPM. The reduced inspection requirements for small primary system pipe welds associated with smaller than four-inch nominal pipe size piping are not applied to the welds between the CITFs and the CIVs because a break at one of these weld locations (break exclusion) would result in an RCPB leak outside the containment. Therefore, ASME Class 1 welds between the CITFs and the CIVs undergo a volumetric examination each interval in accordance with the requirements of IWB-2500 of ASME Code, Section XI. The staff finds this acceptable because it meets the requirements of ASME Code, Section XI, and that the augmented volumetric examination at the break exclusion zone verifies the assurance of the structural integrity of the welds during every 10-year inspection interval.

Flanges on the RPV have dual O-rings with a leak port tube between the O-rings to allow for leakage testing. The applicant stated that leakage testing is performed following installation of the O-rings each time they are removed to ensure the seals are seated as designed. The staff finds this methodology is acceptable for performing leakage testing after the installation of the O-rings.

5.2.4.4.2 Preservice Inspection and Testing Program

The applicant stated that the preservice examinations required by the design specifications and preservice documentation are in accordance with 2017 Edition ASME Code, Section III, Paragraph NB-5281. The volumetric and surface examinations shall conform to ASME Code Section III, Paragraph NB-5282. Components described in Paragraph NB-5283 of ASME Code, Section III, are exempt from PSI. The staff finds this acceptable because it meets the requirements of ASME Code, Section III.

As stated above, the welds between the CITFs and the CIVs require an augmented inspection because the consequences of a failure at this location would result in an RCPB leak outside the containment. Therefore, the consequences of a failure at this location would also include the CITF and CIV valve bodies. While pressure testing at 1.5 times the operating pressure of the valve body provides some assurance, it does not provide adequate assurance of the integrity of the valve bodies for the proposed application. Hydrostatic testing provides assurance that the valve body can withstand the pressure it is designed for and detects leaks from surface defects but does not detect defects within the valve body volume to ensure such defects in the valve body volume would not compromise its integrity during its operation. Therefore, the applicant stated that the forged valve bodies in the Class 1 CIV and CITF on the pressurizer spray, reactor pressure vessel high point degasification line, CVCS injection, and CVCS discharge will be subject to a one-time augmented volumetric examination during construction using the procedures and acceptance criteria of ASME Code, Section III, Paragraph NB-2540. The staff finds that the augmented volumetric examination provides assurance that defects in the forged valve bodies during fabrication are minimized so there is adequate assurance that the probability of rupture is extremely low per the guidance of Branch Technical Position 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside of Containment."

Surfaces of the RPV requiring examinations shall conform to the applicable requirements of ASME Code, Section III and XI. Welds requiring UT PSI shall have a surface finish that meets the requirements of ASME Code, Section III, Subsubparagraph NB-4424.2(a). The staff finds this acceptable because it meets the requirements of ASME Code, Section III.

The applicant stated that PSI for ASME Code Class 1 pressure boundary and attachment welds shall conform to ASME Code, Section III, Paragraph NB-5280 and ASME Code, Section XI, Subarticle IWB-2200. The preservice examinations include essentially 100 percent of the pressure boundary welds. The staff finds this acceptable because it meets the requirements of ASME Code Section III, and Section XI for PSI.

Preservice eddy examinations for SG tubing are in accordance with the applicable requirements of the EPRI Steam Generator Management Program guidelines and ASME Code, Section XI. The staff finds this acceptable because it meets the industry guidelines and ASME Code, Section XI for the PSI of SG tubes.

5.2.4.5 Combined License Information Items

SER Table 5.2.4-1 lists COL information item number and description from FSAR, Table 1.8-1.

Table 5.2.4-1 NuScale COL Information Item for FSAR, Section 5.2.4

Item No.	Description	FSAR Section
5.2.4	An applicant that references the NuScale Power Plant US460 standard design will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the American Society of Mechanical Engineers Operations and Maintenance Code and will establish implementation milestones. If applicable, an applicant that references the NuScale Power Plant US460 standard design will identify the implementation milestone for the augmented inservice inspection program. The 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.	5.2.4

5.2.4.6 Conclusion

The staff concludes that the information provided by the applicant demonstrates that the design of the RCPB incorporates provisions for access to enable the performance of ISI examinations in accordance with 10 CFR 50.55a(g) and ASME Code, Section XI. The final ISI program is required to meet the latest ASME Code, Section XI Edition 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 inspections of the areas of the RCPB that are not readily accessible to inspection personnel. The final ISI program will consist of a PSI and ISI program. The periodic inspections and pressure testing of the 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

SDAA Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection,” describes the RCS leakage detection systems, which are designed to detect and, to the extent practicable, identify the source of reactor coolant leakage.

5.2.5.2 Summary of Application

The applicant discussed RCPB leakage detection in SDAA Section 5.2.5 and Section 9.3.6, “Containment Evacuation System”. SDAA Section 5.2.5, provides information on the RCS leakage monitoring by using containment evacuation system (CES) condensate and containment pressure. Section 9.3.6 describes leakage detection methodology and important components for the systems description. SDAA Figure 9.3.6-1, “Containment Evacuation System Diagram,” shows a simplified diagram.

ITAAC: SDAA (Part 8) Table 2.3-1, “Containment Evacuation System Inspection, Test, Analyses, and Acceptance Criteria,” provides the ITAAC for RCS leakage detection. These ITAAC are evaluated in Section 14.3 of this SER.

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 (ML15355A505), 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 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

5.2.5.4 Technical Evaluation

SDAA Section 5.2.5.1, “Leakage Detection and Monitoring,” states that there are three methods of leakage monitoring to detect and identify leakage into containment. Two of these methods, CNV pressure and CES sample vessel level detection, can quantify leakage into the CNV. Additional monitoring using radiation monitoring assist in assessing the source of leakage. The applicant’s design uses an approach for the RCPB leakage monitoring that is different from other current PWR designs.

The staff reviewed the RCPB leakage detection systems described in SDAA in accordance with Design Specific Review Standard (DSRS) (ML15355A295) Section 5.2.5. As indicated in DSRS Section 5.2.5, the alternative leakage detection systems should be reviewed in sufficient detail according to RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System

Leakage," Revision 1, issued May 2008. GDC 30 requires that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. DSRS Section 5.2.5 states that, for GDC 30, the review of RCPB leakage detection is based on meeting 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.

SDAA Section 5.2.5.1, "Leakage Detection and Monitoring" states the following:

> Regulatory Positions C.2.1 and C.2.2 of RG 1.45 are satisfied because leakage into the CNV from unidentified sources can be detected, monitored, and quantified for flow rates greater than or equal to 0.05 gpm using CNV pressure or CES sample tank level timing, and 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. Radiation detectors in the CES condenser vent line and sample tank provide an early indication of RCS leakage. They provide the ability to discern changes in CES process radiation levels and assist the operator in assessing the source of leakage into the CNV. Section 11.5, Process and Effluent Radiation Monitoring Instrumentation and Sampling System, describes radiation monitoring for the CES.

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

In SDAA Section 9.3.6.2.3, the applicant provided 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 addition, 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 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.

Further, 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 SDAA Section 5.2.5, Section 9.3.6, and the applicant's RAI-10135-R1 submittal, 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, of sensitivity of 0.05 gpm and leakage detection response time of 1 gpm leakage detection within 1 hour, respectively.

5.2.5.4.2 Leakage Detection Systems

GDC 30 states that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. 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.

In addition, RG 1.45, Section B, regarding 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).

In addition to the CNV pressure and CES tank level discussed above, SDAA Section 5.2.5, discusses CES gaseous discharge process radiation monitoring, which can be used to identify the source of leakage with respect to leakage from the primary or secondary side. 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).

SDAA Section 9.3.6.2.2 indicates grab samples and sample vessel radiation instrumentation provide an indication of the leakage source. Additionally, the licensee will regularly perform an RCS inventory balance to determine RCS leakage quantity as part of TS surveillance requirement (SR) 3.4.5.1

Based on the above, the staff finds that the applicant has adequately addressed RG 1.45, Regulatory Position C.2.3, on supplemental leakage detection systems of radiation monitoring grab sample analysis and the inventory mass balance method.

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:

> 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.)

SDAA Sections 5.2.5 and 9.3.6 address the alarms in the main control room. LCO 3.4.7 requires leakage detection instrumentation to remain operable and perform routine surveillance on leakage instrumentation channels. COL Item 5.2-5 provides information for the COL applicant to address the procedures to monitor the prolonged low-level leakage. Based on the above, the NRC staff finds that the applicant has adequately addressed RG 1.45, Regulatory Position C.3.3, on leakage monitoring systems in the main control room.

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 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.”

SDAA Section 9.3.6.3 states the following CES design:

> The CES inlet pressure instrumentation and its connecting piping, up to and including isolation valves are 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.

Table 9.3.6-1 “Classification of Structures, Systems, and Components” defines the CES pressure instrumentation (CE vacuum pump suction pressure transmitter) as Seismic Class I.

The NRC staff found CES inlet pressure instrumentation acceptable in meeting GDC 2 on seismic qualification because the design is consistent with RG 1.45, Regulatory Position C.2.4. Therefore, GDC 2 is satisfied with respect to the leakage monitoring systems withstanding the effects of seismic events.

5.2.5.4.5 Identified and Unidentified Leakage

As noted in DSRS Section 5.2.5, the reviewer verifies whether the provisions for collecting, detecting, and monitoring unidentified leakage are separate from those for identified leakage.

However, if separation is not practicable for the applicant, 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.

SDAA 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 of treating both identified and unidentified leakage of the RCS leakage as unidentified leakage is conservative because unidentified leakage has more stringent criteria. This is consistent with 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 5.2-5 to address the concern of the prolonged low-level RCS leakage:

> An applicant that references the NuScale Power Plant US460 standard design 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 RCPB leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.

The staff found the information in COL Item 5.2-5 to be adequate for the COL applicant to address the issue of prolonged low-level RCS leakage because it is consistent with RG 1.45 in managing the prolonged low-level RCS leakage.

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. 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 SDAA Section 5.2.5.2, “Reactor Pressure Vessel Flange Leak-Off Monitoring”; Section 5.2.5.3, “Reactor Safety Valve and Emergency Core Cooling System Valve Leakage Monitoring”; Section 5.2.5.4, “Chemical and Volume Control System Intersystem Leakage Monitoring”; Section 5.2.5.5, “Reactor Component Cooling Water System Leakage Monitoring”; and Section 5.2.5.6, “Primary to Secondary Leakage Monitoring.” In these sections, the applicant described systems that are connected to the RCS and identified potential RCS leakage paths and instrumentation provided to monitor intersystem leakage.

Based on review of the above sections, the staff has determined that the intersystem leakage is acceptable because the proposed approach is consistent with 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

SDAA Table 14.2-36, “Containment Evacuation System,” Test #36, addresses the initial test program (ITP) for RCS leakage detection monitoring capability

5.2.5.6 Technical Specifications

SDAA (Part 4), “Technical Specifications,” provides plant requirements in TS 3.4.5, “RCS Operational Leakage,” to verify RCS operational leakage within specified limits and in TS 3.4.7, “RCS Leakage Detection Instrumentation,” to address the RCS leakage detection instrumentation requirement.

5.2.5.7 Combined License Information Items

COL information item number and description from FSAR Section 5.2.5 and Table 1.8-1.

Table 5.2.5-1 NuScale COL Information Item for FSAR Section 5.2.5

Item No.	Description	FSAR Section
5.2-5	An applicant that references the NuScale Power Plant US460 standard design 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 RCPB 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 adequate, based on the review in SER Section 5.2.5.4.6.

5.2.5.8 Conclusion

Based on the above, the staff concludes that the design of the RCPB leakage detection system follows the guidelines of DSRS Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection,” and RG 1.45 and, therefore, meets the requirements of GDC 2 and 30.

5.3 Reactor Vessel

This section addresses reactor vessel materials; pressure-temperature limits, pressurized thermal shock, and upper-shelf energy data and analyses; and reactor vessel integrity. The contents of this section and the staff’s review are detailed below in 5.3.1 through 5.3.3. In addition, technical reports TR-130877-P, Revision 1, “Pressure and Temperature Limits Methodology” and TR-130721-P, Revision 0, “Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel” are incorporated by reference into Section 5.3 as described in Table 1.6-2, “NuScale Technical Reports.”

For the purposes of this section, the NRC staff will refer to the lower RPV and upper RPV. The lower and upper RPV together form the inner pressure-retaining portion of the NPM enclosing the fuel, steam generator tubes, and pressurizer space. This is illustrated in FSAR, Figure 5.1-1 as the portion within the containment vessel (but not including the containment vessel and containment vessel appurtenances). The boundary between the lower and upper RPV is a flange just above the fuel height as shown in FSAR, Figure 5.3-1. Together, the lower and upper RPV constitute a barrier to the release of fission products.

5.3.1 Reactor Vessel Materials

5.3.1.1 Introduction

This section addresses RPV 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 RPV fasteners. This section of the FSAR should contain pertinent data in sufficient detail to provide assurance that the materials (including weld materials), fabrication methods, and inspection techniques used for the RPV, 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

FSAR: FSAR, Section 5.3.1, “Reactor Vessel Materials,” describes the materials used in the RPV, as summarized, in part, below.

FSAR, Section 5.3.1 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 RPV fasteners.

The RPV is fabricated in accordance with ASME Code, Section III requirements as delineated in FSAR, Sections 5.2.3, “Reactor Coolant Pressure Boundary Materials,” and 5.3.1.1, “Material

Specifications." Low-alloy steel is used to fabricate the upper RPV, clad primarily with austenitic stainless steel (on both the exterior and interior of the RPV). Austenitic stainless steel is used to fabricate the lower RPV. The shell portions of the RPV surrounding the reactor core, pressurizer, and SGs are forged to minimize welding.

The NDE is performed consistent with ASME Code, Section V, as modified by ASME Code, Section III, with several additional requirements. This is detailed further in FSAR, Sections 5.2.3 and 5.3.1.

Welding of ferritic steels is conducted in accordance with ASME Code, Section III, Subarticle NB-4300, and Section IX, as elaborated in FSAR, 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 IX, as elaborated in FSAR, 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 upper RPV comply with the requirements of 10 CFR Part 50, Appendix G, and ASME Code, Section III, Subarticle NB-2300. The lower RPV is made from austenitic stainless steel. Exemption from the normally associated 10 CFR 50.60, and 10 CFR Part 50, Appendix G requirements is requested for the lower RPV in SDAA Part 7, "Exemptions", Section 6 and discussed in SE Section 5.3.4.

A material surveillance program is not included in the SDAA as the lower RPV is made from austenitic stainless steel, and the upper RPV is projected to accrue neutron fluence of less than 1×10^{17} n/cm^2 (E > 1 MeV). An explanation of why 10 CFR 50.60 and 10 CFR Part 50, Appendix H do not apply to the lower RPV is provided; consequently, no material surveillance program is proposed. Exemption from the surveillance program requirements in 10 CFR 50.60 and 10 CFR Part 50, Appendix H is requested in SDAA Part 7, Section 6 and discussed in SE Section 5.3.4.

The RPV main flange closure bolting is to be fabricated from SB-637 UNS N07718 to prevent general corrosion during immersion. FSAR, Section 3.13, "Threaded Fasteners," contains further information on fasteners.

ITAAC: SDAA Part 8, Table 2.1-1, contains the ITAAC associated with FSAR, Section 5.3.1. Table 2.1-1, Items 1 and 2, indicate that inspections will be performed on as-built component data reports to conclude that NPM ASME Code Class 1, 2, and 3 components are constructed in accordance with the requirements in ASME Code, Section III.

Technical Specifications: The are no TS sections associated with this area of review

Technical Reports: There are two TRs supporting this area of review. Both reports are listed as incorporated by reference in FSAR, Table 1.6-2.

Pressure and Temperature Limits Methodology, TR-130877, Revision 1, Oct 2024;

And;

Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel, TR-130721, Revision 0, Dec 2022.

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 FSAR, Section 5.3.1, the staff has determined that the use of ASME Code, Section III, appropriately supports GDC 1 without supplement.

The SDAA includes an Exemption request from the requirements of 10 CFR 50.60, and consequently also from 10 CFR Part 50, Appendices G and H in SDAA Part 7, Section 6. Those exemptions, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption. The staff's review of this request is documented in Section 5.3.4 of this SE.

5.3.1.4 Technical Evaluation

The staff reviewed FSAR, Section 5.3.1, using SRP Section 5.3.1. The ASME Code of record for the applicant is the 2017 Edition. Subject to the conditions of 10 CFR 50.55a, ASME Code, Section III, Subsection NB, contains the construction requirements for the RPV. The applicant noted that the RPV will be 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 RPV 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.

FSAR, Section 5.3.1.1, states that FSAR, Table 5.2-3, lists the materials used in the RPV. Ferrous materials used to fabricate the RPV include SA-508, Grade 3, Class 2 (flange, shells including PZR baffle plate, upper head, steam plenum and feed plenum access ports); and an assortment of Type 304, 316, 410, and similar Type components. The lower vessel (lower head, shell, and flange), as well as several other components, are made of SA-965 FXM-19 austenitic stainless steel and compatible materials. Ni-Cr-Fe Alloy 690 is used for resistance temperature detector interface, flow sensor interface, pressure taps, CNV-RPV lateral support lugs, RPV-CNV support ledge assemblies, support ledge assembly shims (vertical and horizontal), pressurizer heater bundle flange, and some bolting. 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 and are appropriate for the relevant service conditions.

FSAR, Table 5.2-3 also specifies the weld materials used in the RPV, and FSAR, Section 5.3.1.2, states that welding materials for the RPV conform to ASME Code, Section II, and ASME Code, Section III. The weld materials used to fabricate the RPV 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 and are appropriate for the relevant service conditions.

The applicant described all RCPB materials as complying with GDC 1, 4, 14 and 32; and 10 CFR Part 50, Appendix G (with consideration to the related Exemption, reviewed below). Specifically, the applicant noted that the RPV is fabricated according to ASME Code, Section III, Article NB-4000. The reactor vessel internals are fabricated in accordance with ASME Code, Section III, Article NG-4000 (SG lower supports and SG tube supports are described as ASME Code, Section III, Subsection NG components in SDAA Part 2, Section 5.4.1.5). The RPV supports and CRDM seismic support structure is fabricated in accordance with ASME Code, Section III, Article NF-4000. The staff confirmed that these components will be fabricated according to the appropriate and approved ASME Code, Section III, Articles consistent with the pertinent guidance in the SRP.

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 RPV 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.

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

FSAR, 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 A262-15, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless-Steels."

Welding practices and material compositions are controlled to manage sensitization. Unstabilized Type 3XX austenitic stainless steel and corresponding weld filler metals have a maximum carbon content of 0.03 weight percent to prevent undue 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 the information above and found it to be acceptable because it meets the requirements of ASME Code, Section III, and is consistent with the recommendations of RGs 1.43, Revision 1, and 1.44, Revision 1.

5.3.1.4.3 Special Methods for Nondestructive Examination

FSAR, Section 5.3.1.3, states that RPV pressure-retaining and integrally attached materials examinations are to meet the requirements specified in ASME Code, Section III. Specifically, the applicant stated that examinations will meet the requirements of ASME Code, Section V, except as modified by ASME Code, Section III and additional requirements in the SDAA. The applicant provided further details concerning the preservice examinations. The staff discusses this topic as it pertains to the RCPB in more detail in its review of FSAR, Section 5.2.3. The staff reviewed the information in FSAR, Section 5.3.1.3, for accuracy and consistency with FSAR, Section 5.2.3, 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

FSAR, Section 5.3.1.4, references FSAR, Section 5.2.3.3, for welding of ferritic steel components in the RPV, and FSAR, Section 5.2.3.4, for welding of austenitic stainless-steel components in the RPV. FSAR, Section 5.3.1.4, also notes that electroslag welding is not used for joining materials; rather, it is only used for cladding low-alloy steel in compliance with RG 1.43 ("Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"). The staff reviewed FSAR, Sections 5.2.3.3 and 5.2.3.4, for welding for ferritic and austenitic steels, respectively, as described in the associated sections of this SE. The staff reviewed the use of welding for cladding low-alloy steel and found it acceptable, as it is consistent with the recommendations of RG 1.43 as documented in the review of FSAR, Section 5.2.3.4.

