3 DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS AND EQUIPMENT

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the NuScale Power, LLC (NuScale), 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 FSAR, dated April 09, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25099A237).

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.

3.1 Conformance with the U.S. Nuclear Regulatory Commission General Design Criteria

FSAR Section 3.1, "Conformance with U.S. Nuclear Regulatory Commission General Design Criteria," addresses how the applicant's design conforms to the general design criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities."

The applicant has either described how it complies with the individual GDC, proposed an exemption to the GDC, or developed a principal design criterion (PDC) that addresses the GDC for the NuScale design. The staff's review and assessment of how the applicant addressed the NuScale-specific PDC are documented in the relevant chapters of this report, as shown in Table 3.1-1 below.

Table 3.1-1: NuScale-Specific Principal Design Criteria

FSER Section	Principal Design Criteria
5.4 6.4 7 9.4 9.5 9A 11.5 12.3 15	PDC 19—A control room shall be provided from which actions can be taken to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent as defined in 10 CFR 50.2 for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided with a design capability for safe shutdown of the reactors, including necessary instrumentation and controls to maintain the modules in a safe shutdown condition.
5.4 8.2 8.3 10 15	PDC 34—A system to remove residual heat shall be provided. The system safety function shall be 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 reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.
4.2 6.3 8.2 8.3 15	PDC 35—A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.
6.2 8.2 8.3	PDC 38—A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.

FSER Section	Principal Design Criteria
6.2 6.5 8.2 8.3	PDC 41—Systems to control fission products, hydrogen, oxygen, and other substances that may release into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to ensure that containment integrity is maintained. Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to ensure that its safety function can be accomplished, assuming a single failure.
5.4 8.2 8.3 9.2	PDC 44—A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.

The index of the staff's review of the applicant's requested exemptions to the GDC is located in Section 1.14, "Index of Exemptions," of this report.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

3.2.1.1 Introduction

The NRC requires that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. As described in 10 CFR Part 50, Appendix A, SSCs that are important to safety are those that "provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, defines safe-shutdown earthquake (SSE) ground motion as "the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional," and states the following:

> SSCs required to withstand the effects of the SSE ground motion or surface deformation are those necessary to assure:
>
> (1) the integrity of the reactor coolant pressure boundary (RCPB);
>
> (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition; or

(3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1).

The applicant states in FSAR Section 3.2, "Table 3.2-1 identifies the buildings associated with the site layout and their seismic classification. Table 3.2-2 identifies a list of Seismic Category I SSC that provide pressure integrity functions or their supports, for the [RCPB]. Table 3.2-2 also provides the applicable Quality Assurance Program (QAP) requirements and quality group classification. Discussion of systems comprised of Seismic Category II and III SSC are provided in the applicable chapters."

The SSE is based on an evaluation of the maximum earthquake potential and is the earthquake that produces the maximum vibratory ground motion for which safety-related SSCs are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated Seismic Category I in accordance with Regulatory Guide (RG) 1.29, "Seismic Design Classification."

The staff reviewed the applicant's SDAA in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 3.2.1, "Seismic Classification," which references RG 1.29. The objective of the staff's review was to determine whether SSCs that are important to safety have been appropriately classified and designed to withstand the effects of earthquakes without loss of capability to perform their intended functions.

3.2.1.2 Summary of Application

FSAR: FSAR Section 3.2, "Classification of Structures, Systems, and Components," addresses seismic classification, and to meet the NRC seismic requirements for the design for earthquakes. FSAR Section 3.2 states that the seismic classification of SSCs is consistent with the guidance of RG 1.29, Revision 6. FSAR Section 3.2 further states that the SSCs of radioactive waste management systems are consistent with the seismic design recommendations specified in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." FSAR Section 3.2 also states that the seismic classification of instrumentation sensing lines is consistent with the guidance in RG 1.151, "Instrument Sensing Lines," and that the design of fire protection systems is consistent with the guidance in RG 1.189, "Fire Protection for Nuclear Power Plants."

FSAR Chapter 3 states that the applicant's SSCs are classified as Seismic Category I, Seismic Category II, Seismic Category III, and Seismic Category RW-IIa, RW-IIb, and RW-IIc. FSAR Table 3.2-1, "Seismic Classification of Building Structures," identifies the applicable seismic categories. The classifications and quality groups are described in FSAR Section 3.2.

ITAAC: There are no inspections, tests, analyses, and acceptance criteria (ITAAC) associated with this area of review.

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

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

3.2.1.3 Regulatory Basis

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

- GDC 1, “Quality Standards and Records,” in Appendix A to 10 CFR Part 50 and the applicable quality assurance (QA) requirements of 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” as they relate to applying QA requirements to activities that affect the safety-related functions of SSCs designated as Seismic Category I, commensurate with the importance of their safety functions to be performed

- GDC 2, “Design Bases for Protection against Natural Phenomena,” as it relates to the requirements that SSCs important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions

- GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to the design of means to control suitably the release of radioactive materials in gaseous and liquid effluents

- Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” to 10 CFR Part 100, “Reactor Site Criteria,” and Appendix S to 10 CFR Part 50, as they relate to designing SSCs important to safety to withstand the SSE without loss of capability to perform their safety functions

SRP Section 3.2.1 lists acceptance criteria that are adequate to meet the above requirements and provides review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria to demonstrate that the above requirements have been adequately addressed:

- RG 1.29 provides guidance used to establish the seismic design classification to meet the requirements of GDC 2; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix S.

- RG 1.151 provides guidance on seismic design provisions and classification of safety-related instrument sensing lines.

- RG 1.143 provides acceptable methods and guidance used to establish the seismic design and classification of radioactive waste management SSCs.

- RG 1.189 provides guidance for the proper seismic classification of fire protection systems, including seismic design considerations and seismic classifications for certain SSCs. These provisions support an overall system design that meets the requirements of GDC 2, as it relates to designing these SSCs to withstand earthquakes.

3.2.1.4 Technical Evaluation

GDC 2 requires that nuclear power plant SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. As stated in 10 CFR Part 50, Appendix S, some of these SSCs support functions that are safety-related, such as the following:

- integrity of the RCPB;
- capability to shut down the reactor and maintain it in a safe-shutdown condition;
- capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures that are comparable to the guideline exposures of 10 CFR 50.34(a)(1).

In RG 1.29, Revision 5, issued July 2016, Section C states that the following SSCs of a nuclear power plant, including their foundations and supports, should be designated as Seismic Category I:

a. the RCPB as defined in 10 CFR 50.2;

b. the reactor core and reactor vessel internals;

c. systems or portions thereof that are needed for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system);

d. systems or portions thereof (including but not limited to systems such as residual heat removal and auxiliary feedwater) that are needed to:

   (1) shut down the reactor and maintain it in a safe shutdown condition,

   (2) remove residual heat (including heat stored within the spent fuel pool),

   (3) control the release of radioactive material, or (4) mitigate the consequences of an accident.

As described below, the staff reviewed FSAR Section 3.2.1, “Seismic Classification,” and finds that the application appropriately classified components for the seismic design.

FSAR Table 3.2-1 identifies the buildings associated with the site layout and reflect the correct seismic classifications of SC-I and SC-II, consistent with the information presented in FSAR Section 1.2.

In accordance with RG 1.29, Section C.1, FSAR Table 3.2-2 identifies a list of Seismic Category I SSCs that provide pressure integrity functions or their supports for the RCPB. Table 3.2-2 also provides the applicable QAP requirements and quality group classification. Discussion of systems comprised of Seismic Category II and III SSC are provided in the applicable chapters.

RG 1.29, Revision 5, Staff Regulatory Guidance C.1.b, states that the reactor core and reactor vessel internals (RVIs) should be designated as Seismic Category I. FSAR Table 3.2-2, indicates that the Reactor Coolant System (RCS), which includes the RVIs, is designated as Seismic Category I. Staff Regulatory Guidance C.1.c, states that systems or portions thereof that are needed for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system) should be designated as seismic Category I.

FSAR Table 3.2-2, states that the components of Emergency Core Cooling System (ECCS) which includes the reactor vent valve, reactor vent valve trip valve, reactor recirculation valve, reactor recirculation trip valve, reset valve, reset lines, and trip lines, are Seismic Category I. RG 1.29, Revision 6, Staff Regulatory Guidance C.1.d, states that systems or portions thereof (including but not limited to systems such as residual heat removal and auxiliary feedwater) that are needed to (1) shutdown the reactor and maintain it in a safe shutdown condition, (2) remove residual heat (including heat stored within the spent fuel pool), (3) control the release of radioactive material, or (4) mitigate the consequences of an accident, should be Seismic Category I.

The UHS, decay heat removal system (DHRS) and containment systems (CNTS) systems, as described below, provide the functions listed above.

- The pool cooling and cleanup system (PCWS) maintains UHS level and temperature during normal operation. The UHS maintains the core temperature at acceptably low levels following an accident, including a LOCA, that results in the initiation of ECCS. The passive cooling feature provided by the UHS does not include active components and does not rely on electrical power to perform its safety function.

- The DHRS piping is a high-energy system only associated with the NPM.

- Each NPM has a CNTS with a containment boundary designed to prevent or limit release of radioactive materials under postulated accident conditions. The containment boundary is formed by the CNV and by CIVs and passive containment isolation barriers that are used to prevent releases through the penetrations in the CNV.

DHRS and CNTS are designated as Seismic Category I. The portions of the Reactor Building that form the containment for the safety-related water of the UHS pool are designated as Seismic Category I. Portions of the pool liner that may impact the safety system function of the UHS are designated as Seismic Category I or II. This is consistent with the staff's guidance; therefore, the staff finds this acceptable.

RG 1.29, Revision 5, Staff Regulatory Guidance C.3, states that the pertinent QA requirements of 10 CFR Part 50, Appendix B, should be applied to all activities affecting the safety-related functions of Seismic Category I SSCs. In FSAR Section 3.2.2, the applicant described the QA requirements that will be applied for the various Quality Groups. The staff reviewed these QA requirements and finds that they are consistent with staff guidance and are therefore acceptable.

RG 1.29, Revision 5, Staff Regulatory Guidance C.1.i, states that those portions of SSCs of which continued function is not required but the failure of which could reduce the functioning of any plant feature included in Staff Regulatory Guidance items 1.a through 1.h of RG 1.29, Revision 5, to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSC would not cause such failure. Wherever practical, structures and equipment the failure of which could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

The NRC staff conducted a detailed review of NuScale's process for classifying SSCs in accordance with the criteria committed to in FSAR Section 3.2. This detailed review is documented in NuScale US600 design certification application (DCA) safety evaluation as follows:

> In June 2017, the staff audited (Phase 1 audit) the applicant's design specifications to verify that the component design, qualification, and classification in support of the NuScale Standard Plant DCA are being performed in accordance with the methodology and criteria described in the applicant's various portions of DCA Part 2, Tier 2, including Section 3.2, "Classification of Structures, Systems, and Components." Subsequently, the staff performed a Phase 2 audit of the applicant's design specifications to confirm the updated specifications, in which the applicant provided the resolutions to address the staff's Phase 1 audit findings. During the audit, the staff reviewed the applicant's classification documents. The staff also examined detailed P&IDs to verify system classifications. The staff documented the Phase 1 and 2 audits in "Summary Audit Report of Design Specifications," dated January 25, 2018 (ML18018A234), and "U.S. Nuclear Regulatory Commission Staff Report of Regulatory Audit for NuScale Power, LLC; Follow-Up Audit of Component Design Specifications," dated February 11, 2019 (ML19018A140), respectively. The staff finds that the design classification information described in DCA Part 2, Tier 2, was adequately translated into the design specification. Based on the classification process and these documents, sufficient information exists to demonstrate that the applicant has an appropriate classification process for SSCs important to safety and to conclude that the classification criteria and application of those criteria are consistent with the criteria in RG 1.29, Revision 5, and RG 1.26, Revision 4.

The staff reviewed a sample of classification of SSCs in the SDAA and did not identify any discrepancies. On the basis of the spot check and the audit previously conducted, the staff finds that the applicant has an appropriate classification process for SSCs important to safety and that the classification criteria and application of those criteria are consistent with the criteria in RG 1.29, Revision 6, and RG 1.26, Revision 6. The staff finds this acceptable.

3.2.1.5 Combined License Information Items

There are no combined license information items for this area of review.

3.2.1.6 Conclusion

The staff reviewed the applicant's SDAA. Based on its review of FSAR Section 3.2.1, the staff concludes that the applicant's design of safety-related SSCs, including their supports, are properly classified as Seismic Category I in accordance with RG 1.29, Staff Regulatory Position C.1. In addition, the staff finds that FSAR Sections 3.2.1 and 3.2.2 provide an acceptable process to meet RG 1.29, Revision 6, Staff Regulatory Guidance C.1.h, C.2, and C.3 for SSCs not classified as Seismic Category I. This constitutes an acceptable basis for satisfying the portions of 10 CFR Part 50, Appendix A, GDC 1, GDC 2, and GDC 60; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix S, that require that all SSCs important to safety be designed to withstand the effects of natural phenomena, including earthquakes. Therefore, the staff concludes that the above-cited regulatory requirements have been met.

3.2.2 System Quality Group Classification

3.2.2.1 Introduction

The staff reviewed FSAR Section 3.2.2, in accordance with SRP Section 3.2.2, “System Quality Group Classification,” which references RG 1.26.

In addition to the seismic classifications, FSAR Table 3.2-2 identifies the SSC classification, safety classification/quality group (QG) classification, and the QA requirements necessary to satisfy the requirements of GDC 1 for Seismic Category I SSCs. Discussion of other systems are provided in the applicable chapters.

As discussed in SER Section 3.2.1, the applicant has an appropriate classification process for SSCs important to safety, so the staff concludes that the classification criteria and application of those criteria are consistent with the criteria in RG 1.29, Revision 6, and RG 1.26, Revision 6. Thus, there is reasonable assurance that the classification process exercised throughout the FSAR is consistent with classification criteria addressed in FSAR Section 3.2.

Applicable piping and instrumentation diagrams (P&IDs) identify the classification boundaries of interconnecting piping and valves. SRP Section 3.2.2 references RG 1.26 as the principal document used by the staff to identify, on a functional basis, the pressure-retaining components of those systems important to safety as NRC QG A, B, C, or D. As noted in FSAR Table 1.9-2, “Conformance with Regulatory Guides,” the applicant stated that they conform to Revision 6 of RG 1.26. SER Section 5.2.1.1, “Compliance with the Codes and Standards Rule, 10 CFR 50.55a,” discusses the conformance of RCPB components to the requirements of 10 CFR 50.55a, “Codes and Standards.” RG 1.26 designates these RCPB components as QG A.

In GDC 1, the NRC requires, in part, that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions they perform. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability and adequacy and modified as necessary to assure a quality product in keeping with the required safety function. As stated in SRP Section 3.2.2, these SSCs will be relied upon for the following functions:

- to prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB
- to permit the shutdown of the reactor and maintain it in a safe-shutdown condition
- to ensure the integrity of the RCPB

In accordance with 10 CFR 50.55a(c)(1), components that are part of the RCPB must meet the requirements for Class 1 components in ASME BPV Code, Section III, except as provided in 10 CFR 50.55a(c)(2) through (4). In accordance with 10 CFR 50.55a(d)(1), components classified as QG B must meet the requirements for Class 2 components in ASME BPV Code, Section III. In accordance with 10 CFR 50.55a(e)(1), QG C components must meet the requirements for Class 3 components in ASME BPV Code, Section III.

3.2.2.2 Summary of Application

FSAR Section 3.2.2 describes the criteria used for seismic classification and quality group classification that will be applied to the SSCs for the US460. FSAR Table 3.2-2 identifies a list of Seismic Category I SSC that provide pressure integrity functions or their supports, for the RCPB. FSAR Table 3.2-2 also provides the applicable QAP requirements and quality group classification. Discussion of systems comprised of Seismic Category II and III SSC are provided in the applicable chapters. FSAR Section 3.1.1, "Overall Requirements," states that the plant design conforms to GDC-1.

FSAR Section 3.2.2 provides the requirements for the supports for the SSCs that meet each QG classification in RG 1.26. The design requirements for supports for SSCs in QG A, B, C, and D are identified in FSAR Sections 3.2.2.1 through 3.2.2.4, and specifically describe the codes and standards applicable to the supports for the SSCs in each QG.

In FSAR Sections 3.2.2.1 through 3.2.2.4 include the classification information on the supports for the ASME BPV Code Class 1 through 3 systems to meet the criteria of ASME BPV Code, Section III, Division 1, Subsection NF. This is consistent with requirements in the ASME BPV Code and therefore complies with 10 CFR 50.55a(c), (d) and (e).

The design and construction codes that are recommended for SSCs using the QG classifications in RG 1.26 apply to vessels, piping, valves, pumps, and tanks. These codes do not provide complete design and construction rules for instrumentation components; therefore, instruments are considered outside the scope of RG 1.26 and are not given QG designations. FSAR Chapter 7, "Instrumentation and Controls," and Chapter 8, "Electric Power," provide further details on the codes and standards for instrumentation and electrical systems, respectively.

ITAAC: There are no ITAAC associated with this area of review.

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

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

3.2.2.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR 50.55a, as they relate to SSCs important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed

- 10 CFR 50.55a(c)(1), as it relates to components that are part of the RCPB that must meet the requirements for Class 1 components in ASME BPV Code, Section III, except as provided in 10 CFR 50.55a(c)(2) through (4)

- 10 CFR 50.55a(d)(1), as it relates to components classified as QG B that must meet the requirements for Class 2 components in ASME BPV Code, Section III

- 10 CFR 50.55a(e)(1), as it relates to QG C components that must meet the requirements for Class 3 components in ASME BPV Code, Section III

- SRP Section 3.2.2, Revision 3, issued in 2016, provides guidance to the staff and lists the acceptance criteria adequate to meet the above requirements and provides 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:
    - RG 1.26 describes an acceptable method for determining quality standards for QG B, C, and D water- and steam-containing components important to the safety of water-cooled nuclear power plants.

3.2.2.4 Technical Evaluation

To determine whether the applicant’s SDAA conforms to the requirements of QG classifications and quality standards used for design, the staff reviewed FSAR Section 3.2.2, in accordance with SRP Section 3.2.2 and RG 1.26, Revision 6. The review included the evaluation of the criteria used to establish the QG classifications and the application of the criteria to the classification of the Class 1 SSCs in FSAR Table 3.2-2.

To meet the requirements of 10 CFR 50.55a and GDC 1, the applicant must comply with the requirements of 10 CFR 50.55a(c) for the RCPB, 10 CFR 50.55a(d) for QG B, and 10 CFR 50.55a(e) for QG C. The guidance in RG 1.26 is used to establish the QGs for other safety-related components that contain water, steam, or radioactive material.

As discussed in Section 3.2.1 of this report, the staff reviewed NuScale’s process for classifying SSCs in accordance with the criteria committed to in FSAR Section 3.2. The staff reviewed a sample of SSC classifications and did not identify any discrepancies. Further, the staff relied on the results of a series of detailed audits that were conducted during the DCA review, which concluded that the applicant has an appropriate classification process for SSCs important to safety and application of those criteria are consistent with the criteria in RG 1.29, Revision 6, and RG 1.26, Revision 6.

3.2.2.5 Combined License Information Items

There are no combined license information items for this area of review.

3.2.2.6 Conclusion

Based on its review of the applicable information in the SDAA and on the above discussion, the staff concludes that the QG classifications of the pressure-retaining and non-pressure-retaining SSCs important to safety are in conformance with RG 1.26 and therefore, are acceptable. FSAR Table 3.2-2 identifies major Class 1 components in fluid systems. Conformance to RG 1.26, as described above, and applicable ASME BPV Codes and industry standards provides assurance that component quality will be commensurate with the importance of the safety functions of these systems. This constitutes the basis for complying with the requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1, and is therefore acceptable.

3.3 Wind and Tornado Loading

3.3.1 Wind Loadings

3.3.1.1 Introduction

The staff reviewed FSAR Section 3.3, "Wind and Tornado Loadings," which addresses the design of structures that are required to withstand the effects of severe wind loads and extreme wind loads (Tornado and Hurricane Loads). The staff considered the information provided by the applicant in the SDAA FSAR in establishing the reasonable assurance of safety conclusion.

3.3.1.2 Summary of Application

<u>Severe Wind</u>

The applicant provided the design parameters for severe wind in FSAR Section 3.3.1.1, "Design Parameters for Severe Wind," and FSAR Table 2.0-1, "Site Parameters."

For Seismic Category I structures, the applicant adopted an operating wind speed of 190 miles per hour (mph) from the American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 7-16, "Minimum Design Loads for Buildings and Other Structures." The applicant used a 3 second gust at 33 feet (ft) above ground with an exposure category C, a wind importance factor of 1.15, and the wind design procedure from ASCE/SEI 7-05 to establish the wind load to be used in the structural design.

For RW-IIa structures, the applicant used an operating wind speed of 190 mph, a 3 second gust at 33 ft above ground with an exposure category C, and the wind design procedure from ASCE/SEI 7-16 to establish the wind load to be used in the structural design. In addition, the applicant analyzed RW-IIa structures for reactions from the Seismic Category III superstructure of the Radioactive Waste Building (RWB) from severe wind.

<u>Extreme Wind (Tornado and Hurricane)</u>

The applicant provided design parameters for the design-basis tornado and hurricane in FSAR Section 3.3.1.2, "Design Parameters for Extreme Wind (Tornado and Hurricane)," and FSAR Table 2.0-1, "Site Parameters."

The applicant used the design parameters applicable to the tornado, including the tornado wind translational and rotational speeds, the tornado-generated atmospheric pressure change, and the spectrum of tornado-generated missiles to establish the wind load to be used in the structural design. The applicant also used the design parameters applicable to the hurricane, including the hurricane windspeed and hurricane missile spectra to establish the wind load to be used in the structural design. The applicant adopted the wind pressure design procedure from ASCE/SEI 7-05, which is a reference of practice in wind design. In addition, the applicant applied loading combinations using the individual components of tornado and hurricane loads and their corresponding load factors.

For the NuScale Power Plant US460 standard design (NuScale design, or NuScale US460 design), the applicant used the maximum design basis tornado windspeed of 270 mph with the translational speed of 55 mph, the maximum rotational speed of 215 mph, the radius of maximum rotational speed of 150 ft, the pressure drop of 1.6 pounds-force per square inch (psi)), and the rate of pressure drop 0.9 psi per second in the structural design of seismic Category I structures and RW-IIa structures. The applicant used the maximum design-basis hurricane windspeed of 290 mph in the structural design of Seismic Category I structures and

RW-IIa structures. In addition, the applicant applied the three-fifths factor from Table 2 of RG 1.143 to the maximum tornado and hurricane design parameters to determine the extreme wind loadings in the structural design of RW-IIa structures, and the applicant analyzed RW-IIa structures for reactions from the Seismic Category III superstructure of the RWB from extreme wind.

ITAAC: There are no ITAAC associated with this area of review.

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

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

3.3.1.3 Regulatory Basis

The staff evaluated the applicant's compliance with the following NRC regulations during this review:

- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the importance of the safety functions to be performed.

SRP Section 3.3.1, Revision 3, "Wind Loading," issued March 2007, lists the acceptance criteria adequate to meet the above requirement and provides review interfaces with other SRP sections.

SRP Section 3.3.2, Revision 3, "Tornado Loadings," issued March 2007, lists the acceptance criteria adequate to meet the above requirement and provides review interfaces with other SRP sections.

RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," describes acceptable codes and standards for the design of SSCs in radwaste facilities.

3.3.1.4 Technical Evaluation

SER Sections 2.3.1, "Regional Climatology," and 2.3.2, "Local Meteorology," document the staff's evaluation of the most severe regional and local meteorological data used to specify design wind load parameters.

SER Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," documents the staff's evaluation of the hurricane and tornado wind--generated missiles, respectively.

<u>Wind Loadings</u>

In FSAR Section 3.3.1.1, the staff assessed and accepted the operating wind speed of 190 mph, which is consistent with ASCE/SEI 7-16 for enveloping the majority of the continental US (excluding minor regions such as Dade County, Florida) because it is more conservative than the operating wind speed from ASCE/SEI 7-05, which is a reference of practice in wind design.

For the Seismic Category I structures, the staff assessed and accepted an importance factor of the structures and an exposure category of the site described in FSAR Section 3.3.1.1 because the importance factor of 1.15 and exposure category C are the highest coefficient and category for the wind and cover the worst site conditions for a generic site. The assigned value of the importance factor and the exposure category for the wind are in accordance with ASCE/SEI 7-05. In FSAR Section 3.3.1.3, "Determination of Wind Forces," the staff assessed the applicant's procedures to transform the windspeed into an equivalent pressure to be applied to structures and parts or portions of structures and finds that the applicant's procedures to transform the windspeed into an equivalent pressure are in accordance with ASCE/SEI 7-05. The staff assessed and accepted the minimum value of 0.87 for the velocity pressure exposure coefficient because it provides a more conservative estimate of the design wind load than the design based on ASCE/SEI 705 for a generic site and because it is consistent with the acceptance criteria in SRP Section 3.3.1. Therefore, the staff finds the design wind pressure calculations to be acceptable.

For the RW-IIa structures, the staff reviewed FSAR Section 3.3.1.1 and finds the applicant's methodology for determining the wind pressures acceptable, because (1) the applicant used design parameters for severe wind, which are consistent with ASCE/SEI 7-16; (2) the applicant used an exposure category C, which is the category for the wind and covers the worst site conditions for a generic site; (3) the applicant applied the wind pressure on the building in a more rigorous approach than that described in ASCE/SEI 7-95 per the Table 2 of RG 1.143, Revision 2, and ASCE/SEI 7-05 without the use of an importance factor; and (4) the applicant properly considered reactions from the Seismic Category III superstructure of the RWB from severe wind in the structural design of RW-IIa structures.

<u>Combination of Forces</u>

In FSAR Section 3.3.1.4, "Combination of Forces," the staff assessed the loading combinations of the individual tornado and hurricane loading components and their load factors and finds them acceptable because (1) the applicant properly considered the load from wind effect, the load from tornado atmospheric pressure change effect, and the load from missile impact effect and (2) the loading combinations and their load factors are based on the engineering design principle and consistent with SRP Section 3.3.2, Acceptance Criterion II.3.E.

In addition, the applicant applied the three-fifths factor to the maximum tornado and hurricane design parameters to determine the extreme wind loadings, which is acceptable to the staff since it is consistent with the approach described in Table 2 of RG 1.143. The applicant properly considered reactions from the Seismic Category III superstructure of the RWB from extreme wind in the structural design of RW-IIa structures, which is acceptable to the staff since it reflects the real condition of load transfer from the Seismic Category III superstructure of the RWB to the RW-IIa structures.

The staff reviewed the COL information item listed in Table 3.3.1-1 pertaining to interaction of non-Seismic Category I structures with Seismic Category I structures and Figure 1.2-1, "Conceptual site layout," and finds that the applicant properly accounted for the non-seismic Category I structures that are adjacent to the Seismic Category I portions of the RXB and CRB.

This conclusion is discussed further in SER Section 3.7.2.4.8, where the staff documents its evaluations of interaction of non-Seismic Category I structures with Seismic Category I structures. In addition, the applicant stated in FSAR, COL Item 3.3-1, that a COL applicant will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Seismic Category I portions of the RXB or Seismic Category I portion of the CRB.

3.3.1.5 Combined License Information Items

A COL applicant that refers to the NuScale Power Plant US460 standard design will assess whether the actual site characteristics of severe and extreme wind are within the corresponding severe and extreme wind characteristics considered in the NuScale design. If the actual site characteristics of severe and extreme wind are not within corresponding severe wind characteristics considered in the NuScale design, a COL applicant should reevaluate the design of SSCs to the actual site-specific characteristic. Table 3.3-1 lists the COL information item number and the description of this COL information as provided in FSAR Table 1.8-1. The staff reviewed the COL information items in Table 3.3.1-1 pertaining to interaction of non-Seismic Category I structures with Seismic Category I structures discussed in FSAR Section 3.3, and found it to be acceptable based on the staff’s technical evaluation presented in SER Section 3.3.1.4.

Table 3.3.1-1: NuScale COL Information Item for Section 3.3.1

Item No.	Description	FSAR Section
COL Item 3.3-1	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Seismic Category I portions of the Reactor Building or of the Control Building.	3.3

3.3.1.6 Conclusion

The staff finds that the applicant has adequately used the severe wind loadings in the design of the SSCs for the NuScale US460 design in accordance with the acceptance criteria set forth in SRP Section 3.3.1 and on this basis, the staff concludes that the NuScale severe wind design meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2. The staff has determined that the information in the FSAR provides reasonable assurance that SSCs important to safety will be designed to withstand the effects of severe winds.

The staff also finds that the applicant has adequately used the extreme wind loadings in the design of the SSCs for the NuScale US460 design in accordance with the acceptance criteria set forth in SRP Section 3.3.2, and that the applicant has adequately used the maximum tornado parameters including maximum windspeed, translational speed, rotational speed, and atmospheric pressure change as well as the maximum hurricane wind speed, defined in SRP Sections 2.3.1 and 2.3.2. In addition, the applicant accounted for missiles generated by the tornado wind in accordance with the guidance in RG 1.76 and RG 1.221. Therefore, the staff concludes that the NuScale extreme wind design meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, because the information presented in the FSAR provides reasonable assurance that SSCs important to safety will be designed to withstand the effects of the tornado and hurricane phenomena.

3.4 Water Level (Flood) Design

3.4.1 Internal Flood Protection for Onsite Equipment Failure

3.4.1.1 Introduction

The NRC staff reviewed FSAR, Section 3.4.1, "Internal Flood Protection for Onsite Equipment Failures," in accordance with SRP Section 3.4.1, "Internal Flood Protection for Onsite Equipment Failures."

The review of the flood protection of the equipment considers all SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity. The facility design and equipment arrangements are typically reviewed with respect to the protection against internal flooding resulting from pipe breaks, tank failures, or other equipment failures.

3.4.1.2 Summary of Application

FSAR: The applicant provided internal flooding analyses methodology for the RXB and CRB to confirm that flooding from postulated failures of tanks and piping or actuation of fire suppression systems does not cause the loss of equipment that is required to (1) maintain the integrity of the RCPB for any module, (2) shut down the reactor for any module and maintain it in a safe shutdown condition, or (3) prevent or mitigate the consequences of accidents that could result in unacceptable offsite radiological consequences. FSAR Section 3.4.1 provides the information related to the flooding analyses.

ITAAC: The applicant provided ITAAC associated with internal flooding barriers in the RXB and CRB in FSAR Table 3.11-1, "Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria," Item 02, and Table 3.13-1, "Control Building Inspections, Tests, Analyses, and Acceptance Criteria," Item 02, respectively. These ITAAC are evaluated in Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria," of this SER.

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

Technical Reports: There are no TRs related to internal flood protection.

3.4.1.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, GDC 2, as it relates to the SSCs important to safety being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the SSCs important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation; maintenance; testing; and postulated accidents, including loss-of-coolant accidents (LOCAs).

The staff review guidance in SRP Section 3.4.1, Revision 3, lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections. In meeting GDC 2, full-circumferential ruptures of non-seismic moderate energy piping are assumed in a seismic event. The requirements of GDC 4 are met if SSCs important to safety are designed to accommodate the flooding of discharged fluid resulting from high- and moderate-energy line breaks that are postulated in SRP Sections 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” and 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping.”

SRP 3.4.1 (Revision 3) indicates meeting the requirements of GDC 4 ensures that the SSCs important to safety will be appropriately protected from potential flooding from liquid-carrying component in the plant.

3.4.1.4 Technical Evaluation

The staff reviewed FSAR Section 3.4.1, in accordance with SRP Section 3.4.1 to ensure compliance with the regulations. Based on the information provided in the FSAR, the staff’s evaluation was limited to the review of the methodology and assumptions used in performing flood analyses to identify potential internal flooding sources inside RXB and CRB due to postulated pipe ruptures and fire suppression activities. As stated in FSAR Section 3.4.1, the applicant conducted a flood analysis using various zones consisting of the following steps:

- identification of potential sources of internal flooding;
- division of flood-able areas of each building into flood zones;
- calculating maximum steady-state flood heights based on identified sources for use in informing flood mitigation requirements; and
- determining flood mitigation design features.

FSAR Table 3.4-1 lists the limiting water sources and maximum flooding height in the RXB and CRB within zones and elevations. FSAR Section 3.4.1 indicates the flooding analysis performed considers zones that contain SSC subject to flood protection, but not the specific SSC themselves. FSAR Table 3.4-1 and FSAR Section 3.4.1 do not define location and impact on SSCs within the defined zones.

In FSAR Section 3.4.1.1, the applicant indicates analysis has been performed to identify potential break locations or eliminate the piping from consideration of potential breaks, high- and moderate-energy piping greater than 1-inch nominal diameter is assumed to have either a full-circumferential break or through-wall leakage crack, depending on the pipe energy and seismic classification. The applicant provides the break type evaluated and summarizes results of the flooding analysis in FSAR Table 3.4-1. As shown in FSAR Table 3.4-2, (Refer to RAI-10167-R1 (Q3.4.1-3) response, ML24222A594) within the RXB, the applicant defines the bounding flood sources considered to calculate maximum flood heights using breaks in the fire protection system, fire suppression activities, site cooling water system, main steam line breaks, main feedline breaks, and breaks of other auxiliary fluid systems (e.g., CVCS, pool cooling and cleanup, utility water, and demineralized water). FSAR Table 3.4-1 shows the Fire Protection System and Utility Water System are the most limiting cases in terms of internal flooding water sources within RXB. FSAR Table 3.4-1 also provides a summary of the CRB flooding analysis. Within the CRB, as shown in FSAR Table 3.4-2, the applicant considers breaks in the fire protection system, fire suppression activities, and breaks in the Chilled water, Utility water and potable water systems. Based on the evaluated breaks and sources, the staff determined the applicant appropriately defined bounding flooding sources to calculate flood heights.

The staff reviewed considerations and assumptions used in the internal flooding analysis with respect to calculation of flood levels. FSAR Section 3.4.1.1 describes the considerations and assumptions used for deriving bounding flood height. The staff audited the applicant's flood analysis, and the evaluation adequately derives the bounding maximum flood levels. The applicant used pipe parameters and duration of leakage from piping system ruptures which assumed to be 40 and 30 minutes between leak initiation and leak isolation for the RXB and CRB, respectively. FSAR Section 3.4.1.1 further indicates this timing is based on plant personnel operative walk-downs, the use of plant monitoring equipment, and the use of closed-circuit video monitoring systems for keeping visuals on many sections of the plant. This is a reasonable value for a normally occupied structure. The fire suppression system is assumed to be isolated after 60 minutes and 120 minutes for CRB and RXB, respectively. The staff finds these values reasonable based on the walkdowns, use of plant monitoring equipment, and video monitoring systems as described above. The staff finds that the applicant has provided an adequate measure consistent with the guidance in SRP Section 3.4.1, to ensure that potential pipe breaks in the RXB and CRB can be isolated in a timely manner.

As indicated above, FSAR Table 3.4-1 provides the results of the flooding analyses related to the maximum flood heights based on bounding break within specific flood zones analyzed. In areas containing equipment subject to flood protection, FSAR Section 3.4.1 indicates the applicant stated that mitigation of potential flooding in the identified zones will be accomplished by providing watertight or water-resistant doors, elevating equipment above the flood level, enclosing or qualifying equipment for submersion, or providing other similar types of flood protection. Although it includes maximum flood heights, the FSAR is lacking information defining the location and description of equipment subject to flood protection and mitigation of consequences from flooding. Therefore, the staff is unable to evaluate adequate protection of SSC within specific zones or locations containing undefined SSC subject to flood protection. As discussed in Section 3.4.1.5 of this report, COL Items require the COL applicant to complete the evaluation for the SSCs subject to flood protection and appropriate mitigation features.

Based on the above, the staff finds the applicants method used to identify the potential sources of internal flooding and calculation of the corresponding flood heights is acceptable. As specified in COL Item 3.4-1, 3.4-2 and 3.4-3, the staff understands a COL applicant will be required to further evaluate adequate protection of the SSCs from internal flooding and their ability to withstand maximum flood height defined in FSAR Table 3.4-1, including submergence, for each flood zone to ensure SSCs remain capable of performing their safety function. Based on the content provided in the FSAR, the staff is unable to conclude SSCs important to safety will be capable of withstanding the effects of internal flooding in accordance with GDC 2 and GDC 4 as part of the SDAA review. This review is appropriate to be deferred to the COL applicant based on the COL information items included in the FSAR, which are shown below.

3.4.1.5 Combined License Information Items

FSAR Section 3.4.1.5 and Table 1.8-1 list the COL information item descriptions related to internal flood protection. Additional analysis of SSCs subject to flood protection and their location will be required, per COL Items below, to conclude adequate protection of SSCs to withstand effects of internal flooding, without losing the ability to perform their safety function.

Table 3.4.1-1: NuScale COL Information Items for Section 3.4.1

Item No.	Description	FSAR Section
COL Item 3.4-1	An applicant that references the NuScale Power Plant US460 standard design will confirm the final location of structures, systems, and components subject to flood protection. The final routing of piping, and site-specific tanks or water source tanks are placed in locations that will not cause unanalyzed flooding to the Reactor Building or Control Building.	3.4.1.5
COL Item 3.4-2	An applicant that references the NuScale Power Plant US460 standard design will develop the on-site program addressing the key points of flood mitigation consistent with the methodology described in Section 3.4.1. The key points to this program include the procedures for mitigating internal flooding events; development of the equipment list of structures, systems, and components subject to flood protection in each plant zone; and analysis providing assurance that the program reliably mitigates flooding to the identified structures, systems, and components consistent with flood levels identified in Table 3.4-1.	3.4.1.5
COL Item 3.4-3	An applicant that references the NuScale Power Plant US460 standard design will develop an inspection and maintenance program to ensure that each water-tight door, penetration seal, or other "degradable" measure remains capable of performing its intended function.	3.4.1.5

3.4.1.6 Conclusion

Based on the discussion above, the staff concludes the NuScale Power Plant US460 standard design, as it relates to internal flood protection, provides a reasonable methodology for a COL applicant to demonstrate compliance with requirements of 10 CFR Part 50, Appendix A, GDC 2 and GDC 4 for SSCs important to safety to accommodate effects of internal flooding and ensure that the SSCs important to safety will be appropriately protected from potential flooding from liquid-carrying components in the plant.

3.4.2 Flood Protection from External Sources

3.4.2.1 Introduction

The staff reviewed FSAR Section 3.4.2, "Flood Protection from External Sources," which addresses the design of Seismic Category I structures that are required to withstand the effects of the highest flood and ground water levels specified for the NuScale design. The staff considered the information provided by the applicant in the FSAR in establishing the reasonable assurance of safety conclusion.

3.4.2.2 Summary of Application

The applicant provided the flood and ground water site parameters in FSAR Section 3.4.2.1, "Probable Maximum Flood," and Table 2.0-1, respectively. The applicant stated that the probable maximum flood elevation (including wave action) of the design is one foot below the baseline plant elevation and the maximum ground water elevation for the design is two feet below the baseline plant elevation. In addition, the applicant described the bounding parameters for both rain and snow and the design features necessary to protect the safety-related and risk-significant SSCs from ground water intrusion without the use of a permanent dewatering system. The applicant described the analysis procedures that are used to transform the static effects of the highest flood and ground water levels into effective loads applied to Seismic Category I structures.

ITAAC: SDAA Part 8, Tables 3.11-1 and 3.13-1 provide the ITAAC for protecting the Seismic Category I RXB and the Seismic Category I CRB against external flooding. These ITAAC are evaluated in SER Section 14.3.13.

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

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

3.4.2.3 Regulatory Basis

The staff evaluated the applicant's compliance with the following NRC regulations during this review:

- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the importance of the safety functions to be performed.