FSAR, Section 5.3.1.4, references FSAR, Sections 4.5.2, "Reactor Internals and Core Support Structure Materials," and 4.5.1, "Control Rod Drive System Structural Materials" concerning tools for abrasive work and use of cold worked austenitic stainless steel. Specifically, FSAR, Subsections 4.5.2.4, "Fabrication and Processing of Austenitic Stainless-Steel Components," and Section 4.5.1.1, “Materials Specifications,” provide detailed information pertinent to these topics. The staff has documented its review of these topics in the associated SER sections.

5.3.1.4.5 Fracture Toughness

FSAR, Section 5.3.1.5, “Fracture Toughness,” describes how the fracture toughness requirements of 10 CFR Part 50, Appendix G, are met for RPV beltline materials in the applicant’s 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 with the exception of the lower RPV due to requested Exemptions 6 and 15 in SDAA Part 7.

The staff verified that application of ASME Code, Section XI, Appendix G analysis in concert with testing required by ASME Code, Section III, Subarticle NB-2300 will ensure that the construction materials subjected to design stresses and conditions will have sufficient fracture toughness to appropriately withstand these stresses and conditions. Specifically, the toughness requirements of NB-2331 testing will ensure that materials that fail testing requirements will not be used. In concert, the ASME Code, Section XI, Appendix G analysis provides allowable loadings for materials with properties compliant with NB-2331 requirements based on postulated flaws and linear elastic fracture mechanics.

The staff reviewed the information above and finds that the applicant has addressed fracture toughness appropriately, specifically through reference to 10 CFR Appendix G, and ASME Code, Section III, Paragraph NB-2311 testing requirements, as documented further in TR-130877-P, Revision 1, “Pressure and Temperature Limits Methodology,” and the Exemption requests 6 and 15 in SDAA Part 7, because the RPV materials were appropriately analyzed according to 10 CFR Part 50, Appendix G (as it applies), and ASME Code, Section III, NB-2311, and ASME Code, Section XI, Appendix G.

5.3.1.4.6 Material Surveillance

FSAR, Section 5.3.1.6 provided discussion concerning the inapplicability of the 10 CFR Part 50, Appendix H requirements pertaining to ferritic materials as the lower RPV materials are austenitic stainless steel and not ferritic low alloy steel. The upper RPV, while constructed primarily from ferritic low alloy steel, is not projected to receive sufficient neutron fluence to generate measurable embrittlement. The applicant provided further information in SDAA Part 7, Exemptions 6 and 15. The staff’s review of these Exemptions is discussed in SE Sections 5.3.4 and 5.3.5.

The FSAR incorporates by reference technical report, “Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel,” (see FSAR, Table 1.6-2) that provides the technical basis for the performance of the proposed austenitic stainless steel lower RPV under the design radiation conditions. The review of this technical report is documented in SE Section 5.3.1.4.8. Based on the above, the staff finds that there is no need for a material surveillance program because austenitic stainless steel must receive substantially more radiation than proposed to undergo significant radiation effects threatening the integrity of the lower RPV; and the upper RPV is not subject to sufficient neutron radiation to require surveillance.

Based on the review described above, the technical report, and Exemptions 6 and 15, the staff finds that there is no need for materials surveillance of the RPV because RPV materials will not receive sufficient neutron radiation damage during the proposed design life to justify requiring an Appendix H material surveillance program.

5.3.1.4.7 Reactor Vessel Fasteners

FSAR, Section 5.3.1.7, "Reactor Vessel Fasteners," states that the bolting material for the RPV closure flange is fabricated from SB-637. 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 evaluated this bolting material as part of its evaluation of FSAR, Section 3.13.

FSAR, section 5.3.1.7 provides a detailed discussion of the configuration, design, and function of the RPV threaded fasteners, threaded inserts, and the function thereof. The staff reviewed this description and finds that the configuration, design, and function of the RPV threaded fasteners, threaded inserts, and their function are acceptable.

FSAR, 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, paragraph NB-1132.1(c)(2), and ASME Code, Section III, paragraph NB-4435.

5.3.1.4.8 TR-130721 Use of Austenitic Steel for NPM Lower Reactor Vessel

TR-130721-NP, "Use of Austenitic Stainless Steel for NPM Lower Reactor Vessel," (hereafter "the report") presents a discussion of the acceptability of SA-965 Grade FXM-19 austenitic stainless steel base metal and E/ER209 or E/ER240 weld filler metal for use in the NPM lower RPV. Traditionally RPVs are made of ferritic material clad with austenitic stainless steel on the interior. The report notes that several applicable NRC regulations, specifically 10 CFR 50.60; 10 CFR 50.61; and the associated 10 CFR Part 50 Appendices G and H specifically refer to or presume the use of ferritic materials. These regulations traditionally support the NRC findings regarding GDCs 14, 31, and 32. The licensee states that consequently the subject lower RPV cannot be evaluated using these regulations. The purpose of the report is to summarize known data relating to the austenitic base and weld metals used in the lower RPV in support of the associated SDAA Exemptions from 10 CFR 50.60 and 50.61. Those exemptions, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption

The NRC staff reviewed the presented information to verify the inapplicability of the associated regulations, the relevant material properties of the proposed materials, and appropriateness of using Type 3XX results to inform the review of SA-965 Grade FXM-19. The staff noted that austenitic stainless steel is not a novel material for nuclear applications, being used as a traditional RPV cladding material and for reactor vessel internals. Similarly, the staff noted that austenitic stainless steel pressure vessels are not a novel use of the material and have been used successfully in many non-RPV applications. The applicant stated that austenitic stainless steel was selected for its superior ductility; and lower susceptibility to neutron and thermal embrittlement as compared to traditional ferritic materials. Austenitic stainless steels are known for their superior toughness, ductility, lack of ductile-to-brittle transition temperature (a core topic regulated by 10 CFR 50.60 and 50.61), and superior corrosion resistance.

The licensee presented the chemical composition of SA-965 Grade FXM-19 and the proposed weld filler materials in report Table 3-1 and tensile properties in report Table 3-2. The staff confirmed that based on chemistry and tensile properties these materials are appropriately compatible. Other aspects of welding and weld control are discussed in the preceding sections of this subsection.

Comparability of Type 3XX Results to SA-965 Grade FXM-19

The staff performed an independent literature review concerning the applicability of Type 3XX materials testing to the subject properties of SA-965 Grade FXM-19. Specifically, the staff confirmed that embrittlement, toughness trends, material properties, and general aging characteristics were sufficiently similar to support the use of Type 3XX results in the technical report. The staff found that while SA-965 Grade FXM-19 contains additional alloying elements relative to Type 3XX specifications, these aspects would not substantially affect the use of Type 3XX data presented in the technical report. This is in part due to the general differences between traditional low alloy ferritic steels and austenitic stainless steels being more significant to the review of this technical report than specific differences between SA-965, FXM-19 and Type 3XX materials.

Neutron Embrittlement

Results of a literature review are presented in report Section 3.3. Results from several comparable studies and materials were presented such as materials irradiated in the RBT6 reactor (a Russian test reactor) and Idaho National Laboratory Advanced Test Reactor (ATR). In addition, information was drawn from studies of 3XX austenitic stainless steels which have been used extensively in high radiation applications such as reactor vessel internals. Results included reports prepared for the EPRI, and on behalf of the NRC by Argonne National Laboratories (e.g. NUREG/CR-7027, “Degradation of LWR Core Internal Materials Due to Neutron Irradiation,” ADAMS Accession No. ML110100377.) The applicant identified through literature review that four light-water reactors not regulated by the NRC had or have RPVs made from austenitic stainless steel. These included US military reactors and a test reactor (the ATR).

The staff reviewed the presented literature review results and independently confirmed them through a staff conducted literature review. Specifically, the staff evaluated the impact of neutron fluence, verifying the licensee’s presentation of ductility effects. The staff found that the fluences necessary to impact ductility are substantially larger than the design life fluences proposed by the applicant. Even at fluences sufficient to affect the ductility of austenitic stainless steel it would retain substantial ductility. Similarly, the temperature effects of embrittlement (e.g. the temperature at which embrittlement studies were conducted) did not substantively affect this conclusion. For example, Fig. 3-4 of the report illustrates that the family of austenitic stainless steels commonly used in nuclear applications (304, 316, etc.) begin to exhibit reductions in J_{Ic} (a measure of toughness) near the 1 dpa level regardless of irradiation temperature. This 1 dpa level translates to roughly 7 x 10^{21} n/cm^2 (E > 1 MeV), or approximately 20+ times the projected design fluence of the inner RPV surface. Fig. 3-5 of the report illustrates results for mill-annealed FXM-19 consistent with the results shown in Fig. 3-4 of the report.

Based on the materials testing results presented by the applicant and the independent literature review conducted by the staff, the NRC finds that FXM-19 austenitic stainless steel will retain sufficient ductility at design fluences such that the monitoring and regulatory requirements related to neutron embrittlement for ferritic RPVs do not meaningfully apply to this material.

Thermal Embrittlement

Wrought austenitic stainless steel is not generally subject to thermal embrittlement at PWR operating temperatures. However, substantial work has been done concerning cast austenitic stainless steels which can undergo thermal embrittlement at PWR operating temperatures. Consequently, criteria for 3XX austenitic stainless steels have been developed to evaluate austenitic stainless-steel susceptibility to thermal embrittlement. The applicant cited criteria from EPRI Report MRP-175, “Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,” 2005 revision. These included fluences roughly 10 times the peak design fluences for the lower RPV. In addition, criteria concerning the molybdenum and ferrite contents were compared to the lower RPV materials. Based on this comparison, the proposed welding materials should be controlled below 16 FN to avoid synergistic effects between neutron and thermal embrittlement. The staff confirmed that the 16 FN limit is acceptable because it is consistent with up-to-date guidance used in the management of cast austenitic stainless steels in the operating fleet. The staff further confirmed that these ferrite limits are appropriately noted in FSAR Section 5.2.3, specifically in 5.2.3.4.1 where all austenitic stainless steel weld materials for the RCPB are analyzed for delta ferrite content and limited between 5 FN to 16 FN for E316, E316L, ER316, and ER316L, and 5 FN to 20 FN otherwise.

Based on the above the staff find that, when ferrite content is controlled as noted above, the selection of FXM-19 and the associated weld metals do not require further consideration of thermal embrittlement.

<u>Regulatory Evaluation – 10 CFR 50.60 and Appendix H</u>

The report states that no materials surveillance program would be necessary based partly on the above, and because austenitic stainless steels do not exhibit a ductile-to-brittle transition as traditional ferritic materials do. Consequently, the requirements of 10 CFR 50.60 to comply with 10 CFR 50 Appendix H do not directly apply to the lower RPV. Furthermore, no analogous program would be necessary for the subject materials. Based on the discussion above, the staff concurs that no surveillance program would be necessary because of the superior ductility of the subject materials, embrittlement trends of said materials, ferrite limits imposed by the applicant, and the large margin between the proposed design fluence and impactful fluences for the subject materials.

The staff recognize, however, that application of ASME Code, Appendix G is still necessary for the remaining ferritic portions of the RPV. This aspect of review is addressed in the review of FSAR Reference 5.3-6, the “Pressure-Temperature Limits Methodology.” Pertinent COL Item 5.3-1 supports demonstrating compliance with ASME Code, Appendix G, and is included below. The staff review of COL Item 5.3-1 is described further in Section 5.3.2 of this SER.

<u>Regulatory Evaluation – 10 CFR 50.61</u>

The applicant states that the methods employed in 10 CFR 50.61 to protect against pressurized thermal shock events are based on ferritic RT_{NDT} values (e.g. the ductile-to-brittle transition temperature). As the subject material does not have a ductile-to-brittle temperature and will not credibly reduce in toughness at the design fluence or temperatures, the methodology in 10 CFR 50.61 does not appear to apply. Based on the discussion above, the staff find that the methodology in 10 CFR 50.61, explicitly noted to be for ferritic materials, does not meaningfully apply to the lower RPV because the proposed materials do not exhibit the material properties of concern to the regulation as written.

The management of 10 CFR 50.61 as it relates to the upper RPV is discussed in FSAR, Section 5.3.2.3 and the associated staff review of that section below.

5.3.1.4.9 Combined License Information Items

SER Table 5.3.1-1 lists the COL information item number and description from FSAR, Table 1.8-1.

Table 5.3.1-1 NuScale COL Information Item for FSAR, Section 5.3.1

Item No.	Description	FSAR Section
5.3-1	An applicant that references the NuScale Power Plant US460 standard design will choose the final transients to generate the reactor vessel pressure-temperature limits report and limiting conditions for operation. An applicant that references the NuScale Power Plant US460 standard design will confirm that the design geometries, final transients, and material properties of the reactor pressure vessel are bounded by (or identical to) those used in the Pressure and Temperature Limits Methodology to confirm that the example curves in the Standard Design Approval Application are applicable. Operating procedures will ensure that pressure-temperature limits for the as-built reactor are not exceeded. These procedures will be based on the limits defined in the pressure temperature limits report and material properties of the as-built reactor vessel.	5.3.2.2

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 directs a COL applicant to provide site-specific information, including an implementation schedule for operational programs.

This COL item is necessary as the pressure-temperature limits presented in the SDAA are calculated using preliminary design information and do not include all aspects of the final design that should be verified, validated, or otherwise considered in the final generation of pressure-temperature limits.

The list above adequately describes actions related to the RPV materials considerations that remain for the COL applicant for FSAR, Section 5.3.1.

5.3.1.5 Inspections, Tests, Analyses, and Acceptance Criteria

ITAAC: The applicant gave the ITAAC associated with FSAR, Section 5.3.1, in FSAR, Table 2.1-2. SDAA Part 8, Table 2.1-2, Item 2, indicates that as-built components conform to the rules of construction of ASME Code, Section III, and that reports exist that conclude, “The NuScale Power Module ASME Code Class 1 and 2 components conform to the rules of construction of ASME Code.”

This ITAAC is evaluated in Section 14.3.3 of this SER.

5.3.1.6 Conclusion

The staff concludes that the applicant’s RPV 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 (as applicable and consistent with the Exemption review) 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, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

5.3.2.1 Introduction

Neutron radiation is known to cause embrittlement, or a reduction in ductility, in ferritic steels typically used for RVs. Neutron induced embrittlement 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 Upper Shelf Energy (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 RPV. An additional requirement placed on (ferritic) beltline materials is that they must maintain Charpy USE at an acceptable level throughout the life of the RPV. Pressure temperature (PT) 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 RPV that can cause severe overcooling of the RPV wall, followed by immediate repressurization. The thermal stresses caused when the inside surface of the RPV 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 traditionally most susceptible to PTS are the materials in the RPV beltline where neutron radiation gradually embrittles the material over time.

Austenitic stainless steels react to neutron radiation differently than ferritic steels, being more resistant to loss of toughness. Radiation effects on austenitic stainless steels are generally a concern at fluences several orders of magnitude higher than those typically experienced by RPV beltline materials. In addition, austenitic stainless steels do not generally exhibit a ductile to brittle transition as temperatures decrease (e.g. they do not have “upper shelf energy” as they do not exhibit upper and lower shelf behaviors).

The NuScale design includes a lower RPV constructed of austenitic stainless steel, which does not undergo neutron embrittlement to the same degree or kind as traditional ferritic steel as noted above. The applicant consequently requested exemption (Exemptions 6 and 15, SDAA Part 7) from requirements in 10 CFR 50.60, 10 CFR 50 Appendix G, 10 CFR 50 Appendix H, and 10 CFR 50.61 for the lower RPV and described how these requirements apply to the upper RPV which receives substantially less neutron radiation than the lower RPV. Those exemptions, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale’s SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

5.3.2.2 Summary of Application

FSAR: The applicant described how it addresses Limit Curves (pressure-temperature limits, P-T limits), Operating Procedures, PTS, and Upper-Shelf Energy (Charpy USE) in FSAR, Section 5.3.2, “Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses,” summarized, in part, below.

FSAR, Section 5.2.3, 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 Figs. 5.3-2, 3, and 4. The methodology for developing the P-T limit curves is documented in NuScale Technical Report, “Pressure and Temperature Limits Methodology,” included in FSAR, Chapter 5. A COL applicant that references the NuScale SDAA will develop PT limit curves based on plant specific final design transients, material properties, and similar 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 applicant’s 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: FSAR, 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 PTLR.

Technical Reports: The staff has documented its review of the technical report, “Pressure and Temperature Limits Methodology” in detail in a separate SER (ML24332A008).

5.3.2.3 Regulatory Basis

SRP Section 5.3.2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock,” 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, 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 typically include the following:

- RG 1.99, Revision 2, as it relates to RPV 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

The criteria in RG 1.99, Rev. 2, and RG 1.190 were not necessary to reach regulatory conclusions discussed here in the SE, Section 5.3.2 as is otherwise typical for LWR designs.

The requirements of 10 CFR Part 50.60, 50.61, and 10 CFR Part 50, Appendices G and H apply in part, or in whole, to ferritic steels used as part of the RPV and not to austenitic stainless steels.

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 provided Figures 5.3-2 through 5.3-4, with corresponding numerical values tabulated in Tables 5.3-2 and 5.3-3. These figures and tables were generated using a pressure temperature limits methodology (PTLM) submitted as TR-130877-P, “Pressure Temperature Limits Methodology,” for NRC review and approval. This PTLM follows the guidelines of GL 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits,” dated January 31, 1996, and provides the methodology to generate bounding P-T limits and LTOP system limits. The staff has documented its review of the PTLM in detail in a separate SER (ADAMS Accession No. ML24332A008). The staff concluded that the PLTM satisfied the requirements of 10 CFR Part 50, Appendix G to the extent it could be verified without adequate completion and review of information consistent with COL Item 5.3-1. Several important aspects of the PTLM necessary to generate a Pressure Temperature Limits Report (PTLR) were incomplete or preliminary and could not be verified by the staff as complete and acceptable. These aspects are described in COL Item 5.3-1.

The PTLM sample output is based on preliminary design information and material properties. The results in the SDAA and PTLM cannot be applied directly by the applicant without adequate completion and review of information consistent with COL Item 5.3-1. The staff confirmed that the general approach and quality of the PTLM can support a future approval under the GL 96-03 once the incomplete or preliminary aspects are addressed consistent with COL Item 5.3-1 and a PTLR is provided for review. The staff reviewed aspects of the approach such as clad stress modeling, implementation of finite element analysis (FEA), identification procedure for limiting stress location, and selection of appropriate transients consistent with PTLM Section 3.3. The staff could not verify that the FEA model will bound or represent a COL application design; that the full set of COL application transients are represented or bounded by the transients analyzed in the PTLM or SDAA; and what the final as-built material properties will be.

Should a complete PTLR be submitted and approved, the staff concludes that the PTLR approach would be compatible with the applicant's GTS and that the PTLR-related TS provisions would meet the technical criteria of GL 96-03 to the extent they apply (some aspects of GL 96-03 would be inapplicable consistent with Exemptions 6 and 15). Consequently, the staff concludes that a PTLR approach could be acceptable for generic use by NuScale COL applicants for establishing limiting PT limit curves, LTOP system limits, and related input parameters. Pursuant to TS 5.6.4c, future NuScale COL holders would be required to provide the PTLR to the NRC upon issuance for each RPV neutron fluence period and for any PTLR revision of supplement thereto. Finally, in accordance with GL 96-03, the NRC must approve any subsequent changes in the method used to develop the P-T limits.