SRP Section 3.4.2, Revision 3, "Analysis Procedures," lists the acceptance criteria adequate to meet the above requirement and provides review interfaces with other SRP sections.

3.4.2.4 Technical Evaluation

SER Sections 2.4.3, "Probable Maximum Flood on Streams and Rivers," and 2.4.12, "Groundwater," document the staff's evaluations of the flood and ground water site parameters, respectively.

In FSAR Section 3.4.2, the staff assessed the applicant's analysis procedures that are used to transform the static and dynamic effects of the highest flood and ground water levels into effective loads applied to Seismic Category I structures. FSAR Section 3.8.4.3.3, "Earth

Pressure," provides the applicant's detailed analysis procedures to calculate the hydrostatic ground water pressure. The staff reviewed the analysis procedures, FSAR Table 3.8.4-6, and FSAR Figures 3.8.4-3 through 3.8.4-6 and finds that the applicant properly accounted for flood and ground water in the analysis and that the total horizontal pressure is calculated as the sum of the surcharge loads, hydrostatic pressure, and effective lateral soil pressure, considering the buoyancy effects. The staff assessed the design-basis flood level (including wave action) and determined that there are no dynamic flood loads on the Seismic Category I structures because the highest flood level is below the proposed plant grade and because the design does not use a permanent dewatering system. Based on its review, the staff finds that the analysis procedures to transform the static effects of the highest flood and ground water levels into effective loads applied to Seismic Category I structures are acceptable and that the analysis procedures are in accordance with general engineering design principles and SRP Section 3.4.2, Acceptance Criterion II.2.

In addition, the staff assessed the protection of the below grade portions, mentioned in FSAR Section 3.4.2.1, of the Seismic Category I RXB from ground water intrusion. The staff assessed the specified design life for water stops, waterproofing, damp proofing, and watertight seals and considered how the duct bank connection to the RXB is protected from the ground water intrusion. Additionally, the applicant proposed inclusion of COL Item 3.4-4, which will instruct a COL applicant to determine the extent of waterproofing and damp proofing needed for the underground portion of the RXB, including the duct bank connection to the RXB, based on site-specific conditions and provide the specified design life for water stops, waterproofing, damp proofing, and watertight seals. The staff reviewed COL Item 3.4-4 directing the COL applicant to address the water-leak-tight function of the below grade portions of the RXB, including the duct bank connection to the RXB, based on site-specific conditions, and finds the COL Item 3.4-4 to be appropriate for this case, because it is the responsibility of the COL applicant to ensure the protection of the below-grade portions of the Seismic Category I structures from ground water intrusion.

FSAR Section 3.4.2.2, "Probable Maximum Precipitation," discusses the bounding parameters for both rain and snow in the NuScale design. SER Section 3.8.4 documents the staff's evaluation of the bounding rain and snow loads.

FSAR Section 3.4.2.3, "Interaction of Non-Seismic Category I Structures with Seismic Category I Structures," indicates that nearby structures are assessed or analyzed to ensure that there is no credible potential for interactions that could adversely affect the Seismic Category I portions of the RXB and Seismic Category I portions of the CRB. The staff reviewed FSAR Section 3.4.2.3 and Figure 1.2-1, "Conceptual site layout," and finds that the applicant properly accounted for the non-Seismic Category I structures that are adjacent to the Seismic Category I portions of the RXB and CRB. This conclusion is discussed further in SER Section 3.7.2.4.8, where the staff documents its evaluations of interaction of non-Seismic Category I structures with Seismic Category I structures. In addition, the applicant stated in COL Item 3.4-5, that a COL applicant will confirm that nearby structures exposed to external flooding will not collapse and adversely affect the RXB or Seismic Category I portion of the CRB.

3.4.2.5 Combined License Information Items

The COL applicant that refers to the NuScale Power Plant US460 standard design will assess whether the actual data of the highest flood and ground water levels are within corresponding site parameters of the NuScale design. The COL applicant should reevaluate the SSCs important to safety in the NuScale design if site characteristics of flood and ground water are not within the corresponding site parameters of the NuScale design. The staff reviewed the COL information items in Table 3.4.2-1 pertaining to flood protection from external sources discussed in FSAR Section 3.4.2, and found these to be acceptable based on the staff's technical evaluation presented in SER Section 3.4.2.4.

Table 3.4.2-1 lists the COL information item numbers and descriptions of these COL information.

Table 3.4.2-1: NuScale COL Information Items for Section 3.4.2

Item No.	Description	FSAR Section
COL Item 3.4-4	An applicant that references the NuScale Power Plant US460 standard design will determine the extent of waterproofing and damp proofing needed for the underground portion of the Reactor Building, including the duct bank connection to the Reactor Building, based on site-specific conditions. Additionally, the applicant will provide the specified design for waterstops, waterproofing, damp proofing, and watertight seals. If the design life is less than the operating life of the plant, the applicant will describe how continued protection will be ensured.	3.4.2.1
COL Item 3.4-5	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to external flooding will not collapse and adversely affect the Seismic Category I portions of the Reactor Building or of the Control Building.	3.4.2.3

3.4.2.6 Conclusion

The staff finds that the NuScale design, as it relates to protection of structures against flood from external sources, is in accordance with the acceptance criteria set forth in SRP Section 3.4.2. The staff also finds that the applicant has adequately considered the ground water and the maximum flood level, both of which are below the grade level, as hydrostatic loads in designing the walls and foundation mat. Therefore, staff concludes that the NuScale flood protection design meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, because the staff has determined that the information presented in FSAR provides reasonable assurance that SSCs important to safety will be designed to withstand the effects of the highest flood and ground water levels specified for the NuScale design.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Introduction

This portion of the SER addresses both FSAR Section 3.5.1.1, "Internally-Generated Missiles (Outside Containment)," and FSAR Section 3.5.1.2, "Internally-Generated Missiles (Inside Containment)." Turbine missiles are evaluated in Section 3.5.1.3 of this report.

SRP Sections 3.5.1.1, “Internally Generated Missiles (Outside Containment),” and 3.5.1.2, “Internally-Generated Missiles (Inside Containment),” delineate that SSCs important to safety are to be protected from internally generated missiles to ensure compliance with GDC 4 requirements. This includes internally generated missiles from component overspeed failures; missiles that could originate from high energy fluid system failures; and missiles caused by, or as a consequence of gravitational effects (e.g., falling or dropping equipment). An internally generated missile is a dynamic effect of such failures, and its impact on SSCs that are important to safety must be evaluated. Protecting SSCs from the effects of internally generated missiles ensures the capability to shut down and maintain the reactor in a shutdown condition and the capability to prevent a significant uncontrolled release of radioactivity.

3.5.1.1.2 Summary of Application

FSAR: FSAR Section 3.5.1.1 and Section 3.5.1.2 describe credible and noncredible internally generated sources and missile protection for SSCs. These FSAR sections also present the basis for identifying credible and noncredible missiles and the design measures to limit missile generation and provide protection to SSCs.

ITAAC: There are no ITAAC associated with this area of review.

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

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

3.5.1.1.3 Regulatory Basis

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

- GDC 4, as it relates to the design of SSCs important to safety to protect them against the dynamic effects of internally generated missiles outside containment.

- GDC 4, as it requires, in part, that SSCs important to safety shall be appropriately protected against the dynamic effects of internally generated missiles outside containment that may result from equipment failures.

The guidance in SRP Sections 3.5.1.1 and 3.5.1.2 provides the relevant regulatory requirements, as well as interfaces with other SRP sections.

3.5.1.1.4 Technical Evaluation

The staff reviewed the applicant’s design for protecting SSCs important to safety against internally generated missiles in accordance with the guidance of SRP Sections 3.5.1.1 and 3.5.1.2. The staff reviewed FSAR, Section 3.5, “Missile Protection.” The staff also reviewed other FSAR sections noted below.

FSAR Section 3.5, in part, addresses protection from internally generated missiles both inside and outside containment. Plant system sections within the FSAR contain classification tables identifying safety-related and nonsafety-related SSCs throughout the plant, including the associated seismic category, Quality Group, and equipment classifications for SSCs. FSAR Section 1.2, “General Plant Description,” provides the general arrangement drawings that define the building locations. The information in Classification of Structures, Systems, and Component tables located within the individual plant system sections is sufficient to identify the SSCs important to safety that are subject to missile protection.

The applicant stated that the design achieves protection from external missiles by locating SSCs that require missile protection inside Seismic Category I portions of the RXB or CRB. It is further noted that GDC 2 and GDC 4 are met because SSCs are designed to withstand the effects of internally and externally generated missiles without losing their safety function. FSAR Section 3.5 indicates FSAR Table 17.4-1 defines list of safety related and risk-significant SSC relied upon following a missile-producing event. The applicant stated that the following methods will provide building missile protection:

- having design features to prevent missile generation.

- orienting or physically separating potential missile sources away from equipment subject to missile protection.

- providing local shields and barriers for equipment subject to missile protection.

SER Section 3.5.3 addresses the staff’s evaluation of the design of structures, shields, and barriers used for missile protection.

FSAR Section 3.5.1.1, describes the methodology for protection from the potential of internally generated missiles that could result from failure of plant equipment. The applicant stated that internally generated missiles can be generated from pressurized systems and components, rotating equipment, explosions, or improperly secured equipment.

The staff reviewed the potential for missiles generated from pressurized systems. FSAR Section 3.5.1.1.1, “Pressurized Systems,” considers the following potential missiles from pressurized systems as noncredible ($P_1 < 10^{-7}$):

- moderate- and low-energy systems with operating pressures of less than 1.90 megapascals (MPa) (gauge) (275 pounds-force per square inch, gauge (psig)), because of insufficient stored energy to generate a missile

- piping and valves designed in accordance with ASME BPV Code, Section III, and maintained in accordance with the ASME BPV Code, Section XI, inspection program

- threaded valve stems with back seats because they are designed to prevent ejection of the stems and valve stems with power actuators because they are effectively restrained by the actuator

- nuts, bolts, and a combination of the two because of the small amount of stored energy

The staff reviewed the reasons stated above to eliminate certain missile sources. These missile sources are either designed to a high level of quality in accordance with ASME BPV Code, Section III, thus demonstrating that missile generation is unlikely, or they do not have sufficient energy to generate a credible missile. Further review of the mechanical system and component design, as well as applied ASME code, is contained in Section 3.9 of this report. Therefore, the staff finds the above list of noncredible missile sources acceptable.

The staff also reviewed information on the potential missiles generated from rotating components. FSAR Section 3.5.1.1.3, "Rotating Equipment," states that the NuScale design has a limited amount of rotating equipment because there are no reactor coolant pumps, turbine -driven pumps, or other large rotating components inside safety--related structures. FSAR Section 3.5.1.3, "Turbine Missiles," discusses protection of essential SSC against the effects of turbine missiles. SER Section 3.5.1.3 evaluates the turbine missile protection. FSAR Section 3.5.1.1.3 also states that the catastrophic failure of rotating equipment, such as fans and compressors, is not considered a credible missile source because the equipment is designed to preclude having sufficient energy to pass through the housing in which it is contained. The staff finds the applicant's FSAR information on the potential missiles generated from rotating components acceptable and consistent with the guidance in SRP Section 3.5.1.1.

In reviewing the potential for missiles generated from pressurized gas cylinders, the applicant stated in FSAR Section 3.5.1.1.2, "Pressurized Cylinders," that cylinders, bottles, and tanks containing highly pressurized gas cylinders are considered missile sources unless appropriately secured. For example, the control room habitability system air bottles are mounted in Seismic Category I racks, and plates and straps restrict horizontal and vertical movement. Therefore, these measures prevent the control room habitability system air bottles from becoming missiles. In addition, procedures developed in accordance with FSAR Section 13.5.2.2, ensure that portable pressurized gas cylinders or bottles are moved to a location where they are not a potential hazard to equipment subject to missile protection or are seismically restrained. The staff finds the applicant's information on the potential for missiles generated from pressurized gas cylinders acceptable and consistent with the guidance in SRP Section 3.5.1.1.

The staff evaluated the potential for missiles generated from explosions. FSAR Section 3.5.1.1.4, "Explosions," states that battery compartments in the CRB and RXB are ventilated to preclude the possibility of hydrogen accumulation. The RWB waste management control room, battery and battery charging rooms are each served by two dedicated 100-percent capacity recirculating fan cooling units. The staff reviewed the above design features of the batteries and battery compartments and agrees with the applicant that these measures ensure that missiles generated from a hydrogen explosion are unlikely.

The applicant also addressed the potential for gravitational missiles from falling objects. If the drop of non-seismically designed SSCs could adversely affect safety-related systems or risk -significant SSCs, the applicant specified that it will be designed to Seismic Category II to protect the SSCs from the impact of dropped objects. FSAR Section 9.1.5, "Overhead Heavy Load Handling Systems," discusses measures used to address the safe operation of the overhead heavy load handling equipment, and SER Section 9.1.5 evaluates such measures. In addition, procedures developed in accordance with FSAR Section 13.5.2.2, ensure that unsecured equipment is seismically restrained, is removed from the building, or is moved to a location where it is not a potential hazard to equipment subject to missile protection. The staff finds the applicant's information on the potential for gravitational missiles from falling objects acceptable and consistent with the guidance in SRP Section 3.5.1.1.

The staff also reviewed the potential for internally generated missiles from inside containment. FSAR Section 3.5.1.2, states that there is no rotating equipment inside containment. All pressurized components inside containment are ASME BPV Code Class 1 or 2 and therefore are not considered credible missile sources. The applicant indicates the CRDM housing failure, sufficient to create a missile from piece of the housing or that allows a control rod to be ejected rapidly from the core is not credible based on being a Class 1 appurtenance. This design and protective features are discussed in Section 3.8 and 3.9 of this report. The staff reviewed the applicant's bases as described above against the staff's review guidance in SRP Section 3. 5.1.1 and finds the applicant's conclusion on the elimination of the above components as credible missile sources acceptable and consistent with the guidance in SRP 3.5.1.1. FSAR Section 15.4.8, "Spectrum of Rod Ejection Accidents," presents the safety analyses of the rod ejection accident and documents the associated staff review.

Based on its review, the staff finds the applicant's approach to identify potential missiles, determine the statistical significance of potential missiles, and provide measures for SSCs needing protection against the effects of missiles to be acceptable. Therefore, the staff concludes that the applicant's evaluation of potential internally generated missiles resulting from equipment and component failures satisfies the applicable requirements related to GDC 4.

3.5.1.1.5 Combined License Information Items

No COL information items are directly associated with this review area.

3.5.1.1.6 Conclusion

The staff's review concludes that the applicant's design bases for SSCs important to safety necessary to maintain a safe plant shutdown, ensure the integrity of the RCPB, and prevent a significant uncontrolled release of radioactivity meet the requirements in 10 CFR Part 50, Appendix A, GDC 4, for SSCs to be protected from internally generated missiles. The applicant FSAR is consistent with the guidance contained in SRP Section 3.5.1.1 with regard to which SSCs should be protected from missile impacts.

3.5.1.2 Internally Generated Missiles (Inside Containment)

SER Section 3.5.1.1 evaluates internally generated missiles inside containment.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires SSCs important to safety to be appropriately protected against dynamic effects of postulated accidents, including the effects of missiles that may result from equipment failures and from events and conditions outside the nuclear power unit. One potential source of plant missiles is the rotor of the main turbine. The applicant must consider this potential source of turbine missiles in the plant's design and must protect SSCs important to safety from the adverse effects of postulated turbine missiles.

The objective of the staff's review is to determine whether the potential turbine missiles have been appropriately identified and whether the SSCs important to safety have been appropriately protected from any adverse effects that may result from these missiles.

3.5.1.3.2 Summary of Application

FSAR: FSAR Section 3.5.1.3 describe the NuScale SDA, as summarized, in part, below.

In FSAR, Figure 1.2-1, Figure 3.5-1, and Figure 3.5-2 show the turbine generator building layout with respect to safety-related and risk-significant SSCs. Safety-related and risk-significant SSCs for the NuScale design are located principally within the RXB and CRB. FSAR Figures 1.2-8 through 1.2-17 show the RXB at different elevations with respect to their safety-related and risk-significant SSCs. The turbine generator rotor shafts are physically oriented such that the RXB is within the turbine low-trajectory hazard zone and therefore are considered to be unfavorably oriented with respect to the RXB, as defined by RG 1.115, Revision 2, “Protection Against Turbine Missiles,” issued January 2012. The CRB is located outside of the low-trajectory hazard zone and is not assessed for low-trajectory turbine missiles.

The NuScale design employs a barrier approach, in combination with the physical separation of redundant safety-related equipment, for the protection against postulated turbine missiles. The bounding missile for the NuScale design is defined as half of the last stage portion of the turbine rotor with the weight of the associated blades. The rotor piece was selected as the bounding turbine missile based on an evaluation of kinetic energy and total penetration distance, as compared to a turbine blade or turbine blade with a piece of the rotor attached.

ITAAC: There are no ITAAC associated with this area of review.

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

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

3.5.1.3.3 Regulatory Basis

“Design -Specific Review Standard for the NuScale SMR Design,” (DSRS), Section 3.5.1.3, Revision 0, “Turbine Missiles,” issued June 2016 (ML15355A364), provides the relevant NRC requirements for this area of review, which are summarized below, and the associated acceptance criteria, as well as the review interfaces with other sections of the DSRS:

- In GDC 4, the NRC requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure.

The following acceptance criteria are adequate to meet the above requirements:

- In accordance with DSRS Section 3.5.1.3.II.1, consideration of turbine missile protection is relevant for SSCs necessary to ensure (1) the integrity of the RCPB, (2) the capability to shut down and maintain the reactor in a safe condition, and (3) the capability to prevent accidents that could result in potential offsite exposure, which represents a significant fraction of the guideline exposures specified in 10 CFR 50.67(b)(2) or 10 CFR Part 100. RG 1.115, Revision 2, Appendix A, provides examples of systems that are important to safety and that, therefore, should be protected; these systems are denoted as essential SSCs. The effect of physical separation of redundant or alternative systems may also be considered.

- RG 1.115, Staff Regulatory Position C.3, specifies that when barriers provide protection of essential systems, dimensioned plan and elevation layout drawings should include information on wall or slab thicknesses and materials of pertinent structures. The protection is considered acceptable if no missile can compromise the final barrier protecting any essential SSCs. Concrete barriers should be thick enough to prevent backface scabbing. In FSAR Section 3.5.1.3.4, the applicant stated that a turbine missile can result in backface scabbing or penetration of the initial concrete barrier; however, SSCs are sufficiently protected by additional internal barriers and physical separation of redundant SSCs to support essential safety functions.

- A method to meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, is to use installed or existing structures for protecting essential SSCs that meet the acceptance criteria in DSRS Section 3.5.3, “Barrier Design Procedures.” Department of the Army Technical Manual TM-5-885-1, “Fundamentals of Protective Design for Conventional Weapons,” issued November 1986 (ML101970069), provides additional guidance.

3.5.1.3.4 Technical Evaluation

The failure of a rotor in a steam turbine may result in the generation of high-energy missiles that could affect essential SSCs. These essential SSCs should be adequately protected from the effects of turbine missiles such that functions important to safety are maintained. RG 1.115 provides three approaches for protecting essential SSCs: (1) favorable orientation of the turbine unit such that all essential SSCs are outside the missile strike zone, (2) limiting the frequency of turbine missile generation, or (3) use of barriers to protect essential SSCs. The applicant elected to use a barrier approach to protect essential SSCs. Details of the applicant’s approach are provided in NuScale FSAR, Section 3.5.1.3. The staff reviewed this information using the guidelines in DSRS Sections 3.5.1.3 and 3.5.3.

NuScale FSAR, Section 3.5.1.3, states that the turbine generators are unfavorably oriented such that essential SSCs in the RXB are within the low-trajectory turbine missile strike zone, as defined by RG 1.115. The staff agreed with the applicant’s determination that the turbine generators are unfavorably oriented as defined by RG 1.115, based on the plant layout with the RXB. Because the applicant has elected not to demonstrate a low likelihood of turbine missile generation, barriers are necessary to protect essential SSCs in the RXB from low-trajectory turbine missiles. NuScale included low-trajectory missiles in the analysis of the barriers for the RXB as discussed below. As noted in RG 1.115, for unfavorably oriented turbines, evaluation of high-trajectory missiles is not required because the probability of a high trajectory missile exiting the casing at a trajectory that results in striking and damaging an essential SSC is much smaller than the equivalent probability for low-trajectory missiles.