The staff noted that, consistent with TR-130877-P, Rev. 1, "Pressure and Temperature Limits Methodology," it is not expected that the pressure-temperature limits will need to be updated based on changes in fluence. The staff's review of this is documented in section 5.3.1.4 of this SE within the neutron embrittlement discussion.

FSAR, Section 5.3.2.2, "Operating Procedures," states that FSAR, Section 13.5, "Plant Procedures," addresses plant operating procedures to ensure that the P-T limits are not exceeded. COL Item 5.3-1 states that COL applicants referencing the SDAA will develop operating procedures to ensure that the P-T limits for the as-built reactor vessel are not exceeded and that these limits will be generated based on final transients and the material properties of the as-built RPV. The staff reviewed the COL item and finds it acceptable, as it supports conformance with the requirements of 10 CFR Part 50, Appendix G.

Based on the above the staff cannot make a finding regarding Figures 5.3-2 through 5.3-4 or the corresponding Tables 5.3-2 through 5.3-4 at this time due to a lack of information. This information is detailed in COL Item 5.3-1 including confirmation of several inputs leading to the PTLM generated bounding P-T limits. The staff does not make a finding regarding the specific acceptability of the limits presented in the PTLM or the SDAA for use to generate Pressure-Temperature Limits or PTLRs at this time.

5.3.2.4.2 Pressurized Thermal Shock

PTS events are potential transients in a pressurized-water RPV that can cause severe overcooling of the RPV wall, followed by immediate repressurization. The thermal stresses, caused when the inside surface of the RPV 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 RPV beltline where neutron radiation gradually embrittles the material over time.

The lower RPV contains no PTS susceptible material due to being austenitic stainless steel. The applicant documented this in SDAA technical report TR-130721-NP, "Use of Austenitic Stainless Steel for NPM Lower Reactor Vessel," as discussed in section 5.3.1.4.8 of this SE. The applicant submitted Exemption 15 from the requirements of 10 CFR 50.61 as they relate to the lower RPV.

Upper RPV shell materials, though low-alloy ferritic steel, are not predicted to experience sufficient neutron embrittlement (exceeding 1×10^{17} n/cm^2) to be screened for PTS under the requirements of 10 CFR 50.61. The staff confirmed that, based on the SDAA design, the upper RPV cannot credibly age to the extent that it would require monitoring the upper RPV materials against the 10 CFR 50.61 criteria (e.g. initial properties will remain essentially unchanged over the design life).

On the basis of the above, the staff determined that the design is acceptable because the proposed materials and embrittlement conditions preclude credible risk of PTS events challenging the integrity of the reactor power module as regulated in 10 CFR 50.61. This is because the material properties of the subject materials are expected to remain well in excess of the requirements of 10 CFR 50.61.

5.3.2.4.3 Charpy Upper-Shelf Energy

The lower RPV contains no low alloy ferritic material, rather it is to be made of austenitic stainless steel. As described in TR-130721-NP and discussed in section 5.3.1.4.8 of this SE, this austenitic stainless steel will retain substantial toughness considerably beyond the proposed fluence levels for this design. The applicant specified, in SDAA Section 5.3.2.4, that ferritic materials in the upper RPV will meet the requirements of ASME Code, paragraph NB-3210. Namely these materials must have a minimum Charpy upper-shelf energy (USE) of 50 ft-lbs. Further, the upper RPV materials will receive too little fluence (less than 1×10^{17} n/cm^2) for any changes in Charpy Upper-Shelf Energy due to neutron embrittlement to be of concern. The staff reviewed the requirements on material toughness proposed for these materials and found them to be adequate.

This is further discussed in exemption request 6 and discussed in section 5.3.1.4 of this SE. Consequently, the staff find that Charpy Upper-Shelf Energy has been appropriately considered for this design.

5.3.2.5 Combined License Information Items

SER Table 5.3.2-1 lists a COL information item number and description from FSAR, Table 1.8-1.

Table 5.3.2-1 NuScale COL Information Items for FSAR, Section 5.3.2

Item No.	Description	FSAR Section
5.3-1	An applicant that references the NuScale Power Plant US460 standard design will choose the final transients to generate the reactor vessel pressure-temperature limits report and limiting conditions for operation. An applicant that references the NuScale Power Plant US460 standard design will confirm that the design geometries, final transients, and material properties of the reactor pressure vessel are bounded by (or identical to) those used in the Pressure and Temperature Limits Methodology to confirm that the example curves in the Standard Design Approval Application are applicable. Operating procedures will ensure that pressure-temperature limits for the as-built reactor are not exceeded. These procedures will be based on the limits defined in the pressure temperature limits report and material properties of the as-built reactor vessel.	5.3.2.2

The staff finds the above list to be complete and that it adequately describes actions for the COL applicant.

5.3.2.6 Conclusion

The staff does not conclude 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. Information necessary to complete a staff review is detailed in COL Item 5.3-1 and the PTLM SE. The use of operating limits, as determined by the criteria defined in SRP Section 5.3.2, would provide reasonable assurance that nonductile or rapidly propagating failure will not occur pending a complete PTLR or other approach. The staff verified that the general approach of relying on a PTLR methodology as implemented in the SDAA would be acceptable pending a reviewed and approved PTLR based on the PTLM and review of information provided consistent with COL Item 5.3-1.

The staff concludes that a material surveillance program, developed in compliance with 10 CFR Part 50, Appendix H, is not warranted as discussed in Section 5.3.2.4 above. The staff further concludes that PTS and USE have been adequately addressed by the applicant. This constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a, 10 CFR 50.60, and 10 CFR 50.61; 10 CFR Part 50, Appendix G; and GDC 1, 14, 31, and 32, with respect to PTS and USE (but not P-T limits as discussed above in Section 5.3.2.4.1).

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 Section 5.3.1 and 5.3.2, the integrity of the RPV 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 RPV integrity. The only information unique to FSAR, Section 5.3.3, “Reactor Vessel Integrity,” pertains to shipment and installation.

5.3.3.2 Summary of Application

FSAR: FSAR, Section 5.3.3, describes RPV integrity, as summarized, in part, below.

FSAR, Section 5.3.3, provides references to the appropriate FSAR, sections for design, materials of construction, fabrication methods, inspection requirements, operating conditions, inservice surveillance, and threaded fasteners. The applicant stated that packaging, shipping, handling, and storing the RPV 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-1-2015, Subpart 2.2. The use of a nonchloride, noncorrosive desiccant maintains a dry environment for all RPV surfaces. Humidity indicators will be used during shipping. Both the primary and secondary sides of the RPV will be maintained under positive pressure during shipment with the internal atmosphere of the SG tubes evacuated and filled with nitrogen. The applicant stated that “appropriate foreign material exclusion measures” will be taken for shipment by the fabricator.

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

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

Technical Reports: There are no TRs for 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 (note, the SDAA includes an exemption request from 10 CFR Part 50, Appendix H).

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 of this SER. Those sections of this SER associated with the sections of the FSAR, referenced in FSAR, Section 5.3.3, contain a detailed discussion of the staff's findings.

With regard to shipment and installation, the integrity of the RPV is maintained by ensuring that the as-built characteristics of the vessel are not degraded by improper handling. FSAR, 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 RPV. A dry environment is maintained for all RPV 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 RPV 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 SDAA is acceptable to the staff because proper cleanliness and freedom from contamination during all stages of shipping, storage, and installation of the RPV 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 FSAR, Sections 5.3.1 and 5.3.2, and the PTLR referenced in FSAR, Section 5.3.2, the staff concludes that the structural integrity of the applicant's RPV is acceptable because it meets the requirements of GDC 1, 4, 14, 30, 31, and 32; 10 CFR Part 50, Appendix G and Appendix H; and 10 CFR 50.55a (as would be modified by SDAA Exemptions 6 and 15 and the finding in SE Section 5.3.2 should those exemptions be granted to a COL applicant for the reasons provided in NuScale's SDAA). The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the applicant's plant conform to the applicable NRC regulations and the ASME Code. The applicant's design meets the fracture toughness requirements of the regulations and ASME Code, Section III. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with the regulations and the ASME Code (subject to COL Item 5.3-1).

5.3.4 SDAA Part 7, Section 6 Exemption, Acceptance Criteria for Fracture Prevention Measures

5.3.4.1 Introduction

FSAR, Section 5.3 typically establishes compliance with the requirements of 10 CFR 50.60, and 10 CFR 50 Appendix G and H. SDAA Part 7, Section 6 presents an exemption request from the requirements of 10 CFR 50.60, and 10 CFR 50 Appendix G and H for the NPM lower RPV. The request is justified based on the inapplicability of the subject requirements to the NuScale design, specifically the use of austenitic stainless steel for the lower RPV instead of traditional ferritic materials.

5.3.4.2 Regulatory Basis

The relevant regulatory requirements from which Exemption is requested are,

- CFR 52.137(a)(14) requires a standard design approval application final safety analysis report (FSAR) to include, in part, a description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61.

- 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 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

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)).

5.3.4.3 Summary of Application

The applicant provided justification for the exemption by contrasting the requirements of the regulations noted above, which pertain to ferritic materials, to the austenitic stainless steel selected for the subject NPM lower RPV. The applicant described the difference in material behavior and aging demonstrated by austenitic stainless steel in comparison to traditional ferritic steels. This includes decreased susceptibility to thermal and neutron aging effects. For example, austenitic stainless steel such as the selected FXM-19 (and associated welding materials) do not undergo a brittle-to-ductile transition as ferritic steels commonly do. In addition, the selected steel has higher inherent toughness than traditional ferritic materials.

The applicant also provided, in FSAR, Section 5, TR-130721-P, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel." This technical report contains an evaluation of austenitic stainless-steel properties, results of a literature review, and evaluation of austenitic stainless-steel data to address the purpose of the cited regulations relevant to the exemption request.

5.3.4.4 Technical Evaluation

The applicant states that because austenitic stainless steels maintain ductility considerably beyond the peak forecast design fluences of the lower RPV, the "underlying" purpose behind the 10 CFR 50.60 requirements (e.g. application of 10 CFR 50 Appendices G and H) is fulfilled by the material selection in lieu of the requirements.

The staff notes that austenitic stainless-steel materials do not exhibit the same material characteristics and embrittlement effects as ferritic materials. Specifically, austenitic stainless steels have inherently high ductility (e.g. toughness), generally do not have a ductile-to-brittle transition temperature (otherwise managed through the RT_{NDT} correlation) and undergo measurable embrittlement at higher fluences than traditional ferritic materials. As the requirements in 10 CFR 50.60 (specifically use of 10 CFR 50 Appendices G and H) were designed for ferritic steel materials, the staff found that the requirements are not suitable to being applied to the subject austenitic stainless steel.

The technical basis for the applicant's claims, and the staff's detailed technical review of these claims, as they apply to the lower RPV, are documented above in Section 5.3.1.4.9 concerning the TR-130721-P, "Use of Austenitic Stainless Steel for NPM Lower Reactor Vessel." The staff concurs that the neutron and thermal embrittlement susceptibility of austenitic stainless steel, as proposed by the applicant, is inherently superior to traditional ferritic materials such that the requirements of 10 CFR 50.60 (predicated on use of ferritic steels) are inapplicable to the NuScale lower RPV design. This finding is based on the text of the regulations themselves (predicated on use of ferritic steel); substantial and pertinent materials testing data presented in technical report TR-130721-P; and the staff's independent literature review.

The manner in which this request supports the criteria for approving an exemption are presented below:

10 CFR 50.12(a)(1) Authorized by law

The Exemption request is made consistent with the requirements of 10 CFR 50.12.

10 CFR 50.12(a)(1) The requested exemption will not present an undue risk to the public health and safety

The regulatory requirements from which exemption is requested do not apply to the subject design because the applicant is using a different material than which the requirements are for. This is consistent with the introduction/scope sections of 10 CFR 50 Appendix G, and 10 CFR 50 Appendix H (i.e. "for ferritic") which would be otherwise required by 10 CFR 50.60. Lack of compliance with the subject requirements does not negate appropriate consideration of potential analogous effects through demonstration of design adequacy based on design criteria.

10 CFR 50.12(a)(2) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever—

(ii) 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…

(vi) There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption.

The staff finds that special circumstances are present because the applicant has proposed a material for the NPM lower RPV that does not exhibit the material properties subject to the requirements of 10 CFR 50 Appendices G and H. Application of the requirements of these Appendices through the requirements of 10 CFR 50.60 would not serve the underlying purpose of the rule as the rule was written for designs utilizing ferritic steels, not the proposed austenitic stainless steel.

5.3.4.5 Conclusion

Because of the approach used in the applicant's design, as documented above in the technical evaluation, special circumstances are present, consequently textual compliance is not necessary to achieve the underlying purposes of the Appendices (e.g. requirements pertinent to ferritic materials). For the reasons set forth in the evaluation above, the staff finds that the requested exemption to 10 CFR 50.60, 10 CFR Appendix G, and 10 CFR Appendix H, meet the requirements of 10 CFR 50.12(a) and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

5.3.5 SDAA Part 7, Section 15 Exemption, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock

5.3.5.1 Introduction

SDAA Part 7, Section 15 presents an exemption request from the requirements of 10 CFR 50.61, for the NPM lower RPV. The request is justified based on the inapplicability of the subject requirements to the NuScale design, specifically the use of austenitic stainless steel for the NPM lower RPV instead of traditional ferritic materials.

5.3.5.2 Regulatory Basis

The relevant regulatory requirements from which an exemption is requested are,

- CFR 52.137(a)(14) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

  A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61.

- 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.

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)).

5.3.5.3 Summary of Application

The applicant provided justification for the exemption by describing how the requirements of the regulation noted above, which pertains to ferritic materials, cannot be applied to the austenitic stainless steel selected for the subject NPM lower RPV. For example, the chemistry tables provided in 10 CFR 50.61 are based on traditional ferritic material compositions and do not meaningfully apply to austenitic stainless steels.

The applicant also states that austenitic stainless steels exhibit decreased susceptibility to thermal and neutron aging effects in comparison to traditional ferritic materials. For example, austenitic stainless steel such as the selected FXM-19 (and associated welding materials) can be expected to exhibit minor neutron and thermal embrittlement effects under the NuScale design conditions (relative to ferritic materials). The applicant noted that due to this, the underlying purpose of 10 CFR 50.61, managing pressurized thermal shock events, are met by evaluating the ferritic upper portions of the NPM RPV rather than the austenitic NPM lower RPV.

The applicant also provided in FSAR, Section 5, technical report TR-130721-P, “Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel.” This technical report contains an evaluation of austenitic stainless-steel properties, results of a literature review, and evaluation of austenitic stainless-steel data to address the purpose of the cited regulations relevant to the exemption request.

5.3.5.4 Technical Evaluation

The applicant states that traditionally ferritic beltline materials are evaluated for sufficient ductility under postulated PTS events. The requirements of 10 CFR 50.61 require the calculation of a RT_{PTS} value with which to evaluate the adequacy of the subject material toughness. The calculational methods provided in 10 CFR 50.61 are derived from ferritic data and do not represent austenitic material behavior. Consequently, the staff finds that application of the requirements of 10 CFR 50.61 would not produce credible results pertinent to evaluation of the susceptibility of the subject design to PTS events. In addition, the rule language specifies that such calculations are only required for beltline ferritic materials that are predicted to undergo sufficient neutron radiation damage to be considered in selection of the most limiting material with regard to radiation damage, of which there are none in the proposed design.

The applicant noted that austenitic stainless steels exhibit minor neutron and thermal embrittlement effects. The applicant supported this assertion by providing FSAR, Section 5, TR-130721-P, “Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel.” The staff reviewed this technical report in detail, conducted independent literature review, and as discussed elsewhere in this SE (Section 5.3.1.4.8), finds that the behavior of austenitic stainless steel is superior to traditional ferritic steels in this regard. In addition, the staff confirmed that the inherent toughness properties of the proposed FXM-19 (and associated welding materials) are superior to traditional ferritic steels.

Based on the presentation of the superior austenitic properties, the applicant concludes that PTS susceptibility is managed in the design of the lower RPV through selection of superior material. The staff finds this acceptable based on the substantial margin between fluences of concern for austenitic material radiation damage effects and the proposed fluence levels in the NuScale NPM Lower RPV design. In addition, the staff confirmed that thermal aging effects are similarly constrained and do not require further evaluation.

Finally, the staff evaluated the applicant's approach of demonstrating PTS adequacy by evaluating the ferritic NPM upper RPV. The staff found that this is a conservative alternative approach as this material represents the material most susceptible to a postulated PTS event. Consequently, the applicant's PTS evaluation of this material supports findings consistent with the purpose of 10 CFR 50.61.

5.3.5.5 Conclusion

Because of the approach used in the applicant's design, as documented above in the technical evaluation, special circumstances are present because the purpose of the regulation cannot be achieved through compliance. For the reasons set forth in the evaluation above, the staff finds that the requested exemption to 10 CFR 50.61 meets the requirements of 10 CFR 50.12(a) and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

5.4 Reactor Coolant System Components and Subsystem Design

5.4.1 Steam Generators

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 applicant's 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 secondary steam production occurring inside the SG tubes (secondary side).

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

5.4.1.2 Summary of Application

By application dated November 29, 2022 (ML22339A066 (Package)), the applicant submitted information related to the NuScale US460 SG materials and design. The applicant submitted Revision 2 of the SDAA by letter dated April 9, 2025 (ML25099A250). The applicant supplemented the SDAA by letters dated December 19, 2024 (ML24354A160 (Nonproprietary)).

FSAR, 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), Comprehensive Vibration Assessment Program (CVAP), density wave oscillations (DWO), and design features for accessing the secondary side.

FSAR, Section 5.4.1.1, "Design Basis," states that the applicant's SGs provide two safety-related functions—they form a portion of the RCPB and transfer decay heat to the DHRS. FSAR, 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, inlet flow restrictor, feed plenum access port and access cover, SG tubes, 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 applicant's design are inside the RPV, and the RPV 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. FSAR, Section 5.4.1, states that the feedwater plenum is within the feed plenum access port, and the main steam plenum is within the steam plenum cap and PZR baffle plate. FSAR, 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 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). Each NPM has eight sets of SG supports and SG tube support assemblies, consisting of upper and lower SG supports, SG tube supports, support clips, and backing strips, to provide vertical, lateral, vibration, and seismic support. With the intent to limit DWO, the design includes inlet flow restrictors individually installed in each SG tube and seated against the FW plenum tubesheet secondary face. During shutdown conditions, the design includes a single spring-operated thermal relief valve located on each feedwater line that vents directly into the containment, to provide overpressure protection for the secondary side of the SGs, the steam and feedwater piping inside containment, and the DHRS.

On the primary side, 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 downward into the SG region between the RPV 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. FSAR, Section 15.0.5.2.1, "Riser Flow Characteristics," states that the upper riser assembly has four groups of riser holes located at discrete elevation levels along the riser wall for a small amount of the reactor coolant to bypass the top of the riser and flow into the SG tube bundle region. The bypass flow is included for maintaining boron concentration in the reactor coolant in the downcomer during certain non-LOCA transients and small RCS leaks. These scenarios are discussed in FSAR, Section 15.0.5, "Extended Passive Cooling for Decay and Residual Heat Removal." The four groups of holes in the upper riser assembly are included in the CVAP. The staff evaluates the CVAP in SER Section 3.9.2.

On the secondary side, feedwater flows from each feed plenum access port 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 the stability performance of the SGs in FSAR, Section 5.4.1.3, “Performance Evaluation.” The staff reviewed TR-0516-49417-P-A, “Evaluation Methodology for Stability Analysis of the NuScale Power Module,” Revision 1, March 2020 (ML20086Q668). The staff reviewed this information under DSRS Section 15.9.A, “Thermal Hydraulic Stability Review,” and Section 15.9 of this SER contains the staff’s evaluation. The staff also reviewed TR-121353, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” Revision 1, October 2024 (ML24296A470). The staff reviewed this information under 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. There are no TRs for the SG program.

The applicant described the proposed SG program in FSAR, Sections 5.4.1 and 5.4.1.6. FSAR, Section 5.4.1.6, states that the applicant’s SG program will provide monitoring and management of tube degradation and degradation precursors and will permit preventive and corrective actions to be taken in a timely manner. The program is based on Nuclear Energy Institute (NEI) 97-06, RG 1.121, and the STS (NUREG-1431, Revision 5, and NUREG-1432, Revision 5). FSAR, Table 1.9-2 states partial conformance with RG 1.121 with regards to an alternate loading condition, which is discussed further in SER Section 5.4.1.6. The major program elements are assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary to secondary leakage monitoring, shell-side (primary for this design) integrity assessment, primary- and secondary-side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.

There are no TS specifically related to SG materials and design. The applicant provided the TS and bases related to the SG program in FSAR, Chapter 16. Specifically, this information is 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 5.4.1.6 discusses them further.

In SDAA, Part 8, “License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC),” Section 2.1, the applicant described the components of the RCS, including the SGs, and includes design information and ITAAC related to verification that the SGs meet ASME Code design requirements. Section 14.3 of this SER describes the staff’s evaluation of the information in SDAA, Part 8, including the ITAAC related to the SGs. There are no ITAAC for the SG program.

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 requires that the RCPB being designed, fabricated, erected, and tested to ensure 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.

- 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 50.36(c)(3) requires TS to include items in the category of "Surveillance Requirements," and 10 CFR 50.36(c)(5) requires TS to include items in the category of "Administrative Controls."

- 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.a(e) generally require certain groupings of components, including those comprising the pressure boundaries, to meet the requirements of ASME Code, Section III.

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

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

- 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.

In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- DSRS 5.4.2.1, “Steam Generator Materials” (ML15355A532), and DSRS 5.4.2.2, “Steam Generator Program” (ML15355A535), that list the acceptance criteria adequate to meet the above requirements and include review interfaces with other SRP sections.

- Regulatory guidance related to the welding of SG components:

  – RG 1.31, as it relates to control of ferrite content in stainless steel weld metal.

  – RG 1.34, as it relates to control of the properties of electroslag welds.

  – RG 1.43, as it relates to control of stainless-steel weld cladding of low-alloy steel components.
  – RG 1.50, as it relates to control of preheat temperature for welding of low-alloy steel.
  – RG 1.71, as it relates to welder qualification for areas of limited accessibility.

- 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.

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

- NEI 97-06, “Steam Generator Program Guidelines.”

- NUREG-1431, “Standard Technical Specifications—Westinghouse Plants,” Revision 5, issued September 2021 (ML21259A155 and ML21259A159).

- NUREG-1432, “Standard Technical Specifications—Combustion Engineering Plants,” Revision 5, issued September 2021 (ML21258A421 and ML21258A424).

- TS Task Force (TSTF) Traveler-577, “Revised Frequencies for Steam Generator Tube Inspections,” Revision 1, dated March 1, 2021 (ML21060B434).

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

5.4.1.4 Technical Evaluation

The staff reviewed FSAR, Section 5.4.1, in accordance with 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 5.4.4.2. SER Sections 5.4.4.4.2 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 SDAA Section 5.2.3; Section 5.4.1.5, "Steam Generator Materials;" Table 5.2-3; and Table 5.4-3, "Steam Generator System Component 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 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.

The applicant defined the quality groups in FSAR, Section 3.2.2, "System Quality Group Classification," and identified the design criteria in FSAR, Table 3.2-2. FSAR, 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. FSAR, Section 5.4.1.5, also states that the RCPB materials used in the SGs are classified as Quality Group A and are designed, fabricated, constructed, tested, and inspected as ASME Code Class 1. FSAR, Section 5.4.1.5, also states that the integral steam plenum, integral steam plenum caps, feed plenum access ports, and feed plenum access port covers are classified as Quality Group A and are designed, fabricated, constructed, tested, and inspected as ASME Code Class 1. The steam plenum access ports and steam plenum access port covers are classified as Quality Group B and inspection conforms to ASME Code Class 2. As permitted by the ASME Code, the applicant has elected to fabricate, construct, and test these components in accordance with ASME Code, Section III, Subsection NB. The feedwater and main steam piping from the CNTS to the plenum nozzles, including the thermal relief valves, are classified as Quality Group B and are designed, fabricated, constructed, tested, and inspected as ASME Code Class 2. FSAR, Section 5.4.1.5, further states that the welding of the RCPB and secondary 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.

FSAR, Section 3.9.3.1.2, "Load Combinations and Stress Limits," states that the lower SG supports, support clips, and SG tube supports are designated as internal structures in accordance with ASME Code, Section III, Subsection NG, and the upper SG supports are designated as ASME Code Class 1 in accordance with ASME Code, Section III, Subsection NF. FSAR, Section 3.9.3.1.2, states that the lower SG supports, upper SG supports, SG tube supports, and support clips are seismic Category I components. FSAR, Section 5.4.1.2, states that the upper SG supports are welded to the pressurizer baffle plate and the inner surface of the RPV, and the lower SG supports are welded to the inner surface of the RPV. In accordance with ASME Code, Section III, Subparagraph NB-1132.2(d), the welds connecting the SG supports to the RPV are considered part of the RPV; therefore, the welds are classified as ASME Code Class 1 and conform to ASME Code, Subsection NB. FSAR, Section 5.2.3, addresses the design, fabrication, inspection, testing, and quality assurance of the welds connecting the upper and lower SG supports to the RPV. ASME Code, Section XI, Subsection IWB, includes the ISI requirements for ASME Code Class 1 components.

FSAR, 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. 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, Alloy 690 TT has resisted degradation by corrosion mechanisms. FSAR, Table 5.4-2, “Steam Generator Design Data,” notes that the tubes have a tube wall outer diameter of 0.625 inch (15.9 millimeter (mm)) and a tube wall thickness of 0.050 inch (1.27 mm). 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 (ML072330588)). FSAR, Section 5.4.1.6, “Steam Generator Program,” states that the greater wall thickness is based on incorporation of a substantial degradation allowance. FSAR, Section 5.4.1.3.1, 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.

FSAR, Table 5.4-3, notes that the SG supports, SG tube supports, and SG backing strips are Type 304. FSAR, Table 5.2-3, notes that the steam plenum and feed plenum access ports, including the main steam and feedwater supply nozzles that are part of the steam plenum access ports and feed plenum access ports, are SA-508, Grade 3, Class 2, low alloy steel, and the steam plenum cap is Alloy 690. FSAR, Section 5.4.1.5 states that the secondary side surfaces of the steam plenum and feed plenum tubesheets are clad with Alloy 52/152, and the remaining inside and outside surfaces of the steam and feedwater plenums are either Alloy 690 or low alloy steel clad with austenitic stainless steel. FSAR, Table 5.4-2, notes that the steam tubesheet thickness is 4.0 inches (10 centimeters (cm)) without clad and 4.750 inches (12.065 cm) with clad, and the feed tubesheet thickness is 4.5 inches (11.43 cm) without clad and 5.25 inches (13.335 cm) with clad. The cladding on the outside surface of the steam and feed tubesheets is 0.250 inch (6.35 mm), and the cladding on the inside surface of the steam and feed tubesheets is 0.500 inch (12.7 mm) (total steam and feed tubesheet cladding thickness of 19.05 mm). FSAR, Table 5.2-3, notes that the steam and feed plenum access port covers are Type 304 or FXM-19, the studs, nuts, and washers for the steam and feed plenum access port covers are nickel-based SA-193 Grade B8 Class 1, Grade B8R, or Grade B8M Class 1; SA-194 Grade 8; SB-637 UNS N07718; or SA-564

Type 630 H1100; and the threaded inserts for the steam and feed plenum access ports are SA-193 Grade 8 Class 1, Grade B8M Class 1, Grade B8R, or Grade B8S Carbide Solution Treated. FSAR, Table 5.4-3, notes that the ASME Code Class 2 SG piping is Type 304 or 316, the SG piping support materials include Type 304/304L, 316/316L, 405, or 410/410S, and the SG pipe fittings are Grade F304 or F316. FSAR, 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 FSAR, Table 6.1-3. FSAR, Table 5.4-3, also notes that the flow restrictor flanged sleeve, collet, and locking plate are Type 304; and that the flow restrictor mandrel with orifice and hardware are Type 316. FSAR, Section 5.4.1.5, states that the SG weld filler metals are in accordance with ASME Code, Section II, Part C. FSAR, Tables 5.2-3 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 material for pressure-retaining bolting. 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.

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 applicant's 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 5.4.4.4.1, 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 plenum nozzles, including the thermal relief valve, 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.

FSAR, 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 ASME Code Class 1 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 and are welded to the tubesheets on the secondary side. FSAR, 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. FSAR, Section 5.4.1.1, states that the load combinations on the RPV, including the SG tubes, are identified in FSAR, Table 3.9-3, "Load Combinations for Reactor Pressure Vessel Shell and Head."

Operating experience has shown that crevices around SG tubes in the tubesheet region have caused corrosion-related degradation. FSAR, Section 5.4.1.2, states expansion of the tube within the tubesheet bore minimizes crevice depths. The staff determined that the applicant's design meets the acceptance criteria in 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.

FSAR, 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). DSRS Section 5.4.2.1 states that the residual stresses from tube bending should be evaluated and relieved, as needed. The applicant evaluated this need according to industry guidance for procurement of Alloy 690 SG tubes and determined no bend radii are small enough to require stress relief. On this basis, the applicant proposed no thermal stress relief for the tube bends. The staff finds this acceptable based on following the industry guidance, which, for the applicant's tubing diameter, would specify thermal treatment for bends with a radius less than 15.88 cm (6.250 inches). The staff does not review or approve EPRI's "Guidelines for Procurement of Alloy 690 Steam Generator Tubing," but the guidelines were developed using technical expertise and operating experience and have been applied to numerous replacement SGs at operating plants with no incidents of SCC.

The acceptance criteria in 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 applicant's SG design has eight sets of austenitic stainless steel (Type 304) SG supports and SG tube support assemblies that provide vertical, lateral, vibration, and seismic support. The SG tube support assemblies 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 RPV and the integral steam plenum and are anchored at the bottom by connection to the lower SG supports that are welded to the RPV shell below the SG tube bundle. FSAR, Section 5.4.1.2, states that backing strips between the upper riser assembly and the first column of tubes completes the enclosure of the column 1 tubes and is the interface between the upper riser assembly and the SG tube support. FSAR, Section 5.4.1.2, also states that a SG tube support spacer affixed to the column 21 SG tube support by a socket head screw cap provides support for the column 21 tube support in the region where the RPV inner diameter flares.

FSAR, Section 5.4.1.2, states that crevices are minimized along the tubes, tube supports, and tubesheets to limit corrosion product buildup. Because the SG tube-to-tubesheet contact is within the primary coolant environment corrosion products are minimal. Expansion of the tube within the tubesheet bore minimizes crevice depths. FSAR, Section 5.4.1.2, also states that the SG support and SG tube support material limits the generation and buildup of corrosion products, and that the geometry of the SG supports, and SG tube supports minimizes crevices and facilitates flow. The staff determined that the design of the SG supports, and SG tube supports meets the guidance in 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. FSAR, Table 5.4-3, notes that the SG piping support materials include Type 304/304L, 316/316L, 405, or 410/410S, and the SG pipe fittings are Grade F304 or F316. FSAR, 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, FSAR, 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, and that the materials also meet the requirements of ASME Code, Section III, Article NC-2000. FSAR, Section 5.4.1.5, states that the structural supports for the SG piping, including weld materials, conform to the fabrication and construction requirements of ASME Code, Section III, Subsection NF.

FSAR, Section 5.4.1.2, describes the SG tube inlet flow restrictors used in the applicant's SG design. Each inlet flow restrictor is individually installed, seats against the secondary side face of the feedwater plenum tubesheet and extends into a portion of the tube. A metallic collet on the inlet flow restrictor expands to seal with the tube inner diameter, secondary side water from the feedwater plenum flows through a center orifice in the mandrel, and the flanged sleeve allows secondary side water to enter the space between the sleeve and tube to feedwater plenum tubesheet weld. FSAR, 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.

FSAR, Section 5.4.1.2, also states that the flow restrictors can be removed to support SG tube inspection, cleaning, tube plugging, repairs, and maintenance activities. When the inlet flow restrictors are removed for SG tube examinations, FSAR, Section 5.4.1.4, states that they will be examined by VT-3. In addition, the applicant stated that the VT-3 examination will look for degradation due to wear or cavitation mechanisms, and the inlet flow restrictor may be reinstalled if acceptance criteria are met. The flow restrictor flanged sleeve, collet, and locking plate are Type 304; and that the flow restrictor mandrel with orifice and hardware are Type 316. The CVAP evaluated the inlet flow restrictors for FIV and leakage flow instability (LFI), and SER Section 3.9.2 further discusses the staff's evaluation of the CVAP.

The purpose of the inlet flow restrictors is to limit flow instability. FSAR, Section 5.4.1.3, states that the inlet flow restrictors are designed to "provide the necessary pressure drop to preclude unacceptable secondary flow instabilities," and it defines acceptable flow instabilities as "flow fluctuations that do not cause reactor power oscillations that could exceed fuel design limits, and that result in applicable ASME BPVC criteria being met."

The staff found that secondary side flow and temperature oscillations do not affect the primary flow stability and fuel design limits. In reaching this conclusion, the staff did not analyze secondary flow stability; rather, the secondary flow was assumed to be unstable. This is discussed in SER Section 15.9 and in the staff's SER for TR-0516-49417-P-A.

One potential type of secondary-side flow instability in the NuScale SGs is DWO. Density wave oscillations in the NuScale SGs could result in temperature oscillations, pressure oscillations, flow oscillations, and flow reversals, which could produce loads on the tubes and inlet flow restrictors that challenge the applicable ASME Code criteria. FSAR, Section 5.4.1.3, states that the "approach temperature" is determined from comparing the RCS hot temperature and main steam temperature, and that the approach temperature correlates with in-tube conditions of potential DWO onset. FSAR, Figure 5.4-16, identifies conditions of reduced margin to DWO onset based on approach temperature. FSAR Section, 5.4.1.3, states that (1) time is counted in DWO when operating in Region 1 shown on FSAR, Figure 5.4-16, or when the RCS average temperature is less than 271 ºC (520ºF) or when there is no main steam superheat, and (2) time counted in DWO is accrued against the cyclic limits provided in FSAR, Table 3.9-1.

The staff review of the approach temperature is provided in SER Section 5.4.1.4.2.1, "Approach Temperature Limit for DWO," and determined that the approach temperature limit provides reasonable assurance of adequate protection against DWO onset for the NPM SG. However, staff approval of the approach temperature limit does not also approve the general use of the NuScale NRELAP5 DWO evaluation model for use in DWO calculations (including the prediction of DWO onset or the prediction of thermal-hydraulic behavior during DWO).

The staff's review of coherent DWO is provided in SER Section 3.9.2.4.3.11. The review concluded operating with coherent DWO conditions is unlikely and operating procedures developed per FSAR, Section 5.4.1.3 and COL item 13.5-3 will describe how to monitor and mitigate system level instabilities such as coherent DWO.

The applicant's design includes a single spring-operated thermal relief valve located on each feedwater line that vents directly into the containment to provide overpressure protection during shutdown conditions. FSAR, Section 5.4.1.2, states that the thermal relief valves are Quality Group B and designed, fabricated, constructed, tested, and inspected as ASME Code Class 2, in accordance with ASME Code, Section III, and are seismic Category I components. FSAR, 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 FSAR, Table 6.1-3.

FSAR, Section 5.4.1.2, states that the applicant's SG design includes provisions to reduce the potential for tube damage caused by loose parts. FSAR, Section 5.4.1.2, also states that the internal (secondary) and external (primary) sides of the tubesheets can be accessed for inspection and removal of foreign objects. As discussed in SER Section 5.4.4.4.1, 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 BTP 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators," is not applicable to the applicant's SG design (i.e., not a top feed or preheat design and no auxiliary FWS), as noted in FSAR, Table 1.9-3, "Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard," the applicant partially conforms to the tests and test procedure guidance. FSAR, Section 10.4.6.2.4, "Condensate and Feedwater Piping," states that the FWS and SG contain design features that minimize the potential for and effect of water hammer. FSAR, Section 3.12.5.7.3, "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. FSAR, Section 10.3.5, "Water Chemistry," states that the "secondary system components and piping exposed to wet steam, flashing liquid flow, or turbulent single-phase flow where 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 FSAR, 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.

The staff determined that the SG design meets the acceptance criteria in DSRS Section 5.4.2.1, as they relate to limiting the potential for degradation of the tubes and other secondary side components. Additionally, the staff determined that the approach temperature limit, as described in FSAR, Section 5.4.1.3, provides reasonable assurance of adequate protection against DWO onset (see SER Section 5.4.4.4.2.1). These criteria, in conjunction with the acceptance criteria for interfacing 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.

5.4.1.4.2.1 Approach Temperature Limit for DWO

The applicant demonstrated that the approach temperature limit provides margin to DWO through NRELAP5 calculations. Specifically, the applicant's calculations show that the approach temperature limit is reached prior to DWO onset being predicted by NRELAP5. The staff evaluated the approach temperature limit by (1) evaluating the approach temperature limit against NRELAP5 DWO calculations (see SER Section 5.4.1.4.2.1.1), (2) evaluating the adequacy of NRELAP5 to predict DWO onset for the SG design (see SER Section 5.4.1.4.2.1.2), (3) evaluating the potential for static instability coupling (see SER Section 5.4.1.4.2.1.3), and (4) evaluating the uncertainty in DWO onset predictions with respect to appropriate accrual of time in potential DWO conditions and the risks associated with DWO (see SER Section 5.4.1.4.2.1.4).

Based on the staff's review, documented in SER Sections 5.4.1.4.2.1.1 through 5.4.1.4.2.1.4, the staff finds that the approach temperature limit provides reasonable assurance of adequate protection against DWO onset for the SG design. This finding is based on the following:

1. The approach temperature limit provides margin to DWO with respect to DWO onset calculations (see SER Section 5.4.1.4.2.1.1),
2. DWO onset calculations provide reasonable insight into the likelihood of DWO (see SER Section 5.4.1.4.2.1.2),
3. Static instability coupling is precluded (see SER Section 5.4.1.4.2.1.3), and
4. Uncertainties in the prediction of DWO onset are reasonable considering the risk associated with DWO (see SER Section 5.4.1.4.2.1.4).

The staff approval of the approach temperature limit does not approve the general use of the NRELAP5 DWO evaluation model for use in DWO calculations (including the prediction of DWO onset or the prediction of thermal-hydraulic behavior during DWO). The staff is unable to determine the adequacy of the evaluation model due to gaps in model assessment (see SER Section 5.4.1.4.2.1.2).

5.4.1.4.2.1.1 Approach temperature limit provides margin to DWO with respect to DWO onset calculations

By letter dated January 13, 2025 (ML25013A242), Figure 1 of Item 1, the applicant shows the results of calculations that demonstrate that the approach temperature limit is reached prior to predicted DWO onset for all cases. Additionally, by letter dated January 13, 2025 (ML25013A242), Item 2, the applicant provides the range of conditions analyzed for DWO onset and the range of operating conditions for the SG. The staff compared the analysis range used in the assessment of the approach temperature limit against the SG operating range in SER Table 5.4.4-1. Based on this comparison, the staff determined that (1) the analysis range is generally consistent with the operating range except for **{{ }}**, (2) the power range analyzed is reasonable because boiling and associated DWO concerns are significantly reduced in the limited range not analyzed below **{{ }}** and the upper bound of the range analyzed exceeds the expected operating range of the SG, (3) the pressure range is acceptable because analyzing at lower maximum secondary side pressures **{{ }}** is conservative with respect to DWO onset, and (4) the feedwater flow rate per SG is acceptable because **{{ }}**.