The staff notes that Section 3.5.1.3 of the FSAR was revised to state “Essential SSC requiring protection from turbine missiles are located in the RXB and are listed in Table 3.5-2, “Essential SSC Locations.” However, the staff notes that simply removing the CRB from the list does not justify that there are no safety-related or risk significant SSCs in the CRB that should be protected from turbine missiles.

FSAR Table 17.4-1 states that the control room building protects safety-related and risk significant equipment..

NuScale revised Table 17.4-1 to remove protection of the CRB from externally generated missiles and FSAR Section 3.5.1.3 to state that the only safety-related and risk-significant equipment in the CRB is the Module Protection System (MPS) manual functions with all automated functions performed by the MPS contained in the RXB. Therefore, the CRB does not contain essential SSC required by RG 1.115 to be protected. NuScale states that the CRB, including the Main Control Room (MCR) and operators, and SSCs in the CRB are not required to accomplish the criterial from RG 1.115 and are not essential to the plant response to a turbine failure event to (1) maintain the integrity of the RCPB, (2) to maintain the plants capability to prevent accidents which could result in potential off-site exposure, and (3) to maintain the plant’s capability to shut down the reactor and maintain it in a safe shutdown condition.

The staff notes that NuScale FSAR Section 3.5, “Missile Protection,” states that “the design achieves protection from external missiles by locating structures, systems, and components (SSC) that require missile protection inside Seismic Category I portions of the Reactor Building (RXB) or Seismic Category I portions of the Control Building (CRB).” In addition, FSAR Section 3.5.2 states that safety-related and risk-significant SSC in the Seismic Category I RXB and Seismic Category I portions of the CRB are protected and meet the guidance of RG 1.117 for protection of SSC from wind, tornadoes, and hurricane missiles. Therefore, NuScale Power Plant US460 standard design requires that the RXB and CRB be protected from missiles. In addition, RG 1.115 (turbine missiles) uses the same criteria for which SSC and buildings need to be protected from missiles as RG 1.117 (wind generated missiles). Based on this information, it does not matter how the missile originated, just that the RXB and CRB be protected from missiles.

FSAR Sections 3.5.1.3, 3.5.2, and Table 1.9-2 were revised to remove the requirement to protect the CRB from any missiles since the CRB does not contain essential SSCs and is not required to be evaluated for any type of missile. Therefore, based on the NuScale design, the CRB, including the control room and the control room operators are not essential, and therefore are not required to be protected from missiles in order to maintain the plant’s capability to prevent accidents which could result in potential off-site exposure, and to maintain the plant’s capability to shut down the reactor and maintain it in a safe shutdown condition. Based on the NuScale design features, the staff finds that the CRB and the control room operators do not have to be protected from turbine missiles because they are not necessary for the NuScale design and therefore, a barriers analysis for high trajectory missiles on the CRB is not required. However, the staff notes that FSAR Section 3.5.2 was also revised to state that while the CRB does not contain essential SSC, the Seismic Category I portion of the CRB is designed and evaluated for external missiles, except turbine missiles as described in Section 3.5.1.3. The evaluation for protection of other missiles is discussed in Section 3.5.2 of this SER.

The staff reviewed pertinent technical information including the analyses, engineering drawings, design assumptions and the technical bases for the bounding low-trajectory turbine missile parameters (mass, size and velocity), which are to be applied to the barriers for verification that the barriers are designed to withstand local and overall effects of missile impact loadings from postulated turbine missiles that bound turbine generator sets to be used in the NuScale design. In addition, the staff reviewed FSAR Section 3.5.1.3.3, for the size, mass and velocity of the bounding low-trajectory missile. The staff finds that the size of the bounding turbine missile of half of the last stage of the turbine rotor with the blades attached is acceptable based on past turbine missile experience and because it has the largest kinetic energy of the three missiles postulated. Past operating experience includes that in NUREG1275, “Operating Experience Feedback Reports,” Volume 11, “Turbine-Generator Overspeed Protection Systems,” issued April 1995, and guidance from Electric Power Research Institute (EPRI) Report 1006451, “Technical Approach to Turbine Missile Probability Assessment,” issued December 2001, and EPRI Report 1001267, “Assessment of Turbine Missile Probability: Technical and Regulatory Issues,” issued December 2000. The staff concluded that the turbine missile selection of half a rotor with the associated blades attached is consistent with RG 1.115, Revision 2, because it evaluates a turbine missile from a fractured rotor.

For the bounding turbine missile mass in FSAR Table 3.5-1, the staff finds the applicant's proposed bounding missile weight of 2,953.3 kilograms (kg) (6,511 pounds (lb)) acceptable since it is based on a typically sized turbine of 77 megawatts electric (MWe) with a 136.6-centimeter (cm) (54.6-in.) diameter and 34.3-cm (13.5-in.) wide last stage turbine rotor, and a typical turbine rotor material in accordance with American Society for Testing and Materials (ASTM) A470 "Standard Specification for Vacuum-Treated Carbon and Alloy Steel Forgings for Turbine Rotors and Shafts" Class 4 as specified in Table 10.2-1 of the NuScale FSAR.

The staff also finds that the applicant's proposed bounding missile speed of 874 ft/sec is acceptable since it is based on a destructive overspeed of 190 percent, which is consistent with operating experience and guidance from EPRI Report 1006451 and EPRI Report 1001267 where turbine missiles from fractured rotor speeds could be as high as 180 to 190 percent. The determination of the speed of half of the rotor also included the blades. The staff finds this acceptable because when the rotor fractures, the blades are still attached, thereby increasing the centroid of the half rotor and the resulting speed of the missile. In addition, to provide reasonable assurance, the staff confirmed by an evaluation (similar to ASME Code, Section III, Appendix F, used for pump flywheels and other turbine missile analysis) that the total stresses due to centrifugal forces at 190 percent overspeed reached the tensile strength of the ASTM A470 Class 4 material. As the operating stresses in the rotor reach the tensile strength of material, ductile fracture can occur, leading to a destructive overspeed fracture event. Therefore, the staff has reasonable assurance that 190 percent overspeed is the bounding turbine missile speed.

Also, the staff finds COL Information Item 3.5-1 acceptable because it directs the COL applicant to confirm that the selected turbine design parameters are bounded by the parameters used in the NuScale FSAR, analysis for the size, weight, and speed of a postulated low-trajectory turbine missile from the last stage of the turbine rotor.

Section 3.5.3 of this SER provides the results of the analysis of the barriers using the above bounding turbine missile parameters to protect essential SSCs.

3.5.1.3.5 Combined License Information Items

SER Table 3.5.1.3-1 below lists the COL information item number and description related to turbine missiles from FSAR Table 1.8-1.

Table 3.5.1.3-1: NuScale COL Information Items for Section 3.5.1

Item No.	Description	FSAR Section
COL Item 3.5-1	An applicant that references the NuScale Power Plant US460 standard design will demonstrate the site-specific turbine missile parameters are bounded by the standard design analysis, or provide a missile analysis using the site-specific turbine generator parameters to demonstrate that barriers adequately protect essential structures, systems, and components from turbine missiles. Parameters to verify are: limiting turbine missile spectrum (rotor and blade material properties); turbine rotor design, geometry and number of blades; final	3.5.1.3

	design of the Reactor Building exterior wall; and location of the turbines with respect to the Reactor Building.	
COL Item 3.5-3	An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific hazards due to external events, such as turbine failures that can occur at nearby or co-located facilities, which may produce more energetic missiles that impact different locations than the design-basis missiles defined in Section 3.5.1.	3.5.1.3

3.5.1.3.6 Conclusion

Based on the above, including the COL Items, the staff finds that the turbine generator is in an unfavorable orientation with respect to essential SSCs, and therefore, assurance that essential SSCs are protected from the adverse effects of low-trajectory turbine missiles will be provided by the use of barriers and redundancy of SSCs. The staff concludes that the bounding low-trajectory turbine missile is half of the last stage of the rotor with the blades attached and is used in evaluating whether the barriers can protect essential SSCs. Using this bounding turbine missile, in combination with physical separation of redundant safety-related equipment, the applicant demonstrated that essential SSCs are protected from postulated low-trajectory turbine missiles using barriers as discussed in Section 3.5.3 of this SER. The staff bases this conclusion on the applicant having sufficiently demonstrated, in accordance with the guidance of DSRS Sections 3.5.1.3 and 3.5.3 and RG 1.115, Revision 2, that the barriers protect essential SSCs from postulated turbine missiles and therefore meet the relevant requirements of GDC 4.

3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

3.5.1.4.1 Introduction

This section identifies and evaluates missiles generated by extreme winds (such as a tornado or hurricane). A COL applicant that references the NuScale Power Plant US460 standard design will assess whether the actual site characteristics fall within the site parameters specified for the NuScale design (COL Item 2.0-1). If a site characteristic does not fall within the corresponding site parameter, the COL applicant will evaluate the potential for other missiles generated by natural phenomena and the potential impact of these missiles on the missile protection design features of the NuScale plant design.

3.5.1.4.2 Summary of Application

FSAR: FSAR Section 3.5.1.4, “Missiles Generated by Extreme Winds,” describes the spectrum of missiles generated by extreme winds and includes a rigid missile that tests penetration resistance (pipe), a massive high -kinetic -energy missile that deforms on impact (automobile), and a small rigid missile of a size that is sufficient to pass through openings in protective barriers (small steel sphere).

ITAAC: There are no ITAAC directly associated with missiles generated by tornadoes and extreme winds. FSAR Table 2.0-1 provides the parameters for design-basis tornado and hurricane winds and associated missile spectra.

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

Technical Reports: There are no TRs associated with missiles generated by tornadoes and extreme winds.

3.5.1.4.3 Regulatory Basis

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

- In GDC 2, the NRC requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions.

- In GDC 4, the NRC requires, in part, that SSCs important to safety shall be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.

SRP Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," provides the relevant regulatory requirements as well as interfaces with other relevant SRP sections.

- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," describes acceptable design-basis tornado-generated missile spectra for the design of nuclear power plants.

- RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," describes acceptable design-basis hurricane-generated missile spectra for the design of nuclear power plants.

- The method of identifying appropriate design-basis missiles generated by natural phenomena should be consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in SRP Section 2.2.3, "Evaluation of Potential Accidents." A licensee or applicant may justify the acceptability of the use of another methodology.

3.5.1.4.4 Technical Evaluation

The staff reviewed the NuScale design for protecting SSCs important to safety against missiles generated by extreme winds in accordance with the guidance of SRP Section 3.5.1.4. The staff reviewed FSAR Section 3.5.1.4. The staff also reviewed FSAR Chapter 2, "Site Characteristics and Site Parameters," and other FSAR sections noted below.

FSAR Section 3.5.1.4, describes design-basis tornado and hurricane winds and associated missile spectra for the NuScale design as follows:

- design-basis extreme wind parameters (FSAR Section 3.3.1 and Table 2.0-1)
  - A tornado has a maximum wind speed of 120.7 m/s (270 mph).

- A hurricane has a maximum wind speed of 129.64 m/s (290 mph).

- tornado-generated missile spectra

  - A massive high-kinetic-energy missile that deforms on impact, such as an 1,814.4-kg (4,000-lb) automobile with dimensions of 5.00 m by 2.01 m by 1.31 m (16.4 ft by 6.6 ft by 4.3 ft), has a horizontal velocity of 41.1 m/s (135 feet per second (ft/s)) and a vertical velocity of 27.7 m/s (91 ft/s).

  - A rigid missile that tests penetration resistance, such as a 4.6m- -long, 130.2-kg, 15cm- -diameter (15-ft-long, 287-lb, 6-in.-diameter) Schedule 40 pipe, has a horizontal velocity of 41.1 m/s (135 ft/s) and a vertical velocity of 27.7 m/s (91 ft/s).

  - A small rigid missile of a size that is sufficient to pass through openings in protective barriers, such as a 66.7 gram (g), 2.5- cm- -diameter (0.147-lb, 1-in.-diameter) solid steel sphere, has a horizontal velocity of 7.9 m/s (26 ft/s) and a vertical velocity of 5.5 m/s (18 ft/s).

- hurricane--generated missile spectra

  - A massive high-kinetic-energy missile that deforms on impact, such as the automobile described above, has a horizontal velocity of 93.6 m/s (307 ft/s).

  - A rigid missile that tests penetration resistance, such as the Schedule 40 pipe described above, has a horizontal velocity of 76.5 m/s (251 ft/s).

  - A small rigid missile of a size that is sufficient to pass through openings in protective barriers, such as the solid steel sphere described above, has a horizontal velocity of 68.6 m/s (225 ft/s).

  - The design-basis vertical missile velocity for all hurricane missiles is 25.9 m/s (85 ft/s).

The applicant has assumed that the automobile missiles will impact at all altitudes of less than 9 m (30 ft) above plant grade levels if initially located within 0.8 kilometers (km) (0.5 miles (mi)) of the plant structures. FSAR Section 3.5.2, states that the portions of the RXB and CRB that are above the 9-m (30-ft) plant elevation have not been analyzed to withstand the design -basis automobile missile, but they are resistant to the other design -basis missiles. To ensure protection from the missiles defined in SRP Section 3.5.1.4, as indicated in FSAR Section 3.5.3, concrete thicknesses to preclude impact from the design-basis hurricane and tornado pipe and sphere missiles are calculated for the RXB, CRB, and RW-IIa portion of the Radioactive Waste Building external walls and roof. In addition, COL Item 3.5--2 states the COL applicant will confirm that design-basis automobile missile parameters for the reference plant of velocity and maximum altitude of impact will not be exceeded as a result of extreme wind conditions that may occur in the vicinity of the site.

The staff reviewed the above information and finds it acceptable because applying the automobile missile only to elevations below 9 m (30 ft) is consistent with the guidance of RG 1.76, Revision 1, and RG 1.221, Revision 0. In addition, the staff finds that COL Item 3.5-2 addresses the potential for an automobile missile impact higher than 9 m (30 ft).

In addition, the staff reviewed site characteristics contained in FSAR Table 2.0-1, and COL Item 2.0-1, which requires applicant to demonstrate the acceptability of the site-specific values in the appropriate sections of its license application if the site-specific values (including high winds and missiles) are not bounded.

As noted in FSAR Table 1.9-2, NuScale design conforms to RG 1.76, Revision 1, by stating, "Bounding or greater than design parameters postulated by RG 1.76 are used. Confirming the characteristics is the responsibility of the applicant or licensee," and conforms to RG 1.221. Additional actions by COL Applicant are contained in COL Items 2.3-1, COL 3.3-1, and COL 3.5-2 to ensure conformance with RG 1.76 and RG 1.221.

The guidance of RG 1.76 applies only to the continental United States, which is divided into three regions: Region I, the central portion of the United States; Region II, a large region of the United States along the east coast, the northern border, and western Great Plains; and Region III, the western United States. The tornado parameter values specified in RG 1.76, Table 1, for Region I, are most severe and bound all the tornado parameter values specified for Regions II and III. The staff finds that the above design -basis tornado parameters provided by the applicant and tornado -generated missile spectra are in accordance with the guidance in RG 1.76, Table 1, for Region I.

RG 1.221 provides contour maps of the U.S. coastal areas most susceptible to hurricanes and associated design--basis wind and missile speeds. The staff finds that the above design--basis hurricane parameters and hurricane-generated missile spectra for the NuScale Power Plant US460 standard design are in accordance with the guidance in RG 1.221.

SER Section 2.3 contains the staff's evaluation of the meteorological site parameters. The staff evaluates the structural performance of the NuScale design with respect to hurricane and tornado missiles in SER Section 3.8.

Based on its review, the staff finds that the information provided by the applicant conforms to the guidance in RG 1.76, Revision 1, and RG 1.221, Revision 0, for design-basis tornado and hurricane missiles, respectively. Therefore, the staff concludes that the NuScale design meets the requirements of GDC 2 and GDC 4, with respect to the protection of SSCs important to safety from the effects of natural phenomena such as tornadoes and hurricanes.

3.5.1.4.5 Combined License Information Items

SER Table 3.5.1.4-1 lists the COL information item number and description (obtained from FSAR Table 1.8-1) that are related to FSAR Section 3.5.1.4.

Table 3.5.1.4-1: NuScale COL Information Items for Section 3.5.1

Item No.	Description	FSAR Section
COL Item 3.5-2	An applicant that references the NuScale Power Plant US460 standard design will confirm the design-basis automobile missile parameters for the reference plant of velocity and maximum altitude of impact will not be exceeded as a result of extreme wind conditions that may occur in the vicinity of the site.	3.5.1.4

Item No.	Description	FSAR Section
COL Item 3.5-3	An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific hazards due to external events, such as turbine failures that can occur at nearby or co-located facilities, which may produce more energetic missiles that impact different locations than the design-basis missiles defined in Section 3.5.1.	3.5.1.5

3.5.1.4.6 Conclusion

The staff's review concludes that the applicant's design-basis tornado and hurricane-generated missile spectra for the NuScale design comply with the requirements in 10 CFR Part 50, Appendix A, GDC 2 and GDC 4, for SSCs to be protected from missiles generated by extreme winds because the applicant meets the acceptance criteria in SRP 3.5.1.4 and conforms to the guidance in RG 1.76 and RG 1.221 for design--basis wind-borne missiles for nuclear power plants. Staff also finds the COL Items acceptable as described above.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

3.5.1.5.1 Introduction

The COL applicant will analyze and establish the site--specific missile spectra. The potential threat to the plant from site proximity missiles is site specific and therefore cannot be assessed at the SDA stage.

3.5.1.5.2 Summary of Application

FSAR: In FSAR Section 3.5.1.5, the applicant stated that, as described in Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," the NuScale Power Plant US460 standard design does not postulate any hazards from nearby industrial, transportation or military facilities. Therefore, there are no proximity missiles evaluated in the application.

ITAAC: There are no ITAAC associated with Section 3.5.1.5.

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

Technical Reports: There are no TRs for Section 3.5.1.5.

3.5.1.5.3 Regulatory Basis

In 10 CFR 52.137(a)(1), the NRC requires the SDA applicant to provide site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters.

In addition to 10 CFR 52.137(a)(1), the applicable regulatory requirements for identifying and evaluating site proximity missiles include the following:

- 10 CFR 100.20(b), as it requires the nature and proximity of human -related hazards (e.g., airports, dams, transportation routes, and military or chemical facilities) to be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards and whether the risk of other hazards is very low

- 10 CFR 100.21(c)(2), as it requires the applications for site approval for commercial power reactors to demonstrate that the proposed site meets the radiological dose consequences of postulated accidents that meet the criteria in 10 CFR 50.34(a)(1)

- 10 CFR 100.21(e), as it requires potential hazards associated with nearby transportation routes and industrial and military facilities to be evaluated and site parameters to be established to ensure that potential hazards from such routes and facilities will not pose an undue risk to the type of facility proposed to be located at the site

- GDC 4, as it requires SSCs important to safety to be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit

The following guidance, which is provided in SRP Section 3.5.1.5, “Site Proximity Missiles (Except Aircraft),” provides a means for the applicant to meet the relevant requirements:

- The criteria typically involve reviewing the event probability for which the expected rate of occurrence of potential exposure in excess of the 10 CFR Part 100 guidelines is estimated to be less than an order of magnitude of 1×10^{-7} per year.

3.5.1.5.4 Technical Evaluation

Because the information on site proximity is not available at the SDAA stage, the COL applicant will describe the missile, including its size, shape, weight, energy, material properties, and trajectory, and will develop and address the missile effects on the SSCs, if necessary. As noted in FSAR Table 1.8-1, COL Item 2.2-1 directs a COL applicant that references the NuScale Power Plant US460 standard design to demonstrate that the design is acceptable for each accident scenario, which includes site proximity explosions and missiles, or to provide site-specific design alternatives. COL Item 3.5-3 directs a COL applicant that references the NuScale Power Plant US460 standard design to address the effect of turbine missiles from nearby or collocated facilities.