By letter dated January 13, 2025 (ML25013A242), Item 3, the applicant summarizes conservative biases applied to the DWO onset calculations including biasing the **{{ }}**. Additionally, in letter dated January 13, 2025 (ML25013A239), the applicant clarifies that the approach temperature calculations use model nodalization and time step approaches that are consistent with the evaluation model assessment. The staff finds these inputs reasonably conservative because they are either consistent with the model assessment or provide margin to accommodate error beyond what has been calculated in the evaluation model assessment (see SER Section 5.4.1.4.2.1.2).

In addition to the information discussed in the paragraph above, in letter dated January 13, 2025 (ML25013A225), the applicant clarifies that (1) the approach temperature limit calculations use an inlet flow restrictor loss coefficient range that exceeds the bounds of the values used in the NRELAP5 assessment, and (2) the applicant performed a sensitivity analysis using inlet flow restrictor loss characteristics consistent with the assessment basis. The applicant's analysis showed that reducing the loss characteristics of the inlet flow restrictor does reduce the margin to DWO onset as expected, but margin is still maintained between the approach temperature limit and predicted DWO onset. Based on results of the applicant's sensitivity analysis and the use of conservative inputs as described in the previous paragraph, the staff finds that the applicant used suitably conservative inputs for the approach temperature calculations because the inputs are either suitably conservative or justified using sensitivity analysis. Potential use of the evaluation model outside its assessment basis is evaluated by the staff in SER Section 5.4.1.4.2.1.2.

Based on the information discussed in this section, the staff finds that the approach temperature limit provides margin to DWO onset with respect to DWO onset calculations because (1) the approach temperature limit is always reached before DWO onset is predicted to occur, (2) the DWO onset calculation covers an adequate range of operating conditions for the SG, and (3) the calculations use suitably conservative input. The staff review of the evaluation model used to perform the DWO onset calculations is provided in SER Section 5.4.1.4.2.1.2.

Table 5.4.4-1 Comparison of NPM SG Operating Range against DWO Analysis Range

Parameter	Operating Range(1)	Analysis Range(2)
Power	0 – 100 percent rated thermal power (RTP)	**{{ }}**
Secondary Side Pressure(3)	**{{ }}**	**{{ }}**
Feedwater Subcooling(6)	**{{ }}**	**{{ }}**
Feedwater Flow Rate per SG	**{{ }}**	**{{ }}**
RCS Average Temperature	**{{ }}**	**{{ }}**

1. Approach temperature is applicable to RCS average temperature greater than or equal to 271 ºC (520 ºF) and secondary steam pressures that generate steam superheat.
2. **{{ }}**

3. {{

}}

4. {{

}}

5. {{

}}

6. {{ }}

5.4.1.4.2.1.2 *DWO onset calculation capability*

The staff evaluated the applicant's use of NRELAP5 to perform DWO onset calculations by (1) reviewing the capabilities of the evaluation model, and (2) reviewing the applicant's assessment of the evaluation model.

5.4.1.4.2.1.2.1 *Evaluation model capabilities*

<u>The evaluation model is capable of modeling the relevant geometry and materials</u>

TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4 (ML23005A305), describes the thermal-fluids modeling and associated assessment for the NPM SG, which includes capturing the relevant geometry and materials for the NPM SG, and has been approved by the staff in their SE (ML24334A049). Accordingly, the staff finds that the evaluation model is capable of modeling the relevant geometry and materials for the NPM SG because this capability has been previously reviewed and approved by the staff.

<u>The evaluation model contains the necessary physics</u>

By letter dated January 13, 2025 (ML25013A242), Item 4, the applicant (1) summarizes the important phenomena that were identified in the DWO Phenomena Identification and Ranking Table (PIRT), and (2) describes an applicability assessment to justify how important phenomena, related to pressure drop and heat transfer, are captured by the evaluation model. In addition to the applicant's PIRT, the staff developed an independent PIRT to address helical coil SG DWO phenomena (ML23279A051). The staff compared the applicant's PIRT with the staff PIRT and determined that they were in general agreement with the following exceptions:

- The staff PIRT identified tube fouling and Ledinegg instability as of medium and high importance, respectively. These phenomena were not considered by the applicant's PIRT. However, the staff recognizes that (1) the applicant's analysis of DWO does account for tube fouling (see SER Section 5.4.1.4.2.1.1), and (2) the applicant performed a calculation to analyze the susceptibility of the secondary side of the NPM to Ledinegg instability (see SER Section 5.4.1.4.2.1.3). Accordingly, the staff recognizes this difference between the PIRTs does not impact the evaluation model because the purpose of the PIRT, to ensure that key phenomena are incorporated and assessed for use in the evaluation model, is addressed.

- The applicant’s PIRT considered heat loss to containment and heat gain from the pressurizer and identified both of these phenomena to be of **{{**
}} The staff PIRT did not consider these phenomena because it considered DWO phenomena localized to the coil. Accordingly, the staff recognizes this difference between the PIRTs does not impact the evaluation model because the purpose of the PIRT, to ensure that key phenomena are incorporated and assessed for use in the evaluation model, is addressed.
- The two PIRTs provided different importance rankings for four phenomena **{{**
}} During an audit, the staff examined that the applicant’s applicability assessment that addresses these five phenomena. Accordingly, the staff recognizes this difference between the PIRTs does not impact the evaluation model because the purpose of the PIRT, to ensure that key phenomena are incorporated and assessed for use in the evaluation model, is addressed.

Based on the information described in this subsection and the evaluation model assessment (see SER Section 5.4.1.4.2.1.2.2, “Evaluation model assessment,” below), the staff finds that the evaluation model reasonably contains the necessary physics because (1) the applicant followed an accepted practice, the PIRT process, to identify and assess key phenomena, and (2) the applicant’s PIRT is generally consistent with the independent staff PIRT with justifiable differences.

<u>The numerical method is appropriate for capturing relevant phenomena</u>

The staff examined the applicant’s controls on the **{{**
}} to address the potential for **{{**

}}

By letter dated January 13, 2025 (ML25013A242), Item 5, the applicant describes **{{**

}}

[ ]

By letter dated January 13, 2025 (ML25013A242), Item 6, the applicant **{{**

}} Based on the information described in this subsection, and dependent on the overall evaluation model assessment (see SER Section 5.4.1.4.2.1.2.2, “Evaluation model assessment” below), the staff finds that the **{{**

}}

The algorithm to detect DWO onset is appropriate

By letter dated January 13, 2025 (ML25013A235), the applicant summarizes the DWO detection algorithm assessment as using evaluation model assessment data and showing that the DWO detection algorithm detects DWO onset before significant oscillations occur by indicating the point of DWO onset detection on transient plots. The applicant also clarifies that **{{**

}} Based on the applicant’s description of the DWO algorithm assessment, the staff finds that the DWO detection algorithm provides reasonable indication of DWO onset because (1) the transient plots provided in letter dated January 13, 2025 (ML25013A235)appear to show that DWO onset is indicated near the region where flow oscillations begin to appear, and (2) the applicant applies an uncertainty that bounds the error calculated through code-to-code comparisons of DWO detection algorithm implementation. Additionally, by letter dated January 13, 2025 (ML25013A237), the applicant clarifies that the DWO detection algorithm is applied consistently in its use in model assessment calculations and model implementation calculations (i.e., approach temperature calculations). Based on the information described in this paragraph, the staff finds that the DWO detection algorithm is appropriate because it provides reasonable indication of DWO onset and is applied consistently in model assessment calculations and in model implementation calculations.

Conclusions on DWO evaluation model capabilities

The staff has determined that the evaluation model is can model the relevant geometry, materials, and physics, and that the algorithm to detect DWO is reasonable. The staff’s determination of the adequacy of the numerical method is dependent on the evaluation model assessment (see SER Section 5.4.1.4.2.1.2.2, “Evaluation model assessment” below). Based on these assessments, the staff finds that, conditional to an adequate evaluation model assessment against appropriate experimental data, the evaluation contains adequate modeling capabilities.

In addition to the DWO onset prediction, the staff reviewed the generic thermal-hydraulic modeling capabilities of the evaluation model for the purpose of understanding the risks associated with water hammer (see SER Section 3.9.2). Based on the evaluation model being capable of modeling the relevant geometry and materials, the evaluation model containing the necessary physics, and the staff approval of the thermal-fluids modeling for the NPM SG as described in the SE for TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4 (ML24334A049), the staff finds that the evaluation model provides reasonable prediction of the quality profiles and associated boiling lengths in the NPM SG tubes under normal operating conditions.

5.4.1.4.2.1.2.2 Evaluation model assessment

<u>The experimental data used for assessment is not fully appropriate</u>

Assessment data is independent of the data used to develop the evaluation model

By letter dated January 13, 2025 (ML25013A242), Item 7, the applicant provided **{{**

}} that are used to assess the NRELAP5 DWO model. Additionally, by letter dated January 13, 2025 (ML25013A242), Item 8, the applicant **{{**

}} Based on the information described in this paragraph, the staff finds that the assessment data is independent of the data used to develop the model because the **{{** **}}** data was reserved for model assessment.

Assessment data does not fully cover the DWO analysis envelope

The staff compared the power (as average heat flux per SG tube), secondary side pressure, inlet subcooling, and feedwater flow rate per SG from the data to the ranges used for DWO analyses in SER Table 5.4.4-2 below and notes that:

- **{{** **}}**
- **{{** **}}**

Based on the information described in this paragraph, the staff is unable to determine that data has been collected over a test envelope that covers the DWO analysis envelope, but that data has been collected over a significant portion of the analysis envelope. The staff assessment of the uncertainty, associated in part by assessment data not covering the analysis envelope, is provided in SER Section 5.4.1.4.2.1.4.

Table 5.4.4-2 DWO Analysis Range and SIET TF-2 Conditions

Parameter	DWO Analysis Range	SIET TF-2 Data Range

Average Heat Flux per SG Tube	**{{ }}**	**{{ }}**
Secondary Side Pressure[(1)]	**{{ }}**	**{{ }}**
Feedwater Subcooling	**{{ }}**	**{{ }}**
Feedwater Flow Rate per SG	**{{ }}**	**{{ }}**

1. Analysis secondary side pressure **{{ }}**

The experimental data have been accurately measured

FSAR, Section 17.5, "Quality Assurance Program Description," and MN-122626, "NuScale Power, LLC, Quality Assurance Program Description" (ML24326A359), describes the NuScale quality assurance program which is applicable to testing used to collect DWO assessment data. The NuScale quality assurance program is reviewed in SER Chapter 17. Based on the information in FSAR, Section 17.5, the staff finds that the test facility has an appropriate quality assurance program.

By letter dated January 13, 2025 (ML25013A242), Item 8 and Item 9, the applicant **{{ }}** Based on the information described in this paragraph, the staff finds that experimental data are collected using appropriate measurement techniques because **{{ }}** and (2) the DWO detection algorithm is acceptable as described in SER Section 5.4.1.4.2.1.2.1, "Evaluation model capabilities," above.

By letter dated January 13, 2025 (ML25013A242), Item 10, the applicant describes **{{ }}** Based on the information described in this paragraph, the staff finds that the experimental data accounts for sources of experimental uncertainty, including instrumentation uncertainty, because **{{ }}**

Based on the information described in this subsection, the staff finds that the experimental data have been accurately measured because (1) the test facility has an appropriate quality assurance program, (2) experimental data are collected using appropriate measurement techniques, and (3) experimental data accounts for sources of experimental uncertainty, including instrumentation uncertainty.

Tests are not fully representative of prototypical conditions

By letter dated January 13, 2025 (ML25013A242), Item 11, the applicant provides a comparison **{{**

}} the staff assessment of data coverage (discussed above) evaluates the coverage of thermal-hydraulic conditions and clarifies that the **{{**
}} Based on the information provided in letter dated January 13, 2025 (ML25013A242), Item 11, and **{{**
}}, the staff finds that the test facility has some applicability to the full scale NPM SG because **{{**

}}

By letter dated January 13, 2025 (ML25013A242), Item 12), the applicant describes **{{**

}}, and the staff identified a significant distortion related to **{{**
}} and a minor distortion associated with **{{**
}}

By letter dated January 13, 2025 (ML25013A242), Item 12, the applicant describes **{{**

}}

Section III. A, “Thermodynamic Process,” of ANL-76-23, “Study on Flow Instabilities in Two-Phase Mixtures” (Office of Scientific and Technical Information (OSTI) ID 7277361), clarifies that, “[for a two-phase heat transfer system] if the heated section consists of multiple channels which converge to one channel at the [inlet plenum] and [outlet plenum] with sufficiently large volume capacitance, it is sufficient to consider the system between the two volumes.” Based on the description in ANL-76-23, the staff focused their review on individual tube behavior. The staff recognizes that the design of the NPM SG has **{{**

}}

Additionally, by letter dated January 13, 2025 (ML25013A242), Item 9, the applicant **{{**

}} Based on the information provided in letter dated January 13, 2025 (ML25013A242), Item 9 and Item 12, and the uncertainty propagation discussed above, the staff finds that (1) test distortions associated with the feedwater header arrangement are acceptable because **{{**

}} and (2) test distortions **{{**

}}

During audit activities, the staff identified **{{**

}} The staff assessment of using the evaluation model outside the test envelope is provided in SER subsection, “Use of evaluation model is not restricted to use within its test envelope,” below.

The applicant’s DWO PIRT, provided by letter dated January 13, 2025 (ML25013A242), Item 4, identifies **{{**

}}

The staff recognizes that **{{**

}} Accordingly, the staff expects **{{**

}} Based on the information regarding tube thermal resistance, the staff finds that the test facility distortion **{{** **}}** is acceptable because it results in a very small distortion that is not expected to have significant impact on the applicability of the data.

Based on information described in this subsection on test distortions, the staff is unable to determine that distortions or departures from non-prototypical conditions are justified and accounted for in the experimental data because the distortion associated with **{{** **}}**

{{ }} does not appear to be accounted for in the test data. The staff assessment of the uncertainty, associated in part by test facility distortions, is provided in SER Section 5.4.1.4.2.1.4.

Conclusions regarding suitability of assessment datal

Based on the review described in this subsection, the staff is able to determine that (1) assessment data is independent of the data used to develop the evaluation model, and (2) experimental data has been accurately measured. However, the staff is unable to determine that (1) data has been collected over a test envelope that covers the evaluation model analysis envelope, or (2) tests are representative of prototypical conditions. Accordingly, the staff is unable to determine that the test data used for assessment is appropriate. The staff evaluation of the uncertainty, associated with gaps in DWO model assessment, is provided in SER Section 5.4.1.4.2.1.4.

<u>The applicant has not demonstrated that the evaluation model has the ability to predict DWO over the analysis envelope</u>

Evaluation model error is quantified through assessment against experimental data

By letter dated January 13, 2025 (ML25013A242), Item 13, the applicant summarizes the quantification of evaluation model error for the prediction of DWO onset. The applicant states that **{{**

}} The staff performed confirmatory analysis as part of the review using the data from letter dated January 13, 2025 (ML25013A242), Item 7, and obtained results that are consistent with the applicant's analysis. Specifically, the staff (1) confirmed that the data **{{**

}} and (2) calculated the same **{{**
}} as the applicant by using the estimation method described in Section 9.11, "Confidence intervals for unknown σ," of NUREG-1475, "Applying Statistics," Revision 1 (ML11102A076). Based on the information described in this paragraph, the staff finds that evaluation model error is quantified through assessment against experimental data because the applicant quantified the model error through comparison to experimental data and the staff was able to confirm the calculated values using standard statistical techniques.

Evaluation model error is not determined throughout the evaluation model analysis envelope

The staff identified that the error assessment, discussed above, used all the available DWO assessment data. The staff review of data coverage in SER subsection, "Assessment data does not fully cover the DWO analysis envelope," above, identified that data has not been collected over a test envelope that covers the evaluation model analysis envelope. Accordingly, the staff is unable to determine whether the error predictions in the evaluation model are valid throughout the enveloped range. The staff assessment of the uncertainty, associated in part by evaluation model error assessment not covering the NPM SG performance envelope, is provided in SER Section 5.4.1.4.2.1.4.

No sparse data regions or error trends were identified

The staff examined the assessment data provided in letter dated January 13, 2025 (ML25013A242), Item 7 for sparse data regions (regions in the test matrix where little or no data is available) and error trends by plotting the error as a function of power level, secondary side pressure, and inlet subcooling. The staff examination did not identify sparse data regions or error trends that required justification. Based on this examination, the staff finds that justification for sparse data regions is not required because no sparse data regions or error trends were identified. Gaps in data coverage due to the size of the test envelope is discussed in SER subsection, “Assessment data does not fully cover the DWO analysis envelope,” above.

Use of evaluation model is not restricted for usage within its test envelope

By letter dated January 13, 2025 (ML25013A221), the applicant clarifies that the approach temperature is not applied below an average coolant temperature of 271 ºC (520 ºF) or when the main steam is not superheated. Additionally, by letter dated January 13, 2025 (ML25013A227), the applicant clarifies that the approach temperature is not applied in these regions because **{{**

}} The staff finds this limitation on the approach temperature applicability reasonable because it limits the range of applicability for the approach temperature to that region which is supported by analysis. However, the staff is unable to determine that the approach temperature applicability range is supported by assessment data. Specifically, the staff is unable to determine that the evaluation model is restricted to use within its test envelope because (1) the approach temperature calculations performed with NRELAP5 generally use **{{**

}}, (2) test data collected does not cover the range of NPM SG operating conditions where NRELAP5 is used to assess DWO (see SER subsection, “Assessment data does not fully cover the DWO analysis envelope,” above), and (3) the staff is unable to determine that the tests used to obtain the assessment data are fully representative of prototypical conditions (see SER subsection, “Tests are not fully representative of prototypical conditions,” above). The staff assessment of the uncertainty, associated with gaps in model assessment, is provided in SER Section 5.4.1.4.2.1.4.

Conclusions regarding evaluation model predictive capability over the analysis envelope

The staff is able to determine that (1) evaluation model error is quantified through assessment against experimental data, and (2) the assessment data does not contain sparse data regions that require justification. However, the staff is unable to determine that (1) evaluation model error is determined through the analysis envelope, or that (2) the evaluation model is restricted to use within its test envelope. Accordingly, the staff finds that the evaluation model has demonstrated the ability to predict DWO over the test envelope, but the staff is unable to determine that the test envelope covers range needed for the NPM SG or that the evaluation model is restricted to use within its test envelope. The staff assessment of the uncertainty, associated with gaps in model assessment, is provided in SER Section 5.4.1.4.2.1.4.

<u>Conclusions regarding DWO evaluation model assessment</u>

Based on the staff’s review described in this section (SER Section 5.4.1.4.2.1.2.2) the staff is able to determine that several aspects of data collection and assessment are reasonable. Specifically, the assessment data collected is demonstrated to be independent and measured accurately, and the evaluation model assessment demonstrates the ability to predict DWO over the set of data collected. Accordingly, the staff finds that the evaluation model provides reasonable insight into the likelihood of DWO because the evaluation model demonstrated predictive capability over the data that was collected. However, the staff is unable to determine whether the experimental data is adequate and whether the evaluation model demonstrated the ability to predict DWO over the range of needed conditions.