3.5.1.5.5 Combined License Information Items

SER Table 3.5.1.5-1 lists the COL information item number and description (obtained from FSAR Table 1.8-1) that are related to FSAR Section 3.5.1.5.

Table 3.5.1.5-1: NuScale COL Information Items for Section 3.5.1

Item No.	Description	FSAR Section
COL Item 2.2-1	An applicant that references the NuScale Power Plant US460 standard design will describe nearby industrial, transportation, and military facilities. The applicant will demonstrate that the design is acceptable for each of these potential hazards, or provide site-specific design alternatives.	2.2

Item No.	Description	FSAR Section
COL Item 3.5-3	An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific hazards due to external events, such as turbine failures that can occur at nearby or co-located facilities, which may produce more energetic missiles that impact different locations than the design-basis missiles defined in Section 3.5.1.	3.5

3.5.1.5.6 Conclusion

As described above, the COL applicant will provide the —site-specific information under COL Item 2.2-1 and COL Item 3.5-3. Because this information is site specific, the applicant's statement in the FSAR that the COL applicant will supply this site-specific information, and as called for in COL Item 2.2-1 and COL Item 3.5-3 in accordance with SRP Section 3.5.1.5, is considered acceptable. For the reasons given above, the staff concludes that, as this information is site specific, the COL applicant will address it, and therefore, the staff will review the information at the COL stage. This should include the provision of information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL application and the requirements delineated in Section 3.5.1.3 of this report are satisfied.

3.5.1.6 Aircraft Hazards

3.5.1.6.1 Introduction

This section reviews whether the risks from aircraft hazards are sufficiently low. The COL applicant will demonstrate acceptability of the site parameters with respect to aircraft hazards. Additional site-specific analyses may be required at the COL stage.

3.5.1.6.2 Summary of Application

FSAR: In FSAR Section 3.5.1.6, the applicant stated that, as described in Section 2.2, the NuScale Power Plant US460 standard design does not postulate any hazards from nearby industrial, transportation, or military facilities. Therefore, no design--basis aircraft hazards are evaluated in the application.

ITAAC: There are no ITAAC associated with Section 3.5.1.6.

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

Technical Reports: There are no TRs associated with Section 3.5.1.6.

3.5.1.6.3 Regulatory Basis

In 10 CFR 52.137(a)(1), the NRC requires the DC applicant to provide site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters.

In addition to 10 CFR 52.137(a)(1), the following are the applicable regulatory requirements for identifying the evaluation of potential aircraft hazards:

- 10 CFR 100.20(b), as it requires the nature and proximity of human related hazards (e.g., airports) to be evaluated to establish site parameters for use in determining whether a plant design can accommodate commonly occurring hazards and whether the risk of other hazards is very low

- 10 CFR 100.21(c)(2), as it requires the applications for site approval for commercial power reactors to demonstrate that the proposed site meets the radiological dose consequences of postulated accidents that meet the criteria in 10 CFR 50.34(a)(1)

- 10 CFR 100.21(e), as it requires the potential hazards associated with nearby transportation routes and industrial and military facilities to be evaluated and site parameters to be established such that potential hazards from such routes and facilities will pose no undue risk to the type of facility proposed to be located at the site

- GDC 3, “Fire Protection,” as it requires that SSCs important to safety be designed and located to minimize the probability and effect of fires and explosions

- GDC 4, as it requires SSCs important to safety to have appropriate protection against the effects of missiles that may result from events and conditions outside the nuclear power units

The following guidance, which is provided in SRP Section 3.5.1.6, “Aircraft Hazards,” provides a means for the applicant to meet the relevant requirements:

- The criteria typically involve reviewing the event probability for which the expected rate of occurrence of potential exposure in excess of the 10 CFR Part 100 guidelines is estimated to be less than an order of magnitude of $1x10^{-7}$ per year.

3.5.1.6.4 Technical Evaluation

Because the information on potential aircraft hazards near the site is site specific, the applicant stated that the COL applicant that references the NuScale Power Plant US460 standard design will be directed to demonstrate that the design is acceptable for each potential accident, which includes aircraft impact, in accordance with COL Item 2.2-1 in FSAR Table 1.8-1.

3.5.1.6.5 Combined License Information Items

SER Table 3.5.1.6-1 lists the COL information item number and description (obtained from FSAR Table 1.8-1) that is related to FSAR Section 3.5.1.6.

Table 3.5.1.6-1: NuScale COL Information Item for Section 3.5.1

Item No.	Description	FSAR Section
COL Item 2.2-1	An applicant that references the NuScale Power Plant US460 standard design will describe nearby industrial, transportation, and military facilities. The applicant will demonstrate that the design is acceptable for each of these potential hazards, or provide site-specific design alternatives.	3.5.1.6

3.5.1.6.6 Conclusion

As described above, the applicant stated, in FSAR Section 2.2, that the COL applicant will provide the site-specific information under COL Item 2.2-1. Because this information is site specific, the applicant's statement in the NuScale SDAA that the COL applicant will provide this site-specific information as called for in COL Item 2.2-1, in accordance with SRP Section 3.5.1.6, is considered acceptable. The COL applicant should include information sufficient to demonstrate that the design of the plant falls within the values of the actual site characteristics specified in a COL application and the requirements delineated in Section 3.5.1.6.3 of this report are satisfied.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

3.5.2.1 Introduction

The staff's review guidance in SRP Section 3.5.2, Revision 3, "Structures, Systems, and Components to be Protected from Externally-Generated Missiles," states that to satisfy GDC 2 and GDC 4, SSCs needed to safely shut down the reactor and maintain it in a safe condition should be protected from externally generated missiles. This includes all safety -related SSCs and risk -significant SSCs requiring missile protection that support the operation of the reactor.

3.5.2.2 Summary of Application

FSAR: FSAR Section 3.5.2, "Structures, Systems, and Components to be Protected from External Missiles," describes how SSCs requiring protection from externally generated missiles are protected by locating these SSCs inside Seismic Category I structures. The external walls and roofs of the structures provide missile protection.

ITAAC: There are no ITAAC directly associated with SSCs to be protected from external missiles.

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

Technical Reports: There are no TRs associated with SSCs to be protected from external missiles.

3.5.2.3 Regulatory Basis

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

- GDC 2, as it requires, in part, SSCs important to safety to be designed to withstand the effects of natural phenomena, such as tornadoes and hurricanes, without loss of capability to perform their safety functions

- GDC 4, as it requires, in part, SSCs important to safety to be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit

SRP Section 3.5.2, Revision 3, provides the relevant regulatory requirements, as well as interfaces with other SRP sections.

An applicant can meet the requirements of GDC 2 and GDC 4 by conforming to the guidance in the following RGs:

- RG 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis," issued March 2007, as it relates to the capacity of the spent fuel pool cooling systems and structures to withstand the effects of externally generated missiles and to prevent missiles from contacting the stored fuel assemblies

- RG 1.117, Revision 2, Appendix A, as it relates to which SSCs important to safety should be protected from missile impacts generated by tornadoes

3.5.2.4 Technical Evaluation

The staff reviewed the NuScale design for protecting SSCs against externally generated missiles in accordance with the guidance in SRP Section 3.5.2, Revision 3. The staff reviewed FSAR Section 3.5.2 and other sections of the FSAR as noted below.

SRP Section 3.5.2, Revision 3 states that the SSCs required for safe shutdown of the reactor should be identified. RG 1.117, Appendix A provides guidance as to which SSCs should be protected from missile impacts. Classification of Structures, Systems, and Components tables are included in individual plant system FSAR sections to identify the SSCs that are safety-related and risk-significant, and FSAR Table 9A-7, "Safe Shutdown Plant Functions," identifies the SSCs that are needed for safe shutdown.

The staff reviewed the application for the identification of SSCs that are required to be protected against externally generated missiles. The SSCs subject to missile protection are identified in Classification of Structures, Systems, and Components tables located in individual plant system sections in the FSAR as A1 (Safety Related, Risk-Significant), A2 (Safety--Related, Non-Risk-Significant), or B1 (NonSafety-Related, Risk-Significant) and are located inside the RXB or CRB. As indicated in Section 3.5 "Missile Protection", Table 17.4-1 provides a list of risk-significant SSC that have a safety function that might be relied upon following a missile-producing event.

The staff reviewed the NuScale application to determine whether all identified SSCs necessary for supporting the reactor facilities are appropriately protected from externally generated missiles. FSAR Section 3.5.2 states that all safety -related and risk -significant SSCs that must be protected from external missiles are located in the RXB and Seismic Category I portion of CRB, which are seismic Category I structures and are designed for missile protection. The walls, roof, and openings of the Seismic Category I structures are designed to withstand the design -basis missiles described in FSAR Section 3.5.1.4. Based on the above information, the staff determined that the SSCs identified in FSAR plant system tables and Table 9A--7 as requiring missile protection are located within Seismic Category I structures and openings and will be protected. Therefore, the staff finds that this aspect of the NuScale plant design conforms to the guidance in RG 1.13 and RG 1.117. The RXB and CRB missile protection design is addressed in SER Section 3.5.3. Section 3.5.1.3 of this SER contains the staff's evaluation of turbine missiles, including the applicant's conformance to the guidance in RG 1.115. SER Section 3.5.3 addresses the staff's evaluation of the design of Seismic Category I structures and barriers used for missile protection.

3.5.2.5 Combined License Information Items

COL information items associated with this review area are listed in SER Section 3.5.1.4.5.

3.5.2.6 Conclusion

Based on the staff's review of the information in the FSAR, which is documented in the staff's evaluation set forth above, the staff concludes that the safety-related and risk-significant SSCs to be protected from externally generated missiles are inside the Seismic Category I RXB and CRB in conformance with the guidance in RG 1.13 and RG 1.117, and therefore, comply with the requirements of 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

3.5.3 Barrier Design Procedures

3.5.3.1 Introduction

The staff reviewed FSAR Section 3.5.3, "Barrier Design Procedures," following the guidance in SRP Section 3.5.3, Revision 3, "Barrier Design Procedures," issued March 2007, with regard to the procedures used in the design of Seismic Category I structures, shields and barriers to withstand the effects of windborne and turbine missile impact. The staff acknowledges the limitations in the applicability of the National Defense Research Committee (NDRC) formulas to determine penetration depth for turbine missiles and allows the use of an alternative approach as provided for in SRP Section 3.5.3. The staff considered the information provided by the applicant in the FSAR in establishing the reasonable assurance of safety conclusion.

A COL applicant that refers to the NuScale Power Plant US460 standard design will assess whether the actual data on missile parameters are within the corresponding site parameters and the turbine characteristics of the NuScale design. A COL applicant should reevaluate the SSCs important to safety in the NuScale design if the site characteristics of missiles or the characteristics of the turbine are not within the corresponding site parameters or turbine characteristics of the NuScale design.

3.5.3.2 Summary of Application

The applicant provided the barrier design procedures used for the NuScale design in FSAR Section 3.5.3. The applicant describes the design requirements for the barriers with sufficient thicknesses to protect the safety-related and risk-significant SSC from the impacts of windborne and turbine missiles. The applicant also describes the hurricane and tornado (windborne) generated missiles in FSAR Section 3.5.1.4. In addition, the applicant shows the plan views of turbine missile trajectories and the section view of the RXB turbine missile barriers in FSAR Figures 3.5-1 and 3.5-2.

In FSAR Sections 3.5.3.1 and 3.5.3.2, the applicant describes the evaluations for the effects of missile impacts for the local damage and overall damage predictions on the structures or barriers constructed with basic materials of concrete, steel and stee-plate composite.

For the prediction of local damage from windborne missiles, the applicant applied the Modified NDRC formulas for missile protection in concrete barriers, and the Stanford Formula for missile protection in steel barriers. In addition, the applicant uses the Bruhl equations for missile protection in steel-plate composite walls. With regard to the overall damage predicted for a structure or barrier from tornado and hurricane missile impact, the applicant used EPRI NP440, "Full Scale Tornado Missile Impact Tests," issued July 1977, to determine the structural responses for the triangular impulse formulation of the design -basis steel pipe missile. The applicant used the Bechtel Power Corporation Topical Report BC--TOP--9A, Revision 2, “Design of Structures for Missile Impact,” September 1974, (ML14093A217), to determine the structural responses for the design-basis automobile missile.

For the overall damage prediction, the applicant considers localized and global effects for impactive and impulsive loads for the designs of the concrete structures in accordance with ACI 349-06 and for the SC walls in accordance with NuScale TR-0920-71621-P-A, Revision 1, “Building Design and Analysis Methodology for Safety-Related Structures.”. For the overall damage prediction of the RXB and CRB walls, the applicant did not consider the effect of 1-inch solid steel sphere missile since its mass is too small to generate any structural responses.

ITAAC: There are no ITAAC directly associated with SSCs to be protected from external missiles.

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

Technical Reports: There are no TRs associated with SSCs to be protected from external missiles.

3.5.3.3 Regulatory Basis

The staff used the following NRC regulations and guidance to perform this review:

- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires SSCs important to safety to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. GDC 2 further requires design bases for these SSCs to reflect appropriate combinations of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. GDC 2 also requires appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and the importance of the safety functions to be performed.

- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires SSCs important to safety to be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

SRP Section 3.5.3, Revision 3 lists the acceptance criteria adequate to meet the above requirements and provides review interfaces with other SRP sections. For turbine missiles, the acceptance criteria of SRP Sections 3.5.3, “Barrier Design Procedures” and 3.5.1.3, “Turbine Missiles,” are applicable. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.115, Revision 2, “Protection against Turbine Missiles,” issued January 2012

- RG 1.142, Revision 3, “Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments,” issued May 2020

3.5.3.4 Technical Evaluation

The staff reviewed FSAR Section 3.5.3 to determine whether the barrier design procedures used in the NuScale design meet the guidelines of SRP Section 3.5.3, Revision 3, and the requirements of GDC 2 and GDC 4, with respect to the capabilities of the Seismic Category I structures, shields, and barriers to withstand the effects of windborne and turbine missile impacts.

3.5.3.4.1 Local Damage Prediction

In FSAR Section 3.5.3.1, the applicant describes that the predictions of local damage in the impact area are based on material of construction of the structures or barrier. The applicant describes that barriers are installed if a missile-and-target combination is determined to be statistically significant for the limited number of potential internal missiles and a limited number of targets.

For the concrete missile barriers, the applicant describes in FSAR Section 3.5.3.1.1, “Concrete Barriers,” that the predictions of penetration/spalling, perforation and spalling of the concrete missile barriers are based on the empirical equations of the modified NDRC formulas as provided in FSAR Sections 3.5.3.1.1.1, 3.5.3.1.1.2 and 3.5.3.1.1.3, respectively. Further, the applicant also increases the calculated concrete barrier thicknesses for perforation and scabbing by 20 percent per the requirements in F7.2.1 and F7.2.2 of ACI 349-06, “Code Requirements for Nuclear Safety Related Concrete Structures and Commentary.” The staff reviewed FSAR Section 3.5.3.1.1 and finds that the applicant applied the Modified NDRC formulas for missile protection in concrete barriers, which is consistent with SRP Section 3.5.3, Acceptance Criterion II.1.A. The staff confirmed that the calculated concrete barrier thickness increased by 20 percent and is in accordance with the requirements described in Sections F.2.1 and F7.2.2 of ACI 349-06. Based on this review, the staff finds the concrete barrier design procedures used for the prediction of local damage in the impacted area to be acceptable.

For the steel barriers, the applicant describes in FSAR Section 3.5.3.1.2 that the analytical requirements for determining the minimum steel thickness to prevent perforation in the steel missile barriers. During an audit, the applicant described both Stanford and BRL formulas are used to determine the minimum steel missile barriers, however, due to limitations with the Stanford equation, the final design thicknesses are based on the results of BRL formula that the thicknesses are also increased by 25 percent per the recommendations described in Topical Report, BC-TOP-9A, Revision 2 (ML14093A217). The staff reviewed FSAR Section 3.5.3.1.2 and finds that the applicant applied the BRL Formula for missile protection in steel barriers, which is consistent with SRP Section 3.5.3, Acceptance Criterion II.1.B. The staff confirmed that the calculated steel barrier thickness increased by 25 percent is in accordance with the recommendations described in BC-TOP-9A, Revision 2. Based on this review, the staff finds the steel barrier design procedures used for the prediction of local damage in the impacted area to be acceptable.

For the steel-plate composite wall barriers, the applicant describes in FSAR Section 3.5.3.1.3 that the three-step design approach to prevent perforation on the faceplate of SC walls for various missile impacts by Bruhl. During an audit, the applicant referred to NuScale TR-0920-7162-P-A, Revision 1, that describes that the three-step approach to design an SC Wall for a specific missile, where a factor 1.25 is also used to satisfy the requirement of Section N9.1.6c of American Institute of Steel Construction (AISC) N690-18 for the thickness of the faceplate to be

25 percent greater than the calculated faceplate thickness. The staff reviewed FSAR Section 3.5.3.1.3, and the tropical report TR-0920-71621-P-A, Revision 1, and notes that the applicant used a three-step approach to calculate the require faceplate thickness. This three-step approach using the Bruhl equations is accepted based on the staff's SE (ML22056A308) on TR-0920-71621, Revision 1.

3.5.3.4.2 Overall Damage Prediction

In FSAR Section 3.5.3.2, the applicant describes the design of localized and global effects for impactive and impulsive loads in accordance with ACI 349-06 of structural concrete members and NuScale Topical Report TR-920-71621-P for SC Walls.

In FSAR Section 3.5.3.2, the applicant describes that the design basis steel sphere missile is too small to affect the structural response of the RXB and CRB and was not evaluated for its contribution to overall structural response. The staff finds this acceptable because the RXB and CRB are designed to absorb the localized impact energy of the design basis sphere missile through robust structural components, ensuring that overall structural response of the RXB and CRB remains unaffected.

In FSAR Section 3.5.3.2, the applicant used Nuclear Engineering and Design, "Full-Scale Tornado-Missile Impact Tests" to determine the structural responses for the triangular impulse formulation of the design basis steel pipe wind-generated missile. The applicant used BC-TOP-9A, Revision 2, to determine the structural responses for the design basis automobile missile. The staff reviewed FSAR Section 3.5.3.2 and finds that the applicant used approaches that differ from that specified in SRP Section 3.5.3, Acceptance Criterion II.2. The staff reviewed the Nuclear Engineering and Design, "Full-Scale Tornado-Missile Impact Tests" and BC-TOP-9A, Revision 2, and finds that Nuclear Engineering and Design, "Full-Scale Tornado-Missile Impact Tests" and BC-TOP-9A are both widely used and accepted missile impact references. The results of these analyses demonstrate that missile impact has virtually no effect on the overall response of structures as large as the RXB and CRB.

In FSAR Section 3.5.3.2, the applicant describes that the design for impulsive and impactive loads is in accordance with ACI 349-06 for concrete structures, and American National Standards Institute (ANSI)/AISC N690-18, "Specification for Safety Related Steel Structures for Nuclear Facilities," for steel structures, which are acceptable to the staff because their design is in accordance with ACI 349-06 and N690-19. The applicant also describes that the design for impulsive and impactive loads in accordance with the NuScale Topical Report, TR-920-71621, "Building Design and Analysis Methodology for Safety-Related Structures," for SC walls. The staff reviewed FSAR Section 3.5.3.2 and the NuScale Topical Report, TR-920-71621, and finds the applicant's design for impulsive and impactive loads acceptable because the methodology used for the design of impulsive and impactive loads has been approved by the staff and documented in the staff's SE of the Building Design and Analysis Methodology for Safety-Related Structures (ML22056A308). The applicant further describes that stress and strain limits for the missile impact equivalent static load conform to applicable codes and RG 1.142, Revision 3 and that the ductility factors for steel structures from Section NB3.14, Table NB3.1 of AISC N690-18 are used in accordance with requirements of SRP 3.5.3. Based on its review of the information that the applicant provided, the staff finds the applicant's proposed alternative procedures acceptable.