5.4.1.4.2.1.2.3 Conclusions regarding DWO onset calculation capability

Based on the evaluation model assessment against available test data, the staff finds that the NRELAP5 evaluation model calculations provide reasonable insight into the likelihood of DWO because the assessment of the evaluation model demonstrated quantifiable predictive capability for the tests that were performed. However, the staff is unable to determine that DWO onset calculations, as performed using NRELAP5, provide reliable prediction of DWO onset over the applicable range of conditions for the NPM SG. Specifically, the staff is unable to approve the generic use of NRELAP5 evaluation model for the prediction of DWO onset in the NPM SG because (1) the assessment data collected does not cover the DWO analysis envelope, (2) test facility distortions **{{ }}** are not fully justified, therefore, the staff is unable to determine that the test provides fully representative conditions for the NPM SG, and (3) NRELAP5 calculations are performed outside the bounds of the test envelope. However, the staff recognizes that several aspects of experimental data collection are reasonable and that the NRELAP5 assessment indicated predictive capability for the set of data that was collected through quantification of prediction error.

The staff identifies that the gaps in evaluation model assessment causes uncertainty in the prediction of DWO onset for the NPM SG. Accordingly, the amount of margin to DWO provided by the approach temperature limit is uncertain. The staff is unable to quantifiably assess the magnitude of this uncertainty because the staff does not have a basis to support uncertainty quantification (i.e., applicable data or valid alternative analyses over the range of NPM SG operating conditions are not available). The risk associated with uncertainty in DWO onset prediction, attributed to the reasons discussed above, is discussed in SER Section 5.4.1.4.2.1.4.

5.4.1.4.2.1.3 Static instability coupling is precluded

By letter dated January 13, 2025 (ML25013A242), Item 14, the applicant states that they performed a calculation to analyze the susceptibility of the secondary side of the NPM to Ledinegg instability and that the results of this analysis demonstrate that the NPM is not susceptible to the Ledinegg instability because the NPM **{{ }}**. The staff did not perform a detailed technical review of the Ledinegg instability analysis because (1) analyses were performed using NRELAP5 which received staff approval of the thermal-fluids modeling for the NPM SG as described in the staff's SE for TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4 (ML24334A049), and (2) uncertainty associated with the analyses are acceptable based on the associated risk (see SER Section 5.4.1.4.2.1.4). Based on the information described in this paragraph, and the staff's uncertainty assessment in SER Section 5.4.1.4.2.1.2.4, the staff finds reasonable assurance that static instability coupling is precluded because the applicant's analysis determined that the NPM is not susceptible to Ledinegg instability and the uncertainty associated with this analysis are acceptable considering the risk.

5.4.1.4.2.1.4 Uncertainties in the prediction of DWO onset are reasonable considering the risk associated with DWO

The staff review of the approach temperature limit determined that the approach temperature limit provides margin to DWO with respect to DWO onset calculations (see SER Section 5.4.1.4.2.1.1), but the staff was unable to determine that the DWO onset calculations, as performed using NRELAP5, provide reliable prediction of DWO onset over the applicable range of conditions for the NPM SG (see SER Section 5.4.1.4.2.1.2). The staff review of the DWO onset calculation methods determined that the calculations provide reasonable insight into the likelihood of DWO, but quantifiable margins could not be determined due to gaps in evaluation model assessment (see SER Section 5.4.1.4.2.1.2). The gaps in evaluation model assessment result in uncertainties in the prediction of DWO onset over the range of thermal-hydraulic conditions applicable to the NPM SG. Accordingly, the amount of margin to DWO provided by the approach temperature limit is uncertain.

If exceeded, the uncertainty in margin provided by the approach temperature has the potential for the NPM SG to operate with DWO occurring for longer than the operator may have recognized and tracked. As a result, the tubes encountering DWO undergo thermal cycling and associated mechanical degradation leading to thermal fatigue, wear, and possible brittle fracture of the tubes. The tube fracture results in the loss of integrity of one fission product barrier, namely the potential loss of the reactor coolant pressure boundary integrity. To assess this risk, the staff considered (1) the NPM design defense-in-depth, (2) safety margin with respect to DWO, (3) DWO contribution to risk relative to the intent of the Commission's Safety Goal Policy Statement, and (4) performance measurement strategies.

<u>Consistent with defense-in-depth philosophy</u>

NPM SG tube degradation due to DWO would challenge the integrity of one fission product barrier, namely the potential loss of the reactor coolant pressure boundary integrity. Loss of tube integrity is analyzed in FSAR, Section 15.6.3, "Steam Generator Tube Failure (Thermal Hydraulic)," and discussed in FSAR, Section 19.1.4, "Safety Insights from the Internal Events Probabilistic Risk Assessment for Operations at Power." The staff review of the information, provided in SER Section 15.6.3, "Steam Generator Tube Rupture," and SER Section 19.1.4.4.2.1, "Non-LOCA Events," determined that (1) loss of a single train of DHRS due to steam generator tube failure is within the design basis of the NPM, and (2) for events that involve multiple steam generator tube failures and a resulting loss of both trains of the DHRS, the passive heat removal capability provided by the emergency core cooling system is sufficient to prevent core damage. Based on the information described above, the staff finds that, regarding uncertainty associated with DWO onset prediction, the NPM design is consistent with defense-in-depth philosophy because (1) NPM SGs are automatically isolated by safety-related isolation valves in the event of a steam generator tube rupture such that release of primary coolant to the main steam system could be terminated following the rupture, (2) non-safety-related secondary isolation valves can also isolate the NPM SGs as a backup, (3) there are no relief valves between the isolation valves and containment, and there are no relief valves outside containment that could result in a loss of inventory to containment, (4) ECCS is available to mitigate the event if both trains of DHRS are lost, and (5) CVCS can provide inventory makeup for most scenarios as an alternative to ECCS.

<u>Maintains sufficient safety margins</u>

The staff determined that (1) the approach temperature limit provides margin to predicted DWO onset (see SER Section 5.4.1.4.2.1.1) and likely provides margin to DWO (see SER Section 5.4.1.4.2.1.2), and (2) the design margin and associated inspection interval provide margin to NPM SG tube wear limits (see SER Section 3.9.2). Specifically, Table 3.9.1, "Summary of Design Transients," indicates that the NPM SG is designed to withstand 2840 days (approximately 7.8 years) of DWO over the 60-year plant lifetime. The applicant's proposed GTS in Section 5.5.4, "Steam Generator (SG) Program," states provisions for SG tube inspections that includes (1) inspect 100 percent of the tubes in each SG during the first refueling outage following initial startup or SG replacement, and (2) after the first refueling outage following SG installation, inspect 100 percent of the tubes in each SG at least every 72 effective full power months. In addition, COL Item 5.4-1 states that for at least the first NuScale module to undergo a refueling outage, at least 20 percent of the tubes will be inspected during each refueling outage over the 72 effective full-power months after the first refueling outage. More SG tube inspections may be required according to the inspection results and operational assessment performed as part of the SG Program. Based on the information provided in this paragraph, the staff finds that the NPM SG design maintains sufficient safety margins because SG tube lifetime fatigue for DWO is much lower than the applicable ASME design criteria and the tube wear evaluation is much longer than the inspection frequency.

<u>Risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement</u>

The staff evaluation of the risk associated with a steam generator tube failure is provided in SER Section 19.1, "Probabilistic Risk Assessment." The staff reviewed FSAR, Section 19.1, to determine if the estimated risk contribution would be significantly impacted from the uncertainty in predicting DWO onset in relation to the steam generator tube failure. The staff determined, in SER Section 19.1, that the prediction of DWO onset does not significantly impact the information provided in FSAR, Section 19.1, or the staff's evaluation of the success criteria, key assumptions, or sources of uncertainty for the probabilistic risk assessment (PRA) of the steam generator tube failure in SER Section 19.1. In addition, in SER Section 19.1.4.4.1, "Initiating Event Analysis," the staff determined that the applicant performed sensitivity studies that showed that the Core Damage Frequency (CDF) and Large Release Frequency (LRF), and risk insights, are relatively insensitive to the specific estimated initiating event frequency associated with a break in the steam generator tubes.

Since the uncertainty in predicting DWO onset does not significantly impact the success criteria, key assumptions, or sources of uncertainty for the steam generator tube failure, the staff finds that there is no significant impact to the NuScale US460 design-specific PRA and other PRA-related information in FSAR, Section 19.1, and that the estimated CDF and LRF, considering all hazards and all modes remains consistent with the Commission's CDF and LRF goals.

<u>Performance measurement strategies</u>

FSAR, Table 3.9.1, "Summary of Design Transients," indicates that the NPM SG is designed to withstand 2840 days (approximately 7.8 years) of DWO over the 60 year plant lifetime and the applicant's proposed GTS states in Section 5.5.4, "Steam Generator (SG) Program," that the provisions for steam generator tube inspections that includes (1) inspect 100 percent of the tubes in each steam generator during the first refueling outage following initial startup or steam generator replacement, and (2) after the first refueling outage following steam generator installation, inspect 100 percent of the tubes in each steam generator at least every 72 effective full power months. In addition, COL Item 5.4-1 states that for at least the first NuScale module to undergo a refueling outage, at least 20 percent of the tubes will be inspected during each refueling outage over the 72 effective full-power months after the first refueling outage. More SG tube inspections may be required according to the inspection results and operational assessment performed as part of the SG Program. The staff review determined that the NPM SG design with respect to fatigue margin, including the consideration of DWO, is acceptable in SER Section 3.9.1. Based on the design margin provided for the NPM SG providing a tube lifetime that is much longer than the inspection frequency, the staff finds that the applicant has acceptable performance measurement strategies.

<u>Conclusions on risk assessment associated with DWO onset prediction uncertainty</u>

Based on the staff evaluation of the risks associated with DWO, the staff finds that the uncertainty associated with the unquantified margin to DWO is acceptable because (1) the NPM design, regarding DWO, is consistent with defense-in-depth philosophy (see SER subsection “Consistent with defense-in-depth philosophy,” above), (2) the NPM design maintains sufficient safety margins (see SER subsection “Maintains sufficient safety margins,” above), (3) the risk associated with DWO is small and consistent with the intent of the Commission’s Safety Goal Policy Statement (see subsection “Risk is small and consistent with the intent of the Commission’s Safety Goal Policy Statement,” above), and (4) the applicant has acceptable performance measurement strategies (see SER subsection “Performance measurement strategies,” above).

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 ASME Code 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 ASME Code Class 1 and Class 2 components. FSAR, 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 to 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 RGs 1.43, 1.50, and 1.71. FSAR, Section 5.2.3.3.2, “Welding Control—Ferritic Materials,” describes how the applicant’s design meets these welding requirements of ferritic materials used for RCPB components. FSAR, Section 5.4.1.5, states that FSAR, 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 RGs 1.31, 1.34, 1.36, 1.44, and 1.71.

FSAR, Section 5.2.3.4, describes how the applicant’s design meets these requirements for use of austenitic stainless steel in pressure boundary applications. FSAR, Table 1.9-2, “Conformance with Regulatory Guides,” indicates the following:

- RG 1.34 is not applicable because electroslag welding is in accordance with RG 1.43. FSAR, Section 5.3.1.4 states, “In addition, electroslag welding processes are not utilized for joining materials. Cladding low alloy steel allows electroslag welding processes and complies with RG 1.43 requirements.” Also, FSAR, Section 5.2.3.3.2 states, “Stainless steel corrosion resistant weld overlay cladding of low alloy steel components conforms to the requirements of RG 1.43. Controls to limit underclad cracking of susceptible materials also conform to the requirements of RG 1.43.”

- RG 1.36 does not apply to FSAR, Sections 5.2 and 5.4, “Reactor Coolant System Component and Subsystem Design,” because the applicant’s design does not use nonmetallic thermal insulation on RCPB or CNV components.

- RG 1.44 recommends applying RG 1.37; however, RG 1.37 was withdrawn by the NRC, therefore, the applicant partially conforms to RG. 1.44.

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 the 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.

FSAR, Section 5.2.3.2.1, describes the control of the primary water chemistry, and FSAR, Section 10.3.5, “Water Chemistry,” describes the secondary water quality control program. Control of the primary water chemistry is based on the EPRI Guidelines (*see* SER Section 5.2.3.2), while control of the secondary water chemistry is based on EPRI’s “Pressurized Water Reactor Secondary Water Chemistry Guidelines,” and NEI 97-06, “Steam Generator Program Guidelines,” Revision 3, issued January 2011. In addition, the proposed secondary water chemistry program conforms to the latest revision of the Standard Technical Specifications (STS) (e.g., NUREG-1431, “Standard Technical Specifications – Westinghouse Plants,” issued September 2021). The staff has determined that these EPRI guidelines are acceptable for primary and secondary water chemistry control for the applicant’s 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 5.4.4.4.1, the design nominal wall thickness includes a lifetime degradation allowance of 0.25 mm (0.010 inch). FSAR, 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. FSAR, Section 5.4.1.3.1, also states that the “This degradation allowance also includes margin for SG tube wall thickness manufacturing tolerances, including wall thinning due to tube bending.” Based on the description in FSAR, 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 TT SG tubes with the proposed primary and secondary water chemistries, the staff determined the proposed degradation allowance is acceptable. In addition, the staff notes that the Steam Generator Program requires the thickness of all inservice tubes to be measured periodically.

FSAR, Section 5.4.1.5, states that FSAR, Section 5.2.3.4.2, “Cleaning and Contamination Protection Procedures,” contains the cleaning and cleanliness controls for the SGs. FSAR, Section 5.2.3.4.2 states the cleaning, handling, storage, and shipping of RCPB components comply with ASME Code, Subparagraph NCA-4134.13 and ASME NQA-1-2015. FSAR, Table 1.9-2, notes that FSAR, Section 5.2, partially conforms to RG 1.28, Revision 5. Specifically, FSAR, Table 1.9-2 states that NuScale conforms with RG 1.28, Revision 4 and partially conforms with RG 1.28, Revision 5 with regards to technical guides for electronic records, basis for laboratory calibration requirements, and lead auditor qualifications. Because RG 1.28, Revision 4 and Revision 5 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 applicant’s design) access for tools to inspect and remove corrosion products and foreign objects that may affect tube integrity. The staff reviewed the information in FSAR, Section 5.4.1.2. The applicant’s design includes four steam plenum inspection/access ports that provide openings for access to the top of the tube bundle, steam tubesheets, and steam plenums. The design also includes four feed plenum inspection/access ports that provide openings for access to the bottom of the tube bundle, feed tubesheets, feed plenums, and inlet flow restrictors. FSAR, Section 5.4.1.2 states, in part, “the SG design permits periodic inspection and testing of critical areas and features to assess their structural and pressure boundary integrity when the NPM is disassembled for refueling.” In addition, FSAR, Section 5.4.1.2, states that the internal (secondary) and external (primary) sides of the tubesheets can be accessed for inspection and removal of foreign objects.

The staff determined that this level of secondary 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

FSAR, Section 14.3.4, “Treatment of Module-Specific and Shared Structures, Systems, and Components in Inspections, Tests, Analyses, and Acceptance Criteria,” evaluates ITAAC associated with this section.

5.4.1.6 Steam Generator Program

The staff reviewed FSAR, Sections 5.4.1 and 5.4.1.6, in accordance with 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 also reviewed FSAR, Sections 5.4.1 and 5.4.1.6, and assessed how the proposed SG Program related to10 CFR 50.36 (incorporating the SG program into the TS), 50.55a(g) (concerning PSI and ISI requirements), and 50.65 (requirements for monitoring the effectiveness of maintenance at nuclear power plants). The staff also reviewed 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, which are related to the SG program.

FSAR, Section 5.4.1.1, states that the SG system components allow performance of the ISI requirements of ASME Code, Section XI, and the secondary sides of the SGs permit access for SG inspections. FSAR, 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 because of the SG design. The internal surface of the SG tubes can be inspected over their entire length with nondestructive evaluation (NDE) methods and techniques capable of detecting the types of degradation that may occur over the life of the tubes. In the response to Audit Issue A-5.4.1-1 dated August 2, 2024 (ML24215A117), the applicant stated that "The licensee of a US460 design plant will develop and qualify the inspection technique capable of detecting expected degradation mechanisms." FSAR, 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 internal (secondary) and external (primary) sides of the tubesheets can be accessed for inspection and removal of foreign objects. The staff finds the applicant's SG design meets the acceptance criteria in 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.

FSAR, Section 5.4.1.6, states that the SG program is based on NEI 97-06 and is documented in TS. 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). FSAR, 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 DSRS Section 5.4.2.2.

As discussed in SER Section 5.4.1.4.5, the primary and secondary water chemistry programs are controlled in accordance with industry guidelines that conform to the acceptance criteria in 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.

FSAR, Section 5.4.1.6, states that the plant TS documents the SG program, which is part of the overall ISI program. In accordance with DSRS Section 5.4.2.2, the staff reviewed the technical requirements of the SG program in the applicant's proposed Generic Technical Specifications (GTS). Specifically, the staff reviewed the proposed criteria for assessing the as-found condition of an 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 and TSTF-577 as the basis for its review of the applicant's GTS associated with the SG program. 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, and Babcock and Wilcox plants).

The staff observed differences between the applicant's GTS and the STS because of the design of the NPM and the SGs such as operating mode definitions, completion times for required actions, frequency of SRs, and definition of "unit" as an NPM. Some of the differences are found throughout the GTS, and the staff reviews them further in SER Chapter 16. Some nonstandard TS wording is related to the applicant's design. The applicant'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 applicant's design have the higher pressure on the outside. The word "failure" applies to burst, collapse, and buckling of the SG tubes. The staff finds it acceptable that the applicant's proposed GTS Subsection 5.5.4.d.1 uses "initial startup or SG replacement" instead of "SG installation," as stated in corresponding STS Subsection 5.5.8.d.1, because the applicant's SGs are fabricated integral to the upper module. The applicant'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 applicant's SGs.

Although Revision 5 of NUREG-1431 allows the use of "tube repair," "plugging [or repair]," and "plugged [or repaired]" the applicant's proposed GTS use "tube plugging criteria" and "plugged," respectively. For example, TS Subsection 5.5.8.c in Revision 5 of NUREG-1431 is, "Provisions for SG tube plugging [or repair] criteria." The applicant 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 the term "plugging criteria" is consistent with the STS, and NuScale did not propose any bracketed alternative plugging or repair options.

The TS include structural integrity performance criteria (SIPC) to ensure tubes maintain structural design margins against failure by burst, buckle, and collapse under design basis and accident loads. The SG Program in GTS Subsection 5.5.4.b.1 contains the SIPC for NuScale in the form of safety factors against failure over the full range of normal operating conditions (including startup, operation in the power range, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes a safety factor of 1.4 against failure applied to the design basis accident primary-to-secondary pressure differentials, and safety factors of 1.2 on combined primary loads and 1.0 on axial secondary loads that are not normal steady state full power operating or design basis accident loads but are determined to contribute significantly to failure. These factors are consistent with the corresponding SIPC in the STS.

For failure by collapse or buckling under normal steady state full power operation, which is the most likely mode of structural failure for the NuScale design given the higher external pressure (reactor coolant on the outside of the SG tubes), the GTS propose a factor of greater than 2.0 (two times the normal operating pressure differential, or 2xNOPD). The applicant justified this value in the response to RAI No. 10189, NRC Question No. 5.4.1.6.1-1, (ML24354A160) dated 12/19/2024, by referencing several parts of the ASME Code applicable to the NuScale tube design or applicable to the design of externally pressurized cylinders in general. In particular, the applicant stated that the minimum SG tube wall thickness is based on compressive stress from external pressure loading using ASME Code Case N-759-2, which specifies stress reduction factors for design in the range 1.67 to 2.0 applied to the failure stress calculated for externally pressurized cylinders. ASME Code Case N-759-2 is an alternative to the rules of ASME Code, Section III, Division 1, Paragraph NB-3133, “Components Under External Pressure,” and is approved by the NRC with no exceptions in RG 1.84.