3.5.3.5 Combined License Information Items

No COL information items from the SDAA affect this section

3.5.3.6 Conclusion

Based on the above review, the staff finds the procedures used for determining the effects and loadings on Seismic Category I structures and missile shields and barriers induced by design -basis hurricane and tornado missiles selected for the plant to be acceptable because these procedures provide an adequate basis for engineering design to ensure that the structures or barriers are adequately resistant to, and will withstand the effects of, such forces. The staff concludes that the conformance with these procedures is an acceptable basis for satisfying the requirements of 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

With regard to the procedures used for determining the effects and loadings on the RXB and CRB wall barriers from the impact of turbine missiles, the staff finds the procedures used for determining the effects and loadings on Seismic Category I structures induced by design -basis turbine missiles to be acceptable because these procedures provide an adequate basis for engineering design to ensure that the structures or barriers are adequately resistant to, and will withstand the effects of, such forces. The staff concludes that conformance with these procedures is an acceptable basis for satisfying the requirements of 10 CFR Part 50, Appendix A, GDC 4.

3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside Containment

3.6.1.1 Introduction

This section evaluates the NuScale design bases and criteria relied upon to demonstrate that essential systems and components are protected against postulated piping failures outside containment. It identifies high- and moderate-energy systems representing potential sources of dynamic and environmental effects associated with pipe rupture and defines the criteria for the separation and evaluation of adverse consequences.

3.6.1.2 Summary of Application

FSAR: FSAR Section 3.6, “Protection against Dynamic Effects Associated with Postulated Rupture of Piping,” discusses information related to protection against pipe rupture effects. FSAR Table 3.6-1 identifies the piping systems associated with high- and moderate-energy piping systems inside the CNV and NuScale Power Module Bay (under the bioshield).

FSAR Section 3.6.1 describes the methodology used in designing the protection of essential systems and components from the consequences of postulated piping failures outside containment. Such methodology includes the identification of (1) systems and components located near high- or moderate-energy pipe systems that need to be protected, (2) failures for which protection is being provided and assumptions are being used, and (3) protection considerations in the design. In addition, Section 3.6.1 addresses the separation and redundancy of essential systems and methods for analyzing piping failures.

ITAAC: SDAA Part 8, “License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC),” Table 2.1-2, includes ITAAC 02-01-04, which requires the completion of the as--built pipe break hazard analysis report that ensures that the safety -related SSCs are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate -energy piping systems. This ITAAC is evaluated in Section 14.3 of this SER.

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

Technical Reports: In FSAR Section 3.6, “Protection against Dynamic Effects Associated with Postulated Rupture of Piping,” the applicant identified NuScale TR-121507-P, Revision 0, “Pipe Rupture Hazards Analysis,” dated December 2022 (ML23001A003), as providing an analysis of the design bases and measures needed to protect safety -related and essential systems and components inside and outside containment against the effects of postulated pipe rupture.

3.6.1.3 Regulatory Basis

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

- GDC 2, as it requires the protection of SSCs important to safety to withstand the effects of natural phenomena, such as earthquakes

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

The guidance in SRP Section 3.6.1, Revision 3, issued March 2007, provides the relevant regulatory requirements, as well as review interfaces with other SRP sections.

3.6.1.4 Technical Evaluation

In FSAR Section 3.6.1, the applicant described the methodology used in designing the protections for the essential systems and components from the consequences of postulated piping failures outside containment. The steps include the identification of (1) the essential systems and components that are located near high- or moderate-energy piping systems, (2) the failures for which protection is being provided, and (3) the protection considerations that are used in the design to safeguard essential SSCs. The applicant defined essential systems and components as those SSCs that are required to shut down the reactor and to mitigate the consequences of the postulated piping rupture. In addition, the applicant proposed to protect the post-accident monitoring (PAM) functionality provided by various portions of the instrumentation and controls (I&C), even though the equipment is neither safety-related nor essential.

In FSAR Table 3.6-1, “High- and Moderate-Energy Fluid System Piping in the Containment Vessel and NuScale Power Module Bay,” the applicant identified the fluid systems that contain high- and moderate-energy piping.

In TR-121507-P, Table 1-2, the applicant defined a high-energy system as a fluid system or portions of a fluid system that, during normal plant conditions, is either in operation or is maintained pressurized under conditions that meet one or both of the following:

- The maximum operating temperature exceeds 93.3 degrees Celsius (C) (200 degrees Fahrenheit (F)).

* The maximum operating pressure exceeds 1.90 MPa (gauge) (275 psig).

The applicant defined a moderate-energy system as a high-energy system that operates only at those conditions for short periods of time (less than 2 percent of the total time the system operates) or as a fluid system or portions of a fluid system that, during normal plant conditions, are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

* The maximum operating temperature is 93.3 degrees C (200 degrees F) or less.
* The maximum operating pressure is 1.90 MPa (gauge) (275 psig) or less.

The reviews of previous nuclear power plant designs indicated that the functional or structural integrity of systems and components required for safe shutdown of the reactor and maintenance of cold-shutdown conditions could be endangered by fluid system piping failures at locations outside containment. The staff has identified an acceptable approach for the design and arrangement of fluid systems located outside of containment to ensure that the plant can be safely shut down in the event of piping failures outside containment. SRP Branch Technical Position (BTP) 3-3, Revision 3, “Protection against Postulated Piping Failures in Fluid Systems Outside Containment,” issued March 2007, and its companion BTP 3-4, Revision 3, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” issued 2016, describe this approach.

The staff evaluated the applicant’s definitions of high- and moderate-energy systems and found them to be consistent with the definitions provided in BTP 3-3, which delineates the staff’s guidelines for protection against postulated piping ruptures in fluid systems outside the containment. The staff finds the system definitions above to be acceptable.

3.6.1.4.1 General Design Criterion 2

The requirements in 10 CFR Part 50, Appendix A, GDC 2, state that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. During a seismic event, it is postulated that non-seismic SSCs could fail. This section evaluates the impact of full--circumferential ruptures of non-seismic moderate-energy piping in areas close to SSCs important to safety where the effects of a failure are not already bounded by failures of high-energy piping. Acceptance criteria are based on conformance to SRP BTP 3-3.

In TR-121507-P, Section 3.1.1, the applicant stated that the simplicity and passive safety features of the design result in a small number of SSC being required for reactor shutdown and core cooling. The following systems and components are credited to ensure safe shutdown of the reactor and require protection against high-energy line breaks (HELBs):

* RCS
* MPS
* neutron monitoring system
* CVCS
* control rod assembly (CRA) and the control rod drive system (CRDS)
* CNTS
* DHRS
* SGS
* ECCS

- CRB
- ICIS
- RBCM
- UHS

FSAR Table 3.6-1 identifies high and moderate energy lines in CNV and NPM bay. In FSAR Table 3.6-1, the applicant identified the high- and moderate-energy piping systems and the areas where the systems are located. The applicant divided the evaluation into the following five areas:

(1) inside the containment vessel (CNV)
(2) outside the CNV (under the bioshield)
(3) in the RXB (outside the bioshield)
(4) in the CRB
(5) on site (outside the RXB and CRB buildings)

SER Section 3.6.2 evaluates the protection against pipe failure inside the containment.

The HELBs are considered outside the CNV (under the bioshield) and inside the RXB (outside the bioshield), and Section 3.6.1.4.2 of this report evaluates the consequences of these failures. Accordingly, the staff finds that the consequences of full-circumferential ruptures of non-seismic moderate-energy piping are bounded by high-energy failures.

The CRB has no high-energy lines with the exception of the piping associated with the high-pressure breathing air room habitability system. The failure of moderate-energy piping is evaluated for flooding environmental conditions. SER Section 3.4 evaluates flooding, and Section 3.11 of this report evaluates the environmental conditions caused by pipe failure.

No essential equipment is located in the Radioactive Waste Building (RWB) or the outside buildings; therefore, pipe failure in those areas will not affect essential equipment.

The staff finds that the applicant identified the equipment that requires protection. Sections 3.6.1.4.2 and 3.6.2 of this report document the staff's evaluation of the adequacy of the protection of essential SSCs from the impact of full-circumferential ruptures of non-seismic, moderate-energy piping because the applicant has used the "separation" criteria to protect SSCs that are important to safety and because there are no SSCs important to safety outside the RXB or CRB that require protection. The protection measures credited in the CNV are discussed in Section 3.6.2 of this report. The protection measures for the RXB are discussed in Section 3.6.1.4.2.3 of this report. Therefore, the staff finds that the above system description is acceptable in reference to the applicable requirements of GDC 2.

3.6.1.4.2 General Design Criterion 4

The plant design for protection against postulated piping failure in fluid systems outside containment must meet the requirements of GDC 4 as it relates to accommodating the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. These requirements are imposed to ensure that (1) piping failures in fluid systems outside the containment will not cause the loss of needed functions in safety-related systems, and (2) the plant could be safely shut down in the event of such a failure.

In FSAR Section 3.6.1.3, the applicant stated that the relatively small size of the NPM containment results in congestion that makes the use of traditional piping restraints and the separation of essential components from break locations difficult. The applicant evaluated the effects of postulated pipe breaks in high-energy fluid systems to demonstrate that (1) piping failures in fluid systems outside the containment will not cause the loss of needed functions in safety -related systems and (2) the plant could be safely shut down in the event of such a failure.

In FSAR Section 3.6.2.1, the applicant discussed the criteria used to define break and crack location and configuration. FSAR section 3.6 notes that the details of the analyses on protection against dynamic effects associated with postulated rupture of piping are provided in sections 2.2.5 and appendices of Technical Report TR-121507. Branch Technical Position BTP 3-4 contains the following:

- Branch Technical Position BTP 3-4 provides guidance on the selection of the rupture locations within a piping system. Breaks and leakage cracks may be excluded within the containment penetration area if criteria of BTP 3--4, Section B.1.(ii) are met.

- For high- and moderate-energy systems in areas other than containment penetration breaks and leakage cracks are postulated to occur in areas of high stress or fatigue usage from stress analysis. The HELBs are also postulated at terminal ends. The criteria for the specific locations for the postulated breaks are provided in BTP 3-4 section B.1.(iii)).

BTP 3-4 is an acceptable methodology for demonstrating conformance with GDC 4, and applying these criteria to limit the locations where postulated breaks can occur is acceptable. In SER Sections 3.6.2, the staff evaluates the applicant's implementation of the pipe break location methodology. In SDAA, the applicant does not utilize the leak before break (LBB) methodology at all including for MSS and FWS lines. It is noted in FSAR Section 3.6.2.2.3, Main Steam System (MSS) HELB occurrence is limited to the RXB, because MSS breaks inside the CNV and under the bioshield are eliminated by break exclusion. It was also stated in TR-121507-P (Pipe Rupture Hazards Analysis) Section 2.2.3 that LBB methodology is not applied to the US460 standard design. The design instead eliminates postulated breaks in those portions of the MS and FW line by applying BTP 3-4 B.1(ii) containment penetration area design criteria. It was also stated in TR-121507, Section 2.2.4 that although portions of the MS and FW lines may meet the criteria of BTP 3-4 B.1.(ii), eliminating the need to postulate pipe ruptures, BTP 3-3 requires the consideration of "non-mechanistic breaks" in these lines to ensure environmental effects of ruptures in these lines are considered.

TR-121507-P: The methodology and details of analysis applicable to identification and assessment of pipe rupture hazards are provided in technical report TR-121507-P. It addresses determination of postulated rupture locations, characteristics of ruptures, and assessment of the possible dynamic external effects of ruptures.

In FSAR Section 3.6.2.1.1, the applicant stated that the SGS-MS, SGS-FW, and portions of the DHRS lines inside the CNV are included in the containment penetration area and evaluated using BTP 3-4 B.1.(ii) to eliminate the need to postulate terminal end breaks for these systems inside the CNV.

The staff requested a summary of preliminary stress analysis results for MSS and FWS lines inside CNV due to methodology change from LBB to BTP 3-4. The applicant provided the ASME Section III Class 2 preliminary stress analysis results for MSS and FWS lines (ML24240A141). The staff reviewed the information and finds that the MSS & FWS lines satisfy the ASME stress acceptance criteria as well as BTP 3-4 break exclusion zone criteria limits.

In FSAR Section 3.6.2, the applicant also discussed that high-energy systems are analyzed for postulated circumferential breaks in fluid system piping greater than DN 25 (nominal pipe size (NPS) 1), longitudinal breaks in fluid system piping that is DN 100 (NPS 4) and greater, and leakage cracks in fluid system piping greater than DN 25 (NPS 1). The breaks are analyzed for pipe whip; jet thrust reaction; jet impingement (dynamic effects); flooding; spray wetting; and increased temperature, pressure, and humidity (environmental effects) as shown in FSAR Figure 3.6-1 and Chapter 3, Appendix 3C. The leakage is analyzed for localized flooding and environmental effects.

The staff reviewed the methodology discussed above and finds that excluding postulated breaks in piping below a minimum NPS is consistent with the criteria outlined in BTP 3-4. Therefore, the staff finds the exclusions from the break analysis to be acceptable.

In FSAR Section 3.6, the applicant identified TR-121507-P as providing details of the analysis of the design basis and measures needed to protect essential SSCs inside and outside containment against the effects of postulated pipe ruptures. TR-121507-P states that the NuScale methodology is adequate to identify and assess the pipe rupture hazards and the effects of pipe ruptures and leakage cracks on the ability to achieve safe shutdown and cooldown.

The applicant identified five distinct pipe break zones of plant areas and evaluated the impact of the postulated pipe breaks in each zone.

3.6.1.4.2.1 Pipe Breaks Inside the Containment Vessel

In SER Section 3.6.2, the staff evaluates the applicant’s determination of the protection of essential SSCs inside the containment.

3.6.1.4.2.2 Pipe Breaks Outside the Containment Vessel (under the Bioshield)

The applicant stated that essential components located in the area outside the CNV (under the bioshield) include the following:

- MPS temperature sensor under the bioshield
- CNV
- containment isolation valves (CIVs)
- electrical penetration assemblies
- DHRS actuation valves
- neutron monitoring system (submerged)
- DHRS condenser (submerged)
- ECCS trip/reset valves (submerged)

In FSAR Section 3.6.1.1, the applicant identified the MS, FW, RCS injection, RCS discharge, high point degasification, pressurizer (PZR) spray supply, and DHRS as high-energy lines located in this area. The CRDS, containment flooding and drain system, and the containment evacuation system include moderate-energy lines in this area.

In FSAR Section 3.6.2.1.2, “Pipe Breaks Outside the Containment Vessel (under the bioshield),” the applicant described the applicability of the break exclusion criteria defined in BTP 3-4, Section B, Items A(ii) and A(iii). Crediting these criteria, the applicant determined that no breaks need to be postulated under the bioshield.

In BTP 3-3, Section B, Item 1.a(1), the staff indicates that, even though portions of the MS and FW lines meet the break exclusion requirements of BTP 3-4, Item B.A.(ii), essential equipment must be protected from an assumed non-mechanistic longitudinal break with a cross-sectional area of at least 929 square centimeters (cm^2) (1 ft^2). This failure is postulated to establish the environmental conditions that the essential SSCs need to be protected (or designed) against.

In FSAR Section 3.6.2.1.2.1, “Non-mechanistic Secondary Line Breaks in Containment Penetration Area,” the applicant stated that the 929 cm^2 (1 ft^2) flow area is disproportionately large for the NuScale SMR design. In the case of NuScale, the applicant pointed out that the minimum flow area described in BTP 3-3 exceeds the area of a full--circumferential rupture of the MSS piping (DN 300 (NPS 12)) and the feedwater system (FWS) (DN 100 (NPS 4)). The applicant proposed to postulate a non-mechanistic break of 68.71 cm^2 (10.65 square inches ($in.^2$)) for the MSS and 53.68 cm^2 (8.32 $in.^2$) for the FWS.

The staff evaluated the applicant’s justification for a revised minimum flow area and determined that a minimum flow area of 929 cm^2 (1 ft^2) is disproportional for an SMR. This criterion was based on large light water reactors (LWRs) that have significantly larger piping. Due to the smaller size of an SMR, the staff finds it acceptable to proportionally scale the postulated non-mechanistic break size to 68.71 cm^2 (10.65 $in.^2$) for the MSS and 53.68 cm^2 (8.32 $in.^2$) for the FWS.

Appendix H of TR-121507-P, Revision 0, indicates that the essential components fail to a safe condition upon a loss of power signal and that the area outside of the CNV (under the bioshield) is vented to the RXB to limit the peak pressure and temperature in the event of pipe failure. TR-121507-P Section 3.2.3 indicates that the essential SSCs in this area are qualified for the pressure and temperature conditions resulting from a non-mechanistic break in the area.

The staff reviewed the information in the FSAR and TR-121507-P, Revision 0 and determined that the applicant adequately applied the methodology described in FSAR Section 3.6.1 and identified the essential SSCs that require protection against pipe failure. By designing piping systems in accordance with the recommendations of BTP 3-4, the applicant has reduced the likelihood of high -energy failures and thereby protected the SSCs from a postulated high -energy line failure. By designing the essential SSCs to the anticipated environmental conditions resulting from a non-mechanistic pipe failure, the applicant has protected the SSC functions important to safety against a non-mechanistic pipe failure in the area outside the CNV (under the bioshield). Therefore, the staff finds that the plant design for protection against postulated piping failure in the area outside the CNV (under the bioshield) meets the applicable requirements of GDC 4.

3.6.1.4.2.3 Pipe Breaks in the Reactor Building (Outside the Bioshield)

In FSAR Section 3.6.2.1.3, the applicant stated that there are few essential SSCs that require protection from postulated pipe failures in the RXB (outside the bioshield). The applicant stated that the piping routing in the RXB (outside the bioshield) has not been finalized. The applicant proposed COL Item 3.6-1 to ensure that the COL applicant completes the piping design beyond the NPM bay (the area under the bioshield). This includes final equipment location, pipe routing, support placement and design, piping stress evaluation, pipe break mitigation, and evaluation of subcompartment pressurization and multimodule effects.

The applicant evaluated potential rupture locations to bound the dynamic effects of postulated breaks and then to determine whether protection is required. The approach evaluates the following:

- blast, unconstrained pipe whip, and jet impingement caused by rupture of an MS pipe

- subcompartment pressurization, spray wetting, flooding, and other adverse environmental effects caused by MSS or CVCS breaks that are potentially limiting where they might occur in the building

- multimodule impacts in common pipe galleries

The applicant stated that, depending on the final piping layout, a break in a high-energy MSS or FWS line in the RXB could potentially cause breaks or leakage cracks in smaller diameter or pipe schedule lines of other NPMs and thereby introduce an additional transient in a second NPM. The applicant indicated that the COL applicant's final design may arrange the MSS and FWS pipes or provide pipe whip restraints to prevent a collateral rupture, or a pipe whip impact analysis may be conducted to show that a collateral rupture does not occur. However, for the purpose of the bounding analysis, TR-121507-P assumes that an MSS or FWS break causes a subsequent break in an adjacent module.

The staff reviewed the applicant's approach of performing a bounding pipe rupture hazards analysis (PRHA) and adding COL information items for a COL applicant to finalize the design of the piping systems in the RXB (outside the bioshield). The bounding evaluation addresses the as-designed configuration and the protection of essential SSCs. The COL applicant will be responsible for the evaluation of the protection of essential SSCs, based on the as-built configuration of the plant. Therefore, the staff finds this approach acceptable.

TR-121507-P indicates that the safety-related SSCs in the RXB (outside the bioshield) include the CIV hydraulic actuator assembly skids, which are located in the pipe gallery (two for each NPM), and the structural walls of the RXB itself (including the UHS walls). The report evaluated the consequences of an HELB impact on a CIV hydraulic actuator assembly and determined that failure of the component would cause the CIVs and the DHRS actuation valve to go to their safe positions, which is acceptable. Similarly, it evaluated that a loss of power (caused by pipe failure or any other event) would cause the CIVs to move to a safe position.