In addition to ASME Code Case N-759-2, the applicant noted the specified safety factor is also 1.67-2.0 in ASME Code, Section VIII, Division 2, Part 4, 4.4, “Design of Shells Under External Pressure and Allowable Compressive Stresses.” ASME Code, Section VIII, Division 2, Part 5, 5.4.3, “Protection Against Collapse from Buckling: Buckling Analysis – Method B,” requires a safety factor of 1.67, and ASME Code, Section III, Mandatory Appendix XIII, XIII-3200, has an option for plastic analysis with a safety factor of 1.5 for Service Level A and Service Level B. The staff noted that the applicant takes no exceptions in designing the SG tubes for external pressure loading using ASME Code Case N-759-2. The staff finds it acceptable for the applicant to specify a safety factor of greater than 2.0 for collapse and for buckling based on consistency with the safety factor range specified in NRC-approved ASME Code Case N-759-2 and based on a safety factor of 2.0 being consistent with Section VIII rules for external pressure loading of cylindrical pressure vessels.

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 l/m). As noted in NEI 97-06, the primary-to-secondary leakage rate assumed in accident analyses can vary. The applicant selected 150 gallons per day (0.568 cubic meter per day) for the AIL performance criterion, which is equivalent to its operational leakage performance criterion. 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. Because the work to support the longest inspection interval in Revision 5 of the STS is ongoing, the applicant proposed the inspection interval (72 EFPM) for SGs with Alloy 690 TT tubing in Revision 4 of the STS. In addition, COL Item 5.4-1 states that for at least the first NuScale module to undergo a refueling outage, at least 20 percent of the tubes will be inspected during each refueling outage over the 72 effective full-power months after the first refueling outage. More SG tube inspections may be required according to the inspection results and operational assessment performed as part of the SG Program. The staff finds the SG tube inspection interval in the applicant’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 applicant's 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 SGs will be monitored- through the SG program and 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 the applicant's proposed GTS are appropriately consistent with the STS and TSTF-577, considering the differences in the applicant's design. 10 CFR 50.36(c)(5) requires that TS include items in the category of "administrative controls" which are "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The staff's review determined that the SG program is necessary to assure operation of the facility in a safe manner and therefore must be included in the administrative controls in the TS to meet 50.36(c)(5). The staff confirmed that the proposed GTS include the program and thus satisfy 10 CFR 50.36(c)(5). Further, 10 CFR 50.36(c)(3) requires TS to include "surveillance requirements" that are "requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The staff's review determined that the SG program assures that the necessary quality of the systems and components are maintained that the facility will operate within the SLs, and that the SG program assures that the LCOs will be met. The staff confirmed that the proposed GTS LCO subsection 3.4.9, "Steam Generator (SG) Tube Integrity," includes the program in the SRs where the program is necessary. Accordingly, the staff concluded that the applicant met 10 CFR 50.36(c)(3) and (5) relative to including the SG program in GTS. SER Section 16.1 further discusses the staff's review of the applicant's proposed TS and bases.

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. FSAR, Section 5.4.1.2, states that individual SG tubes may be plugged and, if necessary, stabilized to prevent adverse interaction with non-plugged tubes. FSAR, Section 5.4.1.4, states, "Tubes with flaws that could potentially compromise tube integrity before the performance of the first ISI, and tubes with indications that could affect future inspectability of the tube, are also be plugged." FSAR, 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 two times normal operating pressure differential (2xNOPD).

Because the loads and degradation mechanisms applicable to the currently operating SGs are not directly applicable to the applicant's design, a design-specific determination was necessary. A significant difference is that, under normal operating conditions, the tubes in the applicant's 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.

The applicant 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.

The applicant's determination of the generic SG tube plugging criterion value followed, as applicable, the acceptance criteria in RG 1.121, NEI 97-06, and the EPRI Steam Generator Integrity Assessment Guidelines. RG 1.121 describes a method for establishing the amount of tube degradation beyond which tubes should be removed from service. To address the external pressure loading of the tubes in the NuScale design, the applicant partially conforms to RG 1.121. FSAR, Table 1.9-2, identifies the partial conformance as an alternate loading condition. The result is the safety factor of 2.0 applied to tube collapse and buckling, which is described above in this section of the SER. Using this safety factor for collapse and for buckling under normal operating conditions, and safety factors consistent with the STS for accidents and other loading conditions, the applicant performed a finite element analysis (FEA) to evaluate the proposed 40 percent through wall plugging criterion. The applicant described the FEA supporting the proposed 40 percent through wall plugging criterion in the response to RAI No. 10189, NRC Question No. 5.4.1.6.1-1. The evaluation considered the tube material properties, tube and tube support geometry, design and operating conditions, loading conditions, and postulated tube degradation.

The applicant used the FEA to calculate failure pressure for tubes thinned by wear over a finite length from contact with the intrados surface against the adjacent tube support. The evaluation considered a **{{ }}**. The FEA was not applied to cracking of the tubes for the US460 design because the analyses performed for the US600 showed cracking was not limiting because tensile stress potentially generated by seismic loading was not greater than the compressive stress. The analysis found that the limiting loading criterion is the normal operating pressure differential with a safety factor of 2.0. The applicant concluded this criterion was met, with margin, for a **{{ }}** and therefore the 40 percent plugging criterion is acceptable with respect to the RG 1.121 criteria.

The staff evaluated the proposed tube plugging criterion with respect to RG 1.121, which provides guidance to addressing tube degradation, loading conditions, and safety margins to determine a minimum acceptable wall thickness. The staff reviewed the inputs and assumptions in the applicant's FEA, such as the tube geometry, material properties, temperature and pressure, loading conditions, and forms of degradation considered. The staff found the inputs were consistent with the FSAR and assumptions were adequately justified in the analysis. The staff noted that the inputs included the maximum amount of tube ovality allowed in the NuScale design specification, which the applicant identified in the response to RAI No. 10189, NRC Question No. 5.4.1.6.1-1. The staff also estimated collapse pressure for Alloy 690 tubing with the same dimensions as the NuScale tubing using the elevated temperature yield stress of Alloy 690 and the relationship between tube degradation and collapse pressure from collapse tests performed on Alloy 600 SG tubing with uniform thinning (e.g., NUREG/CR-0718 and NUREG/CR-2336). The staff's estimates of failure pressure indicated the 2xNOPD criterion could be met for 40 percent through-wall uniform thinning around the full tube circumference without restriction on the length of the thinned area.

In summary, the staff's evaluation indicates that an SG tube plugging criterion of 40 percent through-wall is a reasonable preliminary value for the type of tube degradation the applicant considers most likely (i.e., wear from an adjacent tube support). The applicant's evaluation did not account for thickness measurement uncertainty during inservice inspections, and some assumptions in the evaluation require verification if used by a COL applicant. Enclosing the plugging criterion value in brackets in the proposed GTS requires a COL applicant to confirm that this value is appropriate or propose and justify a different value.

FSAR, Section 5.4.1.1, states that the SG system components allow performance of the ASME Code, Section XI, ISI requirements, including the ASME Code, Section III, PSI requirements. FSAR, 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. FSAR, Section 5.4.1.4, 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 permits comparisons to the results of the ISIs expected to be performed to satisfy plant TS for SG tube inspections. FSAR, 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 applicant's SG design meets the acceptance criteria in 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 applicant's 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 (primary for this design) for inspection, cleaning, and evaluation of conditions such as loose parts. On this basis, the staff determined that the design of the applicant's SGs is acceptable as it relates to providing access to allow ISIs.

As discussed above, in addition to determining the SGs are designed to be accessible for inspection, the staff determined that the proposed SG program is properly incorporated into the TS, meets the requirements of ASME Code, Section XI, and provides for monitoring the performance of the SG tubes against goals such that there is reasonable assurance the tubes will be capable of fulfilling their intended function. Therefore, the requirements of GDC 32, 10 CFR 50.36, 10 CFR 50.55a(g), and 10 CFR 50.65 are met with respect to the SG program.

5.4.1.7 Combined License Information Items

SER Table 5.4.4-3 lists COL information item numbers and descriptions related to SG materials and design, and implementation of an SG program from FSAR, Chapter 1, Table 1.8-1.

Table 5.4.4-3 NuScale SG Materials and Design and SG Program Implementation Related COL Information Items in FSAR

Item No.	Description	FSAR Section
COL Item 5.4-1	An applicant that references the NuScale Power Plant US460 standard design 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 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. The Steam Generator Program for the NuScale Power Plant US460 will require 100 percent tube inspections at the first refueling outage. In addition to other requirements and inspections, the Steam Generator Program for the first module to undergo a refueling outage will require at least 20 percent tube inspections during each subsequent refueling outage for the first 72 effective full power months after the first refueling outage. Subsequent applicants that reference the Nuscale Power Plant US460 design shall provide justification that the results of the first module's inspections are applicable to the subsequent modules in order to demonstrate that these additional inspection requirements are not applicable to the subsequent modules.	5.4.1.6.1
COL Item 13.4-1	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific information, including implementation milestones, for Operational Programs: • Inservice Inspection Programs (Sections 5.2, 5.4, and 6.6) • Inservice Testing Programs (Section 3.9 and Section 5.2) • Environmental Qualification Program (Section 3.11) • Preservice Inspection Program (Section 5.2 and Section 5.4) • Preservice Testing Program (Section 3.9.6 and Section 5.2) • Containment Leakage Rate Testing Program (Section 6.2) • Fire Protection Program (Section 9.5.1) • Process and Effluent Monitoring and Sampling Program (Section 11.5) • Radiation Protection Program (Section 12.5) • Non-Licensed Plant Staff Training Program (Section 13.2) • Reactor Operator Training Program (Section 13.2) • Reactor Operator Requalification Program (Section 13.2) • Emergency Planning (Section 13.3) • Process Control Program (Section 11.4) • Security (Section 13.6) • Quality Assurance Program (Section 17.5) • Maintenance Rule (Section 17.6)	13.4

Item No.	Description	FSAR Section
	• Initial Test Program (Section 14.2)	

COL Item 5.4-1 directs a COL applicant referencing the applicant's SDAA to develop and implement an SG program based on the latest revision of NEI 97-06. In addition, COL Item 5.4-1 directs a COL applicant referencing the applicant's SDAA to (1) inspect at least 20 percent of the tubes during each refueling outage over the 72 effective full-power months after the first refueling outage, for the first module to undergo a refueling outage, or (2) justify why the additional inspections are not applicable. COL Item 5.4-1 also includes the specific elements that a COL applicant should include in the SG program. COL Item 13.4-1 directs 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 appropriate because they are consistent with DSRS Section 5.4.2.2, 10 CFR 50.55a, and NEI 97-06 for individual licensees to develop an SG program and SG PSI and ISI operational programs.

5.4.1.8 Conclusion

On the basis of its review of FSAR, Section 5.4.1, the staff concludes that the applicant's SG materials satisfy the acceptance criteria for materials selection, design, fabrication, compatibility with the service environments, and shell side (primary for this design) accessibility. The staff further concludes that the 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. The staff also concludes, on the basis of its review of FSAR, Section 5.4.1.6, that the applicant's SG program 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 applicant's 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 (wear at tube supports, including wear at tube supports due to DWO). 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.

5.4.2 Reactor Coolant System Piping

5.4.2.1 Introduction

The RCS piping and connected piping that penetrate the RCS form the RCPB and include the Pressurizer (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, the review scope of SRP Section 5.4, "Reactor Coolant System Component and Subsystem Design," includes this topic. Details about the RCS piping, including the staff's evaluation and conclusion on the RCS piping 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, 5.4.2, 6.1, 6.3, 6.6 and 10.3.6.

5.4.2.2 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.

5.4.3 Decay Heat Removal System

5.4.3.1 Introduction

In traditional light-water reactor (LWR) designs, residual heat removal (RHR) systems are used to cool the RCS following a shutdown. In the applicant's design, safety-related RHR following accidents is accomplished using the passive DHRS, 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 UHS) to remove heat rather than pumping feedwater into the SGs. This function is important for both non-LOCA and LOCA events when normal secondary-side cooling is unavailable to bring the plant to safe-shutdown conditions.

5.4.3.2 Summary of Application

FSAR Section 5.4.3, "Decay Heat Removal System," describes the functions of the DHRS. The DHRS is safety-related and designed to ensure there is passive cooling of the RCS after an initiating event without challenging the RCPB integrity or uncovering the core and reduce RCS temperature below 215.6 degrees C (420 degrees F) within 36 hours. The DHRS has two trains, each containing 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 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. Each train has an orifice located on the common line before the actuation valves to moderate flow during operation. FSAR Figure 5.4-7, "Steam Generator Simplified Diagram," and Figure 5.4-8, "Decay Heat Removal System Simplified Diagram," display DHRS layout in simplified diagrams.

The DHRS is normally isolated and actuates on a collection of signals specified in FSAR Table 7.1-4, "Engineered Safety Feature Actuation System Functions." These signals, in addition to actuating the DHRS valves by removing power from the trip solenoids, also initiate closure 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.

FSAR Table 5.4-5, “Decay Heat Removal System Design Data,” documents the design parameters for the DHRS. FSAR 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. FSAR Figures 5.4-9 through 5.4-12, 5.4-14 and 5.4-15 show the thermal-hydraulic performance of the DHRS under various conditions and inventory configurations.

ITAAC:

SDAA Part 8, Section 2.1, describes the NPM and associated systems, one of which is the DHRS. SDAA Part 8, Table 2.1-3, 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. SDAA Part 8, Table 2.1-4, provides an inventory of the mechanical equipment for the DHRS; this is limited to the four actuation valves and the two condensers. SDAA Part 8, Table 2.1-5, similarly inventories the electrical equipment, which includes the DHRS actuation valves; these valves are also considered for equipment qualification in SDAA Part 8, Table 2.4-1, which also lists the instrumentation required to monitor the DHRS. SDAA Part 8, Table 2.1-1 items 15 and 20 and Table 2.4-1 item 8, contain the specific ITAAC associated with DHRS. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: TS for the DHRS appear in SDAA Part 4, “US460 Generic Technical Specifications,” (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 Section 3.7.1, “Main Steam Isolation Valves (MSIVs),” and TS Section 3.7.2, “Feedwater Isolation,” and the reactor pool level and 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 FSAR, Section 5.4.3.

5.4.3.3 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

- PDC 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 UHS to remove heat from the reactor module or DHRS

- GDC 45, “Inspection of Cooling Water System,” as it relates to the ability to inspect the DHRS as it interfaces with the reactor pool

- GDC 46, “Testing of Cooling Water System,” as it relates to the ability to test the DHRS

- 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 the applicant, 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 the applicant has proposed a principal design criterion (PDC) rather than a GDC for Criteria 34 and 44. The PDC proposed by applicant are functionally identical to the GDC, with the exception of the discussion related to electric power. SER Chapter 8 discusses the applicant’s reliance on electric power and the related exemption to GDC 17, “Electric Power Systems.”

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

Additionally, the NuScale US460 design (NPM-20) credits the DHRS to support the mitigation of a LOCA, including containment response. Therefore, the staff’s evaluation documented in this Section also includes the review of the DHRS to satisfy the requirements of PDC 35, 10 CFR

50.46, and 10 CFR Part 50 Appendix K. The staff's review of the DHRS with respect to the containment response is documented in Section 6.2.2 of this SER.

5.4.3.4 Technical Evaluation

The DHRS is one of a number of areas for which the applicant's design uses a passive system to satisfy safety requirements. The DHRS is credited for events in which secondary-side cooling (which is not safety-related) is not available and includes both LOCA and non-LOCA events; the main condenser and FWS provide routine RHR for the applicant's 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 applicant's 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. FSAR Table 5.4-9. "Classification of Structures, Systems, and Components," classifies the DHRS components as seismic Category I and therefore 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 applicant's 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 (ML24211A09). In its design specifications, the applicant outlined the limiting parameters for the equipment in the DHRS. Much of the piping and the passive condensers 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; FSAR Table 3C-1, "Environmental Qualification Zones," identifies this area as a harsh environment and denotes this as environmental qualification (EQ) Zone RXBP-1. FSAR Figure 3C-5, "Top of Module HELB Composite and Bounding Temperature Profile (Zone RXBP-1)," 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 FSAR Table 3C-7, "Design Basis Event Environmental Conditions," and Figure 3C-5. In addition, TR-121507, "Pipe Rupture Hazards Analysis," Figure A-1 and Figure A-3 show the DHRS piping that is classified as not subject to GDC 4 postulated pipe ruptures. See SER Section 3.6 for the staff's review of GDC 4 dynamic effects and exclusions of their consideration.

The DHRS has no vent pathways available with the exception of thermal relief valves on the feedwater piping providing overpressure protection when the steam generator is water solid and containment is isolated.. The components are designed to RCS conditions, 650 degrees F and 2200 psia. 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 FSAR, 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.

Therefore, because of the considerations mentioned above, the staff finds the DHRS conforms to the requirements associated with GDC 4 with regard to dynamic effects and EQ. Dynamic effects related to protection of the DHRS SSCs are discussed in Sections 3.5 and 3.6 of this SER.

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 FSAR, Section 9.3.4, “Chemical and Volume Control System,” and evaluated by the staff in Section 9.3.4 of this SER. In the event of a tube failure, the DHRS is designed to the same system pressure as the RCS, thus 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 FSAR Chapter 8. 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.

The primary function of the DHRS is providing part of the RHR functions for design-basis events, along with the containment isolations of the MSIVs, the main steam isolation bypass valves, and the FIVs, and thereby satisfies a portion of the requirements associated with PDC 34 and PDC 35. To evaluate the ability of the DHRS to meet decay heat removal requirements, the staff reviewed the system description, design layout, heat removal capacity, performance capabilities, instrumentation, and all associated documentation, as detailed below.

As discussed in FSAR 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 active failure provision for the DHRS for any given train. The DHRS is also dependent on the MSIVs and FWIVs to create a closed loop, but a single active failure of a MSIV can disable the associated train of the DHRS. Therefore, the design includes nonsafety-related backup isolation components in the event a MSIV fails to close. These backup components are reviewed in SER Section 3.9.6 and Section 10.3. Additionally, there are initiating events that also render a single train of DHRS inoperable. This scenario and the staff's review of DHRS cooling performance to satisfy Commission policy is discussed below.

As part of the review of the DHRS, the NRC staff audited the applicant's design documentation related to the DHRS (ML24264A049). FSAR Table 5.4-5, details the parameters of the system, including information on the condenser characteristics and fouling factor considered in the analyses documented in, FSAR Figures 5.4-9 through 5.4-12, 5.4-14 and 5.4-15 .

FSAR Table 5.4-8, provides a 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; the staff confirmed that, as the system is designed such that no single component failure results in inoperability of the system to perform its function because one train of DHRS remains available for conditions requiring the DHRS to cool down.

FSAR Table 14.2-41, "Test #41 Decay Heat Removal System," provides the preoperational tests for the DHRS. The staff finds that the inclusion of a first-plant-only test using the module heatup- system, which will compare the tested performance to an analysis performed using NRELAP, is sufficient to demonstrate that the as-built DHRS performance will be 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. In addition, Table 14.2-98, "Test #98 Reactor Trip from 100 Percent Power," is a startup test that is performed for each module and tests DHRS performance and includes effects of the initial conditions and component performance that affect DHRS inventory. Test #98 includes DHRS actuation and acceptance criteria for acceptable RCS cooling and will demonstrate appropriate water inventory of the system through effective DHRS performance accounting for secondary isolation timing, steam generator inventory, and feedwater pump coast down representative of plant characteristics (ML24346A204). The staff has reviewed the ITAAC related to the DHRS against the requirements of 10 CFR 52.47(b)(1 SDAA Part 8, Table 2.1-1 items 15 and 20 and Table 2.4-1 item 8 specifically address the performance of DHRS components. The ASME report ITAAC, SDAA Part 8, Table 2.1-1, Items 1 through 5, are also applicable to the DHRS. These ITAAC are evaluated in Section 14.3 of this SER.

Based on the information in the FSAR, including the piping and instrumentation diagram in FSAR Figure 5.4-8; system description; audited material (ML24211A09); and proposed testing, the staff determined there is sufficient information to provide assurance that the system will operate in accordance with the analysis assumptions made by the applicant. In FSAR Section 5.4.3.3.2, "System Noncondensable Gas," the applicant noted 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 DHRS 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. FSAR Section 5.4.3.3.2 states that the DHRS performance analysis evaluates a conservative mass of noncondensable gas of 0.73 lb per train. This value is based on the piping volume between the actuation valves and the placement of the level sensors where noncondensable gas can accumulate before a warning is provided to the operators. The level sensors are discussed further below. The staff observed during its audit of the DHRS performance evaluation that **{{**

}} the applicant utilized modeling simplification and assumed the entire first cell in the DHRS steam piping downstream of the DHRS actuation valves is filled with noncondensable gas. This assumption bounds the mass prescribed in FSAR 5.4.3.3.2 by approximately 1.7 times and is present in the DHRS cases for the low- and high-inventory cases documented in FSAR Figures 5.4-11, 5.4-12, 5.4-14, and 5.4-15. The staff believes these are reasonably conservative assumptions and, in conjunction with the ASME design report required as part of ITAAC Item 02.01.01 to verify that 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.

FSAR Section 5.4.3.3 states the DHRS provides the passive, safety-related, and redundant capability to cool the reactor core and coolant to safe shutdown conditions. In implementing PDC 34 requirements for active plant designs, the staff specified in RG 1.139 "Guidance for Residual Heat Removal," and Branch Technical Position (BTP) RSB 5-1 the conditions for cold shutdown (93.3 °C (200 °F) for a PWR) using only safety-related systems within 36 hours. Because the applicant's design, and other passive designs, do 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 safe shutdown in lieu of cold shutdown and defined it as an RCS temperature of 420 degrees F (215.6 degrees C), within 36 hours following a transient. The Commission approved these alternative temperature criteria in the staff requirements memorandum to SECY94084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994.

The DHRS performance Figures 5.4-9 through 5.4-12, 5.4-14, and 5.4-15 assessed a variety cases, including nominal and off-nominal, and single train and two-train cooldown conditions. During the audit (ML24211A09), the staff reviewed the inputs to the calculations underlying these figures and noted that the applicant used a value of approximately 100 °F for the nominal cases, as the initial condition for the reactor pool temperature, in addition to nominal values for all other parameters. The off-nominal cases used the TS maximum of 120 °F, as the initial condition for the reactor pool temperature, and adversely biased parameters for initial core power, SG and DHRS fouling, SG tube plugging, noncondensable gas, and MSIV leakage, and performed assessments of both high and low inventory cases. Of the cases presented, the high-inventory, low DHRS heat transfer case (FSAR Figure 5.4-11, "Primary Coolant Temperature with Decay Heat Removal System Two Trains High Inventory") is the most limiting two-train case from an RCS temperature perspective, and demonstrates the RCS is below 420°F within 36 hours. The staff notes that FSAR Figure 5.4-14, "Primary Coolant Temperature with Decay Heat Removal System One Train High Inventory," presents the most limiting case (high-inventory, low DHRS heat transfer, and single train conditions). The results indicate that under single train off-nominal conditions the DHRS is not capable of bringing the RCS to a safe shutdown condition as defined in SECY-94-084 (420°F within 36 hours). The staff considers deviation from the safe shutdown criteria specified in SECY-94-084 in this case to be reasonable given (1) initiating events that breach the closed system boundary of the DHRS was not envisioned by staff and Commission when the safe shutdown condition was defined for passive designs because previous passive RHR systems were connected to the RCS, thus resulting in any breach of the system being categorized as a LOCA and the policy is only applicable to non-LOCA events, and (2) FSAR Chapter 15 transient analysis demonstrates the specified acceptable fuel design limits are maintained for all design-basis events, including accidents, that could result in the loss of one train of the DHRS, thus satisfying a primary element of PDC 34. The DHRS performance assessment documented in FSAR Section 5.4.3 does not directly parallel design basis events documented in FSAR Chapter 15 but is representative of the limiting events and is discussed further below.

The applicant calculated the performance curves using NRELAP5. The staff reviewed the applicability of the code for performing analyses of this type and found it acceptable, as documented in the staff SERs for TR-0516-49416-P-A, “Non-Loss-of-Coolant Accident Evaluation Model,” Revision 4 (ML24334A049), TR-0516-49422-P-A, “Loss-of-Coolant Accident Evaluation Model,” Revision 3 (ML24312A002), and TR-124587-P-A, “Extended Passive Cooling and Reactivity Control Methodology,” Revision 0 (ML24355A062), and Section 15.0.5 of this SER.

The limiting Chapter 15 transients for the performance of the DHRS is a steam line break. SER Section 15.1.5 and the FSAR highlight representative performance parameters for the steam line break event. As a result of the break, one of the DHRS trains is disabled, and the remaining DHRS train provides core cooling without exceeding the transient acceptance criteria. Nevertheless, adequate DHRS heat removal capability remains available.

For the DHRS transients discussed above, between the conservative off-nominal conditions shown in FSAR Figures 5.4-11, 5.4-12, 5.4-14 and 5.4-15, the feedwater line break, and the increase in feedwater flow, 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 PDC 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 further below.

The DHRS for the US460 design is also credited for LOCA events**.** The DHRS provides decay heat removal during the initial blowdown period of a LOCA. DHRS cooling is most significant for smaller breaks, prior to ECCS valve opening, where RPV level drops slowly allowing more heat to be transferred through the SGs to the DHRS heat exchangers. See Section 6.2.1 of this report for the performance of the DHRS with respect to containment response and Section 15.6.5 of this report for the staff’s regulatory evaluation of the core cooling response following a LOCA.

FSAR 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 steamline, indicates that the system is filled.

Further instrumentation for the DHRS consists of DHRS actuation valve position and accumulator pressure for valve operability, condensate temperature and pressure indication in the condenser header, and steam pressure indication. Each DHRS line has four sensors for steam pressure upstream of the actuation valves, and two temperature sensors and three pressure sensors downstream of the condenser (as shown in FSAR Figure 5.4-8). No flow rate instrumentation exists; instead, a combination of the temperature and pressure sensors at the outlet of the condenser and the RCS pressure and temperature instrumentation indicate DHRS performance. This instrumentation 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 PDC 19, “Control Room,” as applied to the DHRS.

GTS SR 3.5.2.2 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. GTS 3.5.2 also includes SRs associated with the DHRS actuation valves which includes valve accumulator pressure, actuation to the open position, and actuation response time. These valves are required to be operable in order for the DHRS to be operable. As discussed above, the MSIVs and feedwater isolation valves must also be operable for the DHRS to be operable. These components have GTS associated with them, as discussed in GTS Sections 3.7.1 and 3.7.2, respectively.

Ultimately, in the absence of all non-safety-related systems and operator actions, the decay heat removal function will transfer from DHRS to the ECCS 8 hours after a reactor trip. Loss of the low voltage ac electrical distribution system and subsequent reactor trip would be the result of a station blackout event. SER Section 8.4 further discusses this scenario. For the performance of the DHRS before the 8-hour point, the station blackout scenario is bounded by the single-train cases shown in FSAR Section 5.4.3, and the feedwater line break inside the containment, as discussed in FSAR, Section 15.2.8.

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

Under conditions of extended DHRS heat removal the RCS mixture level may drop below the top of the riser and can potentially interrupt natural circulation. Condensation of steam on the SGs under conditions where natural circulation is interrupted may reduce the boron concentration in the downcomer, which, under certain conditions, could cause a positive reactivity insertion when natural circulation is restored, unless prevented. As discussed in FSAR Section 5.4.1.2, the NPM design includes small coolant flow paths in the riser to promote boron mixing and mitigate boron dilution in the downcomer under riser uncovery conditions. The staff evaluation of the riser flow path effectiveness to prevent unacceptable levels of downcomer dilution is in Section 15.0.5 of this SER. SER Section 9.2, which includes the NRC staff’s findings related to PDC 44, GDC 45, and GDC 46, discusses the UHS. FSAR Section 9.2.5, “Ultimate Heat Sink,” gives the parameters for the reactor pool, which serves as the shared UHS for all the modules. The DHRS, as described in the FSAR and evaluated in this SER, can remove heat from the reactor module and transfer it to the reactor pool, the UHS. SER Section 6.2.4, which includes the NRC staff’s findings related to GDC 54 and 57, discusses the containment isolation system.

5.4.3.5 Combined License Information Items

There are no COL information items related to FSAR Section 5.4.3.

5.4.3.6 Conclusion

In conclusion, the staff finds that the applicant has satisfied the requirements in its PDC 34 and 44 as well as the portions of GDC 2, 4, 5, 14, 46, 54, and 57 pertinent to the DHRS. As detailed in the technical evaluation above, the DHRS acts as a robust system to transfer 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 for design-basis events. Based on these considerations, the staff concludes the design of the DHRS is acceptable and meets the applicable regulations specified in SER Section 5.4.3.3.

5.4.4 Reactor Coolant System High-Point Vents

5.4.4.1 Introduction

FSAR Section 5.4.4, "Reactor Coolant System High-Point Vents," addresses the high-point vents for the RCS used in the applicant's 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 application, the applicant 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.

5.4.4.2 Summary of Application

In SDAA Part 7, Section 1.2, the applicant provided the following justification for the exemption:

> The underlying purpose of 10 CFR 50.46a, requiring high-point vents for the RCS, RPV, and for other systems required to maintain adequate core cooling is to preclude an accumulation of noncondensable gases that may inhibit core cooling. As stated in 68 FR [*Federal Register*] 54123:
>
> > This requirement permitted venting of noncondensable 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 noncondensable 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 noncondensable 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 noncondensable gases that may inhibit core cooling during natural circulation.

Further, in SDAA Part 7, Section 1.2, the applicant asserts that its 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 applicant's design, natural circulation is not inhibited by the accumulation of noncondensable gases, and core cooling is not dependent on pump operation. Therefore, the underlying purpose of the requirements is met without high-point vents.

In SDAA Part 7, Section 1.2.1, the applicant provided the following technical basis for the exemption:

> The design includes an RCS that is integral to the RPV; the core, steam generator, and pressurizer are contained in the RPV. The high point of the RCS and pressurizer is the high point of the RPV. The accumulation of noncondensable gases in the RCS and pressurizer steam space is minimized during normal operation by use of the RPV high point degasification line.
>
> As described in FSAR Section 5.4.4, the ECCS includes two 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. The RCS does not include separate post-accident high point vent capability. As described in FSAR Sections 6.2 and 15.0, accumulated noncondensable gases vented to the containment vessel during ECCS operation do not challenge adequate core cooling.
>
> During DHRS cooling events, accumulation of non-condensable gases in the pressurizer does 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 RPV level because the liquid phase is incompressible, and does not impede liquid circulation in the RPV. Noncondensable gas accumulation within the secondary system is calculated and considered in the DHRS performance analysis, summarized in FSAR Section 5.4.3, and determined not to challenge DHRS operation.

According to the applicant, there are no other systems that are necessary to maintain adequate core cooling that require high-point venting. Therefore, the underlying purpose of the 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) are met without the high point vents required by the rules.

5.4.4.3 Regulatory Basis

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

> The 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)....

10 CFR 50.34(f) states, in part, 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….

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

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

10 CFR 50.46a states, in part, 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 noncondensable gases would cause the loss of function of these systems….

10 CFR 52.7, "Specific Exemptions," 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.

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

The two conditions which must be met for granting an exemption 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.
- The Commission will not consider granting an exemption unless special circumstances are present. (Circumstances are enumerated in 10 CFR 50.12(a)(2)).

5.4.4.4 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 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 applicant’s 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 two 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.

The 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).

The applicant states that the ECCS is a two-phase circulation system, and gas accumulation in the RPV cannot disrupt normal flow through the RVV because it is designed for gas flow. The applicant adds that the ECCS accommodates the effects of the noncondensable gases on heat transfer. In topical report TR-124587-A, Revision 1, “Extended Passive Cooling (XPC) and Reactivity Control Methodology” (ML25132A277) dated April 2025, the applicant discusses the treatment of noncondensable gasses in the ECCS analysis. Specifically, the applicant provides the initial conditions and biases and describes noncondensable assumptions for ECCS performance sensitivity calculations. The NRC staff audited the calculation performed for the SDAA in accordance with the methodology outlined in the topical report and the staff confirmed that the SDAA calculation employs a bounding maximum noncondensable gas content.

The NRC staff also reviewed the potential impacts of noncondensable gases to the DHRS. The DHRS is not connected to the RCS; therefore, any highpoint 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 5.4.3 further evaluates these considerations.

Additionally, the applicant’s design does include provisions for venting in the form of a 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 classification as not safety-related to “minimize…[the] accumulation of noncondensable 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.

5.4.4.5 Exemption from 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi)

The applicant 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. Natural circulation is not inhibited by the accumulation of noncondensable gases, and core cooling is not dependent on pump operation. The applicant stated that because of this, the underlying purpose of the requirements is met without high-point vents.

Regulatory Requirements

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

> The 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)....

10 CFR 50.34(f) states, in part, 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….

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

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

10 CFR 50.46 (a) states, in part, 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 noncondensable gases would cause the loss of function of these systems….

10 CFR 52.7, “Specific Exemptions,” 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.

10 CFR 50.12(a) states, in part, that the two conditions that must be met for granting an exemption are the following:

- Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.
- The Commission will not consider granting an exemption unless special circumstances are present. (Circumstances are enumerated in 10 CFR 50.12(a)(2)).

Evaluation for Meeting the Exemption Criteria of 10 CFR 50.12

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, consider exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission's consideration will be governed by 10 CFR 50.12, which states that an exemption may be granted when (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security, and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request.

Authorized by Law

This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended or the Commission's regulations because, as stated above, 10 CFR Part 52 allows the NRC to grant exemptions. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create new accident precursors. The NuScale Power Plant design permits venting of noncondensable gases that could interfere with coolant circulation**.** Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption poses no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption will not alter the design, function, or operation of any structures or plant equipment necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security are not impacted by this exemption.

Special Circumstances

The staff concludes that the requested exemption will not impact the consequences of a design-basis event (DBE), 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 public health and safety, and is consistent with the common defense and security.

The NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the design achieves the underlying purpose of the rules, to provide a means to permit venting of noncondensable gases that could interfere with coolant circulation. As discussed above, because of the nature of the applicant's design, the applicant has demonstrated that 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.

Conclusion

The staff concludes that the requested exemption will not impact the consequences of a DBE, 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 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. The staff therefore determined that an exemption to 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46a, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

5.4.4.6 Combined License Information Items

None.

5.4.4.7 Conclusion

The staff concludes that the applicant's design meets the design requirements to accomplish the ECCS function. For DBEs, the design shows that any accumulated noncondensable gases would be relocated to the containment without impeding heat transfer. The applicant demonstrates the impact of noncondensable gases on the design by employing a bounding maximum noncondensable gas content. Further, the staff concludes that 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 DBE, as there are no local high points in the SG region that exist to impede flow. Additionally, because of the approach used in the applicant's design, as documented above in the technical evaluation, special circumstances are present, as the underlying purpose of the rule is not necessary to achieve compliance. For the reasons set forth in the evaluation above, the staff finds that the requested exemption to 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46a meets the requirements of 10 CFR 50.12(a) and, thus, determined that the exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption. Finally, while the staff made the findings of acceptability on the nature of the RCS and ECCS designs, the staff notes that the design allows for routine degasification during normal operations.

5.4.5 Pressurizer

5.4.5.1 Introduction

The main function of the PZR is to regulate RCS pressure by maintaining a saturated steam-water interface during operating and transient conditions. Steam is formed by energizing immersion heaters in the PZR or 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 flow path between the PZR and the RCS to rapidly communicate pressure changes between the two regions.

Although the SRP does not contain a section specific to review of the PZR, 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 Chapter 15 contain detailed information about the PZR, including the staff's evaluation and conclusion about PZR design features and performance requirements.

5.4.5.2 Technical Evaluation and Conclusion

The staff notes that the NPM-20 design does not include PZR relief valves or PZR block valves, and therefore the applicant stated that the power supply requirements for these valves in 10 CFR 50.34(f)(2)(xx) are not technically relevant. The staff agrees with the applicant's determination because the applicant's design does not include power-operated PZR relief valves or PZR block valves.

The SER sections listed in 5.4.5.1 above discuss the staff's technical evaluation and conclusion on PZR design features and performance requirements.

5.4.5.3 Pressurizer Component Exemptions

5.4.5.3.1 Introduction

In SDAA 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 applicant’s 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).

5.4.5.3.2 Regulatory Basis

The following regulations described in Section 5.4.5.2 of this SER are also applicable to the review of pressurizer component exemptions: 10 CFR 52.47(a), 10 CFR 50.34(f), 10 CFR 52.7, and 10 CFR 50.12(a).

10 CFR 50.34(f)(2)(xiii) states, in part, 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.

10 CFR 50.34(f)(2)(xx) states, in part, 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.

5.4.5.3.3 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 Pressurizer Heaters, Power-Operated Relief and Block Valves, and Pressurizer 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 applicant’s 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 applicant's design meets the underlying purpose of the relevant requirements.

5.4.5.3.4 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 following a LOOP, while decreasing reliance on the ECCS to establish and maintain natural circulation because of concerns that the ECCS may be called upon more frequently than assumed in its design basis. 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. Furthermore, providing emergency power to power-operated relief and block valves would allow closure of the power-operated relief valves after a LOOP to prevent a small-break LOCA.

Unlike traditional PWRs, the applicant's 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 FSAR, Section 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries," show that the applicant's design requires no PZR heater operation, and, therefore, there is 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, safety-related system that does not require electrical power for actuation or operation. Furthermore, it does not actuate the ECCS in the short-term following a LOOP, and long-term cooling after a transient or accident is part of the applicant's ECCS design basis. SER Section 15.2.6 provides the staff's evaluation of the LOOP transient analyses.

5.4.5.4 Exemption from 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.34(f)(2)(xx)

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 applicant's 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).

Regulatory Requirements

10 CFR 50.34(f)(2)(xiii) states, in part, 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.

10 CFR 50.34(f)(2)(xx) states, in part, 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.

Evaluation for Meeting the Exemption Criteria of 10 CFR 50.12

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, consider exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission's consideration will be governed by 10 CFR 50.12, which states that an exemption may be granted when (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security, and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request.

Authorized by Law

This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended or the Commission's regulations because, as stated above, 10 CFR Part 52 allows the NRC to grant exemptions. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create new accident precursors. Because the applicant's design maintains natural circulation following a LOOP without the need for PZR heaters or PZR level indication and without exceeding the ECCS design basis, the design meets the underlying purpose of 10 CFR 50.34(f)(2)(xiii) and the underlying purpose of 10 CFR 50.34(f)(2)(xx) related to PZR level indicators. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption poses no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption will not alter the design, function, or operation of any structures or plant equipment necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security are not impacted by this exemption.

Special Circumstances

In accordance with 10 CFR 50.12(a)(2), the staff finds that special circumstances are present, specifically, provision (ii), that the 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.

Conclusion

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 public health and safety, and are consistent with the common defense and security.

Because the applicant's design maintains natural circulation following a LOOP without the need for PZR heaters or PZR level indication, the design meets the underlying purpose of 10 CFR 50.34(f)(2)(xiii) and the underlying purpose of 10 CFR 50.34(f)(2)(xx) related to PZR level indicators.

5.4.5.5 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, thus, determined that these exemptions, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemptions. Where there are changes to the design material to the bases for the exemptions, the COL applicant that references the SDA would be required to provide an adequate basis for the exemptions.