The staff reviewed the consequences of a postulated pipe failure impacting the CVCS demineralized water supply isolation valves and determined that a failure of these components would not prevent the plant from achieving safe shutdown or mitigating the consequences of a pipe failure. These components are not considered important to safety for a postulated pipe failure in the RXB (outside the bioshield); therefore, no additional protection is needed.

For the evaluation of dynamic impact, the applicant identified that an MSS break would be a bounding break. TR-121507-P provides the results of the impact of blast effect, pipe whip, and jet impingement and determined that the structural walls of the RXB (including the walls of the UHS) are sufficiently thick to prevent failure of the structural components.

The RXB design accounts for the dynamic loads from pipe failures. The staff gives its structural evaluation of the RXB design in SER Section 3.8.

TR-121507-P evaluated subcompartment pressurization in the RXB (outside the bioshield) and identified the bounding breaks for different subcompartments. In the pipe gallery where the MSS and the FWS are routed, the bounding break is defined as a double-ended MSS line rupture that ruptures an MSS bypass line of another module. For the pipe chase and the heat exchanger room, the applicant identified the CVCS break as the bounding break.

As a means of overpressure protection, the NuScale design features dedicated normally open or blowout paths that do not rely on the RXB ventilation system. These vent paths ensure that no subcompartment exceeds the RXB design pressure.

The staff finds that, because no postulated pipe failure in the RXB (outside the bioshield) exceeds the design pressure of the RXB, the structural components important to safety are adequately protected from postulated pipe failures.

In FSAR Section 3.6.1, the applicant indicated that the flooding evaluation related to postulated pipe failures is discussed in FSAR Section 3.4.1. The staff's evaluation and conclusion of acceptability of the applicant's internal flooding protection are discussed in Section 3.4 of this report.

The staff finds that, by designing the SSCs important to safety in the RXB (outside the bioshield) to the anticipated conditions following a bounding pipe break, the applicant has protected the functions important to safety. Therefore, the staff finds that the plant design for protection against postulated piping failure in the RXB (outside the bioshield) meets the applicable requirements of GDC 4.

3.6.1.4.2.4 Pipe Breaks in the Control Building

In FSAR Section 3.6.2.1.4, the applicant indicated that the CRB has no high -energy lines except piping associated with the high-pressure breathing air storage bottles in the control room habitability system. FSAR Sections 3.4 and 3.11 describe flooding and environmental evaluations, respectively.

In SER Sections 3.4 and 3.11, the staff evaluates internal flooding and environmental conditions, respectively.

3.6.1.4.2.5 Pipe Breaks in the Radioactive Waste Building

The applicant stated that the RWB has no high-energy lines or essential equipment. Therefore, no breaks or leakage cracks are postulated.

3.6.1.4.2.6 Pipe Breaks on Site (Outside the Buildings)

The applicant stated that no essential equipment is located outside the RXB and the CRB. Therefore, no breaks or leakage cracks are postulated.

The staff also reviewed the applicant's methodology described in FSAR Sections 3.6.1 and 3.6.2, for conducting a PRHA. The staff finds that the PRHA described in the SDAA conforms to the guidance in SRP Sections 3.6.1 and 3.6.2 and complies with BTPs 3-3 and 3-4. Based on this, the staff concludes that the applicant can apply this methodology to the protection of SSCs important to safety that are outside containment. Based on the information provided by the applicant, the staff did not identify any issues with the implementation of the relevant approved methodology as described in the SDAA.

3.6.1.5 Combined License Information Items

SER Table 3.6.1-1 lists the COL information item number and description (obtained from FSAR Table 1.8-1) that are related to the PRHA for site-specific high- and moderate-energy piping systems.

Table 3.6.1-1: NuScale COL Information Item for Section 3.6.1

Item No.	Description	FSAR Section
COL Item 3.6-1	An applicant that references the NuScale Power Plant US460 standard design will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the Reactor Building (RXB). This analysis includes an evaluation of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The as-built Pipe Rupture Hazards Analysis (PRHA) will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB, or will perform the PRHA of the high- and moderate-energy lines outside the buildings.	3.6.2.1.3

The staff reviewed the COL information item listed in Table 3.6.1-1 pertaining to PRHA discussed in FSAR Section 3.6.1 and found it to be acceptable based on the staff's technical evaluation presented in SER Section 3.6.1.4.

3.6.1.6 Conclusion

Based on the discussion above, the staff concludes that the NuScale design, as it relates to the protection of safety-related SSCs from the effects of piping failures outside containment, conforms to the guidelines of SRP Section 3.6.1 and, therefore, satisfies the requirements of 10 CFR Part 50, Appendix A, GDC 2 and GDC 4, with respect to accommodating the effects of postulated pipe failure.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2.1 Introduction

GDC 4 requires, in part, that SSCs important to safety be designed to accommodate the effects of postulated accidents, including protection against the dynamic effects of postulated pipe ruptures. Dynamic effects of postulated pipe ruptures include pipe whip and the jet impingement loads on proximate SSCs important to safety. Pipe whip is caused by the reactive thrust loads produced by the fluid jet exiting the break location. The objective of the staff's review described in this section is to verify and ensure that adequate protection has been provided such that the effects of the postulated pipe breaks do not adversely affect the functionality of SSCs relied upon for safe reactor shutdown and that the consequences of the postulated pipe rupture have been mitigated.

3.6.2.2 Summary of Application

FSAR: To address its compliance with the applicable requirements in GDC 4, the applicant described its overall PRHA strategy, which included using design provisions of separation, and break exclusion. FSAR Section 3.6, describes the applicant's approach to mitigate the dynamic effects of postulated HELBs. In addition, the applicant submitted TR-121507-P, Revision 0, to supplement the PRHA-related information contained in FSAR Section 3.6.2. Specifically, TR-121507-P, Revision 0, describes the details of the applicant's PRHA methodologies and the associated results for the NuScale Power Plant US460 standard design. The FSAR Section 3.6.2 information is discussed below.

FSAR Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and its associated FSAR Section 3.6.2.1.1, "Pipe Breaks Inside the Containment Vessel"; Section 3.6.2.1.2, "Pipe Breaks Outside the Containment Vessel (under the bioshield)"; Section 3.6.2.1.3, "Pipe Breaks in the Reactor Building (outside the bioshield)"; Section 3.6.2.1.4, "Pipe Breaks in the Control Building"; Section 3.6.2.1.5, "Pipe Breaks Onsite (Outside the Reactor Building and Control Building)," address the applicant's criteria used for postulating breaks and cracks in the fluid system piping inside and outside containment. In addition, these sections also describe areas that preclude postulated breaks and cracks because the design and examination provisions in BTP 3-4, Section B, Item A(ii), have been applied. In FSAR Section 3.6.1.1, "Identification of High- and Moderate-Energy Piping Systems," and Section 3.6.1.2, "Identification of Safety-Related and Essential Structures, Systems, and Components," identify the respective high- and moderate-energy piping systems and the safety-related and essential SSCs in the NuScale plant. FSAR Section 3.6.2.1, "Criteria Used to Define Break and Crack Location and Configuration," and its associated subsections describe the applicant's criteria used to determine the postulated break and leakage crack locations in the high-energy and moderate-energy piping systems designed using either ASME BPV Code Class 1, 2, or 3 criteria or the criteria in ASME B31.1 "Power Piping." FSAR Section 3.6.2.1.6, "Types of Breaks," and Section 3.6.2.1.7, "High- and Moderate-Energy Leakage Cracks," describe the applicant's criteria used in defining the postulated breaks and crack configurations, including circumferential break, longitudinal break, and leakage crack. The applicant discussed and identified those specific segments of piping and the associated welds where certain design and inspection criteria are used to preclude the need for postulating breaks.

FSAR Section 3.6.2.2, "Effects of High- and Moderate-Energy Line Breaks," and its associated subsections discuss the dynamic effects and/or environmental effects associated with the respective postulated pipe ruptures and their protection methods. Specifically, FSAR Section 3.6.2.2.1, "Blast Effects," and Section 3.6.2.2.3, "Jet Impingement," describe the respective methodology used to evaluate the dynamic effects of blast wave and jet impingement resulting from postulated HELBs for the NuScale plant. In addition, FSAR Section 3.6.2.2.2, "Pipe Whip," describes the methodology for assessing the pipe whip effects. FSAR Section 3.6.2.3, "Protection Methods," describes the methods used in the NuScale design for the protection of postulated pipe ruptures in the respective plant areas and their associated piping systems. FSAR Section 3.6.2.7, "Implementation of Criteria Dealing with Special Features," describes the application of the break exclusion area for the bolted connection of reactor vent valves (RVVs) and reactor recirculation valves (RRVs) to the reactor vessel.

COL Item 3.6-1, listed in FSAR Section 3.6.2.1.3 and FSAR Table 1.8-1, directs an applicant that references the NuScale Power Plant US460 standard design to perform the pipe rupture hazards analysis (PRHA) (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the RXB (outside of the bioshield) and the areas outside the RXB...

ITAAC: As discussed above, SDAA Part 8, “License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC),” Section 2.1.1 and Table 2.1-1, Item 4, describe the pertinent design commitment and associated as-built ITAAC related to the protection of the safety-related SSCs of the NPM and the RXB against postulated pipe rupture effects. These ITAAC are evaluated in Section 14.3 of this SER.

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

Technical Reports: The applicant identified TR-121507-P as the relevant TR. FSAR Section 1.6, “Material Referenced,” Table 1.6-2, “NuScale Referenced Technical Reports,” has specified that TR-121507-P, Revision 0, is incorporated by reference into the NuScale SDAA.

3.6.2.3 Regulatory Basis

The following NRC regulation contains the relevant requirements for this review:

- Compliance with GDC 4 requires nuclear power plant SSCs important to safety to be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs are to be protected against the effects of pipe whip and discharging fluids resulting from pipe breaks.

The guidance in SRP Section 3.6.2, Revision 3, lists the acceptance criteria that are adequate to meet the above requirements and provides review interfaces with other SRP sections, including SRP Section 3.6.1 and SRP Section 3.6.3, “Leak-before-Break Evaluation Procedures.” In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- BTP 3-3, Revision 3, delineates the staff guidance for protection against postulated piping ruptures in fluid systems outside the containment.

- BTP 3-4, Revision3, contains the staff’s guidelines for defining postulated rupture locations in fluid system piping inside and outside the containment.

3.6.2.4 Technical Evaluation

The staff reviewed the applicant’s proposed criteria and methodology used for protection against the effects of postulated pipe ruptures in the NuScale Power Plant US460 standard design for consistency with the NRC’s regulations and guidance specified in SER Section 3.6.2.3. The SER sections below discuss the staff’s review of FSAR Section 3.6, and TR-121507-P, Revision 0.

3.6.2.4.1 Criteria Used to Define Pipe Break and Crack Locations and Configurations

PRHA TR-121507-P, Table 1-2 defines high- and moderate-energy piping systems. FSAR Section 3.6.1.1 and Table 3.6-1, list the high- and moderate-energy fluid systems and gives their locations. The staff's evaluation of the applicant's criteria for defining high- and moderate-energy fluid systems and the associated list in FSAR Table 3.6-1 is within the scope of SRP Section 3.6.1 and is described in SER Section 3.6.1.4.

PRHA TR-121507-P, Table 1-2 also states that fluid piping systems that qualify as "high energy" for only short operational periods are considered moderate-energy systems if the fraction of the time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2 percent of the time during which the system is in operation or if the system experiences high-energy pressure or temperature for less than 1 percent of the plant operation time. The staff found the applicant's criterion described above acceptable because it is consistent with the pertinent staff guidance for considering a high-energy fluid piping system as a moderate-energy system because of its short operational period in high-energy, pressure-temperature conditions, as identified in BTP 3-4, Footnote 5.

FSAR Section 3.6.2, and its associated subsections provide the criteria for defining the location and configuration of postulated breaks and leakage cracks for high-energy and moderate-energy fluid system piping for the NuScale plant design. The plant areas that contain high- and moderate-energy lines or safety-related SSCs are considered in five groups, including inside the CNV, outside the CNV (under the bioshield), in the RXB (outside the bioshield), in the CRB, and on site (outside the RXB and CRB). In the sections below, the staff evaluates the applicant's criteria used to define pipe rupture locations and those design provisions used by the applicant to preclude the need for postulating pipe breaks in certain break exclusion areas.

3.6.2.4.1.1 Postulated Rupture Locations for Fluid System Piping in Break Exclusion Areas

To address its compliance with the applicable requirements in GDC 4, the applicant described the criteria used for determining the postulated rupture locations for the NuScale Power Plant US460 standard design in FSAR Section 3.6. FSAR Sections 3.6.2.1.1, 3.6.2.1.2 and 3.6.2.7 state that breaks are not postulated at certain piping segments, including their associated weld locations where some design and inspection criteria are used to preclude the need for breaks to be postulated. The applicant stated that those specific design and inspection criteria applied to the break exclusion areas are in accordance with the staff's guidelines in BTP 3-4, Section B, Item A(ii), which include design stress limits, criteria for welded attachments, piping welds, and 100-percent volumetric inservice examinations of all pipe welds, in addition to surface inspections as required by ASME BPV Code, Section XI. Those break exclusion areas for the NuScale plant design are described below.

FSAR Section 3.6.2.1.2 states that the CIVs for the RCS injection, RCS discharge, PZR spray, and Reactor Pressure Vessel (RPV) high-point degasification lines are each dual independent valves in a single body that is welded to a containment isolation test fixture (CITF) which is directly connected to safe-end that is welded to the respective nozzle on the CNV head. These lines, except for the normally isolated RPV high-point degasification line, also have a check valve (injection and spray) or excess flow check (discharge) valve outboard of the CIV. The applicant further stated that the FWS CIV is similar, except that a single isolation valve with a check valve is the outboard valve in the single piece body. Each of the MSS lines has a single CIV. Between the CNV nozzle and the valve body is a safe-end tee to which the DHRS steam lines attach. In addition, outboard of the valves in each of these lines is a short piping segment welded to a flange used to connect the refueling pipe spools to the NPM module.

FSAR Section 3.6.2.1.2 also states that the NuScale containment penetration area is defined as the section from the CNV safe-end-to-valve (or tee) weld out to and including the piping weld to the outermost section of the CIV. In addition, the applicant stated that, for welds in the containment penetration areas, 100 percent volumetric examination provisions of BTP 3-4, Section B, Item A(ii), have been applied to preclude the need to postulate breaks.

In FSAR Section 3.6.2.1.2, the applicant stated that break exclusion criteria are applied to the ASME BPV Code Class 1 piping (i.e., the four CVCS RCS lines) from the CNV head to the first isolation valve and to the ASME BPV Code Class 2 MS and FW piping from containment to the first isolation valve, as well as the DHRS piping outside containment. The applicant also stated that the remaining piping under the bioshield, including the refueling pipe spools, is designed to comply with BTP 3-4, Revision 2, Section B, Item A(iii), to preclude breaks at intermediate locations, by limiting stresses calculated by the sum of Equations (9) and (10) in NC/ND-3653 of Section III of the ASME BPV Code to not exceed 0.8 times the sum of the stress limits given in NC/ND-3653. The staff requested preliminary representative stress analysis results for welds in the containment penetration area outside CNV for CVCS injection, CVCS discharge, Pressurizer spray, and RPV High Point Degasification lines. In response (RAI-10177-R1 (Q3.6.2-2); ML24281A020), the applicant provided stress and CUF information. The applicant provided preliminary stress analysis results summary for outboard weld A joining CIV to pipe in the containment penetration area for Pressurizer spray, CVCS Injection, CVCS Discharge, and High Point Degasification lines. The stresses meet ASME Code subsection NC criteria as well as more restrictive BTP 3-4 stress limits. The applicant also provided fatigue CUF results for weld C (weld joining CITF to nozzle safe end) and for weld D (weld joining CNV nozzle to Nozzle safe end). The CUF results for Weld C (weld joining nozzle safe end to CITF) for Pressurizer Spray line are qualitatively justified to be applicable to Weld B (weld joining CIV to CITF) because the distance between welds B and C is small (less than 20 inches). The CUF results for Pressurizer Spray line welds C and D are also less than BTP 3-4 limit of 0.4. The staff reviewed the information and finds that the CUF results that considered environmentally Assisted Fatigue meet the applicable ASME subsection NB criteria as well as the more restrictive BTP 3-4 break exclusion criteria with margin demonstrating robustness of the welds in the containment penetration area. The Pressurizer spray line CUF results for welds are representative for CVCS injection and discharge weld CUF results because mechanical loads are greater for PZR spray line compared to injection and discharge lines. The SSE loads and critical transients, and the outboard safe end geometries are similar. Based on review of the preliminary results provided by NuScale, the staff finds that 4 welds (A, B, C, and D) in the containment penetration area in Pressurizer Spray, CVCS Injection, CVCS Discharge, RPV High Point degasification lines are of robust design. Weld C on Degas line slightly exceeded BTP 3-4 stress guidance but is judged acceptable due to conservatism in BTP 3-4. The final SDAA calculations will be completed by the COL applicant.

The staff's guidance in BTP 3-4 is intended to present a means of compliance with the requirements of GDC 4 for the design of SSCs for nuclear power plants. For the fluid system piping in containment penetration areas (i.e., those portions of piping from the containment wall to and including the inboard or outboard isolation valves), the staff's guidance in BTP 3-4, Section B, Item A(ii), provides certain design and inspection provisions to ensure an extremely low probability of pipe failure in these areas and to allow the exclusion of breaks and cracks from the design basis for those portions of piping. In applying the SRP/BTP guidelines for the break exclusion areas, the stress limit is 80 percent of the applicable ASME Section III stress limit, and the fatigue limit is a cumulative usage factor (CUF) of 0.1 for ASME Class 1 piping. The technical rationale for the reduced stress limit and the CUF of 0.1 is to provide a conservative design to take into account unanticipated conditions such as faulty design, improperly controlled fabrication, installation errors, unexpected modes of operation, uncertainty in vibratory load, and other degradation mechanisms (e.g., corrosive environments, water hammer). With respect to the inspection provision, the pertinent BTP 3-4 staff guideline states that a 100-percent volumetric inservice examination of all pipe welds should be conducted during inspection intervals as defined in ASME BPV Code, Section XI, IWA-2400.

Based on its review of FSAR Section 3.6.2.1.2, the staff determined that the applicant had not applied the break exclusion in the containment penetration areas as envisioned in BTP 3-4, Section B, Item A(ii). Specifically, the applicant applied the break exclusion criteria to the areas beyond the scope of BTP 3-4 for the containment penetration areas. Also, the applicant did not consider the welds between the CNV vessel wall and the CNV safe -end for the CIVs to be within the containment penetration area and did not include these welds within the BTP 3-4 break exclusion boundary. In FSAR Section 3.6.2.1.2, the applicant described the NuScale piping design stress and fatigue limit, as well as the augmented inspection requirement for system piping within the break exclusion area. The staff found that those design and inspection provisions are consistent with the pertinent BTP 3-4 staff guidelines as described above. In FSAR Section 3.6.2.1.2.2, and Appendix A, “Break Exclusion—Compliance with Regulatory Acceptance Criteria,“ to TR-121507-P, Revision 0, the applicant provided additional information to justify the departure from the pertinent staff guidance in BTP 3-4; particularly, how the break exclusion area design provisions in FSAR Section 3.6.2.1.2, are considered and applied to the results of the design of these portions of the system piping, including any associated welds. Specifically, Appendix A to TR-121507-P, Revision 0, provides the detailed geometric configurations of piping within the break exclusion zone, the discussion on the overall length and use of piping bends and welds in the piping evaluation, and the access provision for the applicable weld examinations. The applicant stated that, where piping connects to a CNV safe end-, only the weld between the piping and the safe -end is considered to be within the containment penetration area, whereas the weld between the safe -end and the CNV is part of the vessel and therefore is not considered within the scope of BTP 3-4 for the containment penetration area. The applicant further stated that, although the welds between the safe-ends and the vessel are not considered to be within the containment penetration area, these welds do comply with the requirements for 100-percent volumetric inservice examination, and preliminary stress analysis results show that they meet the BTP 3-4 stress and fatigue limits as applicable, to ensure a low probability of rupture. FSAR Section 3.8.2.7, “Containment Vessel Testing and Inspection Requirements,” and Section 6.2.1.6, ”Testing and Inspection,” detail the inservice examination requirements.

The applicant stated that the ASME BPV Code Class 1, 2, and 3 portions of the piping system, including their associated branch piping in the break exclusion area, were evaluated to meet the relevant break exclusion stress and fatigue criteria as delineated in FSAR Section 3.6.2.1.2. In FSAR Section 3.6.2.1.2.2, Section 3.6.2.1.2.3, “Leakage Cracks,” and in Appendix A to TR-121507-P, Revision 0, the applicant provided additional information to further clarify the application of the staff’s guidance in BTP 3-4, Section B, Items A(ii), A(iii), and A(v), for postulating break and crack locations.

Based on the review of the information provided in FSAR Sections 3.6.2.1.2, 3.6.2.1.2.2, 3.6.2.1.2.3, and6.2.1.6, and Appendix A to TR-121507-P, Revision 0, as described above, the staff finds that the applicant has adequately demonstrated its design provisions and specified a 100-percent volumetric inservice examination for all the pipe welds within the break exclusion areas. Section A.1 #7 of TR-121507 clarifies that IWA-2200 definition of essentially 100% is used. This meets the applicable BTP 3-4 break exclusion criteria in the NRC’s guidelines, and therefore, NuScale’s application of the break exclusion areas is acceptable.

In FSAR Section 3.6.2.7, the applicant stated that each of two RVVs and each of two RRVs in the NuScale design are bolted directly to reactor vessel nozzles. These four bolted-flange connections are also classified as break exclusion areas. The applicant provided its justification to ensure that the bolted connection provides confidence that the probability of gross rupture is extremely low and therefore may be classified as a break exclusion area. Specifically, the applicant stated that the components that comprise these bolted connections (valves, bolts, and nozzles) are classified as ASME BPV Code Class 1 components and are designed, fabricated, constructed, tested, and inspected in accordance with the ASME BPV Code, Section III, Subsection NB. The applicant also stated that the stress design criteria specified in ASME BPV Code, Section III, for the RVV and RRV bolt material provide more margin against yielding than do the rules of ASME BPV Code, Section III, NB-3653, for typical piping system materials and that this meets the intent of the guidance in BTP 3-4 for typical piping systems.

In addition, to support its use of a CUF of 1.0 for those bolted connections, the applicant stated that the fatigue evaluation for these bolts utilizes the fatigue curve from ASME Section III, Division 1, Mandatory Appendix I. Also, as required by Mandatory Appendix XIII-4230 for high-strength bolting, a fatigue strength reduction factor of no less than 4.0 is included in the fatigue evaluation for the NuScale RVV and RRV bolted connection. The applicant described phenomena (e.g., faulty design, improperly controlled fabrication and installation errors, unexpected modes of operation vibration, and other degradation mechanisms) that might adversely affect the fatigue evaluation for piping systems. The applicant explained why the NuScale RVV and RRV bolted connections with threaded inserts are not susceptible to these types of phenomena. The applicant also stated that the RVVs and RRVs are within the scope of the NuScale Comprehensive Vibration Assessment Program (CVAP). The CVAP ensures that the structural components of the NPM exposed to fluid flow are precluded from the detrimental effects of flow-induced vibration (FIV). The applicant discussed NuScale's comprehensive bolting integrity program, the sensitive leakage monitoring system (ML24313A066), the augmented fabrication inspections, and the augmented 100-percent volumetric inservice examination requirements specified for the bolts of these flanged connections. ISI requirements for bolting associated with the RPV are provided in NuScale FSAR Section 5.3.3, "Reactor Vessel Integrity." The applicant stated that the highly sensitive leakage monitoring system (being sensitive to a leak rate as low as 189 milliliters (0.05 gallon) per minute), along with the augmented inservice examinations, provides assurance that potential failure mechanisms are detected before the onset of a catastrophic failure of the bolted connections. The staff finds that the applicant's justification, including the conservatism included in the stress and fatigue design criteria for the bolted connection, the highly sensitive leakage monitoring system, as well as the augmented fabrication and inservice examination requirements specified for the bolts of these flanged connections, provide confidence to ensure that the probability of gross rupture at the bolted connection is extremely low and the bolted connection may be considered as a break exclusion area.

The staff requested stress and fatigue limits for representative bolted connection that includes threaded inserts. NuScale provided the information as an Engineering study (ML24313A066). The staff reviewed the summary of the information of the Engineering Study that provided an evaluation from NuScale DCA calculation for bolted connections using stress classification lines (SCLs). The Engineering Study from a previous Finite Element model in DCA calculation (ECCS Valve Flange Bolting Stress Calculation, EC-A011-7207, Revision 0) extracted the stress and

CUF information for the threaded inserts. The stresses and CUFs for threaded inserts meet ASME limits. The stresses and CUFs for bolting meet ASME as well as BTP 3-4 limits. The staff also reviewed the thread engagement length calculations (RPV Sizing calculation EC-126455 Revision 1) for threads of bolting and threaded inserts. The preliminary analysis results provided for ECCS RRV/RVV bolted connections with threaded inserts are representative for other bolted connections with threaded inserts. The calculation for Final SDAA design will be a COL item to be addressed by the COL applicant.

Based on the review of the preliminary information described in the above-referenced applicable FSAR sections RAI response, the staff determines that the applicant's justification for break exclusion at the ECCS valve bolted connections is acceptable because it meets the intent of the BTP 3-4 staff's guideline for break exclusion areas. In particular, it adequately provides a reasonable assurance to ensure that the probability of gross rupture for the ECCS valves bolted connections is extremely low, and therefore, the bolted connections are considered as break exclusion areas.

3.6.2.4.1.2 Postulated Rupture Locations for Fluid System Piping in Areas Other than Break Exclusion Areas

In FSAR Section 3.6.2.1.1, Section 3.6.2.1.2, Section 3.6.2.1.3, Section 3.6.2.1.4, Section 3.6.2.1.5, Section 3.6.2.1.6, Section 3.6.2.1.7, and Section 3.6.4.1, "Postulation of Pipe Breaks in Areas other than Containment Penetration," describe the applicant's criteria for the postulated pipe break locations in high-energy piping systems in areas other than the break exclusion areas. The respective FSAR criteria for postulating HELBs for ASME BPV Code Class 1, 2, and 3 and ASME B31.1 piping are described below.

For the ASME BPV Code Class 1 high-energy piping systems, breaks are postulated at the terminal ends and intermediate locations where the maximum stress range exceeds 2.4 S_m, as calculated by Equation (10) and either Equation (12) or (13) of ASME BPV Code, Section III, NB-3653, and intermediate locations where the CUF exceeds 0.1, or 0.4 with consideration of environmentally assisted fatigue (EAF). S_m is the allowable design stress intensity value. For ASME BPV Code Class 2 and 3 high-energy piping and ASME B31.1 piping, the intermediate break locations are where stresses calculated by the sum of Equations (9) and (10) in ASME BPV Code, Section III, NC/ND-3653, exceed 0.8 times the sum of the stress limits given in ASME BPV Code, Section III, NC/ND-3653. In addition, FSAR Section 3.6.4.1, states that, where break locations are selected without the benefit of stress calculations, breaks are postulated at the location of potential high stress or fatigue, such as piping welds to each fitting, valve, or welded attachment. The NRC staff finds that those SDAA criteria, as described, are acceptable because they are consistent with the pertinent staff guidelines in BTP 3-4, Section B, Item A(iii), for postulating high-energy piping systems.

TR-121507-P, Table 1-2, states that a terminal end is at the extremity of a piping run that connects to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection on a main piping run is a terminal end for the branch run, except where the branch run is classified as part of a main run in the stress analysis or is shown to have a significant effect on the main run behavior. In piping runs that are maintained pressurized during normal plant conditions for a portion of the run (i.e., up to the first normally closed valve), a terminal end is the piping connection to this closed valve. The NRC finds the NuScale definition of a terminal end acceptable because it conforms to guidance in BTP 3-4, Footnote 3, for postulating pipe ruptures.

FSAR Section 3.6.2.1 states that, for high-energy lines, except for those portions of piping within the break exclusion areas as described in FSAR Sections 3.6.2.1.2 and 3.6.2.7, leakage cracks are postulated at locations that result in the most severe environmental consequences unless otherwise selected by stress analysis. For ASME BPV Code, Section III, Class 1, piping for which a stress analysis has been performed, leakage cracks are postulated at axial locations where the stress range calculated by Equation (10) in ASME BPV Code Section III, NB-3653, exceeds 1.2 S_m. For ASME BPV Code, Section III, Class 2 and 3 piping, or ASME B31.1 piping, leakage cracks are postulated at axial locations where the calculated stress that is equal to the sum of Equations (9) and (10) in ASME BPV Code, Section III, NC/ND-3653, exceeds 0.4 times the sum of the stress limits given in NC/ND-3653. The NRC staff finds those criteria, described in the FSAR, acceptable because they are consistent with the pertinent staff guidelines in BTP 3-4, Section B, Item A(v), for postulating high-energy line leakage crack locations.

FSAR Section 3.6.2.1.7, states that leakage cracks need not be postulated in moderate-energy piping located in an area in which a break in high-energy piping is postulated, provided such leakage cracks would not result in more limiting environmental conditions than those of a high-energy piping break. In other areas of the plant, leakage cracks of moderate-energy lines are assumed at locations that result in the most severe environmental consequences. The NRC staff finds those criteria, as described in the FSAR, acceptable because they are consistent with the pertinent staff guidelines in BTP 3-4, Section B, Items B(iii) and B(iv), for postulating moderate-energy line leakage crack locations.

FSAR Section 3.6.4.1, states that, if a structure is credited with separating a high-energy line from an essential SSC, that separating structure is designed to withstand the consequences of the pipe break in the high-energy line that produces the greatest effect on the structure, irrespective of the fact that the criteria in BTP 3-4, Section B, Items A(iii)(1) through (3), might not lead to postulating a break at this location. The NRC staff finds this criterion in the FSAR acceptable because it is consistent with the pertinent staff guidance in BTP 3-4, Section B, Item A(iii)(4), for a structure that separates a high-energy line from an essential SSC.

3.6.2.4.1.3 Postulated Breaks and Leakage Crack Configurations

FSAR Section 3.6.2.1.6 describes the types of postulated HELBs. It states that at the high-energy pipe break locations, either a circumferential or longitudinal break, or both, are postulated. FSAR Section 3.6.2.1.8, describes the postulated high- and moderate- -energy leakage cracks for the NuScale Power Plant US460 standard design. FSAR Sections 3.6.2.1.6 and 3.6.2.1.7 describe the respective criteria for determining the postulated rupture configurations and the sizes for circumferential breaks, longitudinal breaks, and leakage cracks.

FSAR Section 3.6.2.1.6, states that a circumferential break results in pipe severance and separation amounting to at least a one-diameter, lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by an inelastic limit analysis (i.e., a plastic hinge has not developed in the piping). It further states that pipe movement is initiated in the direction of the jet reaction and whipping occurs in a plane defined by the piping geometry and configuration. In addition, the applicant stated that a longitudinal break results in an axial split without pipe severance. Pipe splits are postulated to be oriented (but not concurrently) at two diametrically opposed circumferential locations such that the jet reactions cause out-of-plane bending of the piping configuration. Alternatively, a single split is assumed at the location of highest tensile stress, as calculated by a detailed stress analysis (e.g., FEA). The applicant also stated that pipe movement occurs in the direction of the jet reaction unless limited by piping restraints, structural members, or piping stiffness, as may be demonstrated by an inelastic limit analysis. The NRC staff finds those criteria, as described in the FSAR, acceptable because they are consistent with the pertinent staff guidelines in BTP 3-4, Section B, Items C(i) and C(ii), for the postulated rupture configurations and the sizes for circumferential breaks and longitudinal breaks.

FSAR Section 3.6.2.1.7 stated that the leakage cracks for high-energy piping should be postulated to be in the circumferential locations that result in the most severe environmental consequences. For the moderate-energy piping, leakage cracks should be postulated at axial and circumferential locations that result in the most severe environmental consequences. The NRC staff finds that the information described in FSAR Section 3.6.2.1.7, is consistent with the pertinent staff guidance in BTP 3-4, Section B, Item C(iii), for postulating leakage crack locations of high- and moderate-energy piping and, therefore, is acceptable.

FSAR Section 3.6.2.1.7 also describes that fluid area from a leakage crack should be based on one-half pipe wall thickness in width. The flow from a leakage crack should be assumed to result in an environment that wets the unprotected components within the compartment with consequential flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period necessary to effect corrective actions. The NRC staff finds those criteria, as described in the FSAR, acceptable because they are consistent with the pertinent staff guidelines in BTP 3-4, Section B, Items C(iii)(3) and C(iii)(4), for the postulated rupture configurations and the sizes for leakage cracks.

3.6.2.4.1.4 Analysis Methods Used to Evaluate the Dynamic Effects of Postulated High-Energy Pipe Breaks

FSAR Section 3.6.2.2, and its associated subsections, and Section 3.9, “Mechanical Systems and Components,” as well as Appendix B, “Dynamic Amplification and Potential for Resonance,” Appendix C, “ Pipe Whip,” Appendix E, “Jet Impingement,” and Appendix F, “Blast Effects,” to TR-121507-P, Revision 0 describe the applicant’s methodologies used to evaluate the dynamic effects of a blast wave, jet impingement, and pipe whipping resulting from postulated HELBs for the NuScale plant. The applicant’s respective dynamic analysis methodologies are described below.

3.6.2.4.1.4.1 Blast Effects

SRP Section 3.6.2, Appendix A, “Potential Non-conservatism of ANSI/ANS 58.2 Standard’s Jet Modeling,” identifies a concern about the potential blast wave effects resulting from postulated HELB in nuclear power plants. It states that the first significant fluid load on surrounding SSCs caused by an HELB would be induced by a blast wave. Although a spherically expanding blast wave is reasonably approximated to be a short-duration transient and analyzed independently of any subsequent jet formation, reflections and amplifications in enclosed areas of the plant may need to be evaluated.

The applicant addressed the blast wave effects in FSAR Section 3.6.2.2.1, and TR-121507-P, Revision 0, Appendix F. FSAR Section 3.6.2.2.1 states that the key factors for the potential for a blast wave to occur include the timing of the opening of the break, the initial intact system thermodynamic conditions, and the surrounding environment. It also states that although pipe rupture times of less than a millisecond are unlikely, the break opening time is assumed to be instantaneous to maximize the blast formation for the evaluation of blast wave effects for the NuScale Power Plant US460 standard design.

FSAR Section 3.6.2.2.1 states that the formation and effects of a blast wave resulting from a postulated high-energy pipe break is evaluated using three -dimensional (3-D) computational fluid dynamics (CFD) modeling that reflects the postulated break characteristics and NuScale plant geometry. The analysis is performed using the ANSYS CFX Version 19.2code. The applicant demonstrated the acceptability of using the ANSYS CFX Version 19.2code for assessing the blast effects for the NuScale plant by performing verification and validation (V&V) using eight test problems that exercised different capabilities of the code. TR-121507-P, Revision 0, Appendix F, includes the details of the CFD modeling and the results of the V&V. In Appendix F, the applicant described how the NuScale CFD modeling was benchmarked against the eight test problems to verify its suitability and how the CFD analysis properly considered the potential impact of the mesh size and time step on convergence. The applicant concluded that the agreement between the ANSYS CFX Version 19.2code results and the reference values discussed in the respective literatures of the eight test problems provides validation and confidence that the CFD modeling adequately modeled the blast wave phenomena following a postulated HELB in the NuScale Power Plant US460 standard design.

FSAR Section 3.6.2.2.1 also discusses the key observations from the applicant’s blast wave modeling. In particular, the applicant stated that a blast wave is weakly formed if the surrounding environment is at low pressure (less than 7 kPa (absolute) (1 pound-force per square inch (psia))), which is the case inside the NuScale CNV. Buildup of pressure as blowdown progresses is not relevant because the blast wave is a prompt and short-lived phenomenon. The pressure load applied by a blast wave is of short duration (i.e., an impulse load) and does not apply uniformly across large SSCs at a given instant. Therefore, assuming the peak blast pressure is applied across the entire projected area of a component is inappropriate. The CFD analysis explicitly accounts for the time varying pressure of the rapidly propagating blast wave. The applicant also stated that angled or curved surfaces are loaded differently than a flat surface perpendicular to a line between the blast origin and surface. In addition, use of the NuScale plant specific geometry is necessary because pressures applied to surfaces by reflection can exceed the incoming wave and can reinforce the blast wave pressure load. Therefore, the CFD analysis includes the interaction of incident and reflected waves with each other and nearby surfaces, including how the shape and orientation of surfaces affect reflection. The applicant stated that the NuScale high energy, steam-filled lines are relatively small, which limits the severity of the blast pressure. The energy available to form the blast is less than 1/11th of that for a typical large LWR.

The applicant stated that a small target has a lower peak pressure because of “clearing,” which is a phenomenon in which some of the blast overpressure is relieved by bleeding off around the edge of the target. Because of both pressure-relieving clearing and the short load duration as a blast wave moves over them, small structures are not exposed to significant loading. The only SSCs in the NuScale CNV or RXB that are large are structures such as the CNV, RPV, and RXB walls and floors. The 3-D CFD analyses of blast wave formation for several locations and directions of the CVCS breaks in the CNV and the MSS breaks in the RXB pipe gallery were performed using modeling assumptions that bound the pressurization effects that may occur for HELBs in the NuScale Power Plant US460 standard design. Blast wave force time histories were calculated for nearby SSCs, and the results show that the effects of HELB -induced blast waves in the NuScale plant are low and bounded by the jet thrust forces that subsequently develop.

Based on a comparison of the applicant’s methodology to the pertinent staff guidance in SRP Section 3.6.2, Appendix A, the staff determined that the applicant’s methodology for determining the blast wave effects on the impacted SSCs is technically justified and therefore acceptable.

Specifically, the applicant provided sufficient information to demonstrate the validity and the applicability of the test data and methodology contained in the referenced open literature to the NuScale HELB fluid conditions and geometric configurations. In addition, the applicant's CFD analysis includes numerous assumptions that are technically justified and conservative. The CFD analysis was benchmarked against several experiments and analyses of similar conditions studied in the literature to verify its suitability. The applicant provided sufficient information to demonstrate that appropriate mesh size and time step have been properly considered to ensure the convergence in its CFD analysis. Accordingly, the staff finds the applicant's methodology and approach to evaluate the blast wave effects acceptable because the applicant has adequately addressed the staff's concern about the blast wave effects, as identified in SRP Section 3.6.2, Appendix A.

3.6.2.4.1.4.2 Jet Impingement Loads

FSAR Section 3.6.2.2.3 and TR-121507-P, Revision 0, Appendix E, address the methodologies used to assess the jet impingement loads in the NuScale plant. The applicant's assessment considers jet impingement effects in the NuScale plant for three possible HELB fluid conditions, including an HELB yielding a single-phase steam jet, an HELB yielding a two-phase steam/water jet, and an HELB yielding a single-phase liquid jet. The jet impingement effects for these three different fluid conditions are addressed through different methodologies that consider jet range, shape, and direction, such as the zone of influence (ZOI), the jet blowdown pressure distribution within the jet plume, and the jet impingement force with an applicable thrust coefficient.

FSAR Section 3.6.2.2.3 states that the single-phase liquid jets are assumed to not expand and to not droop with distance (i.e., the cross-sections of their ZOIs are the same as those of the postulated breaks themselves, and the penetration distance for a liquid jet is assumed to extend infinitely until it impinges on a target). In determining the liquid jet thrust force, a thrust coefficient of 2.0 is applied. The staff finds the applicant's criteria for evaluating the liquid jet pressure acceptable because they are consistent with the pertinent staff guidance in SRP Section 3.6.2 for evaluating the liquid jet load.

FSAR Section 3.6.2.2.3 states that two-phase jets are assessed using the methodology of NUREG/CR-2913, "Two-Phase Jet Loads," issued January 1983, for determining the jet impingement load on the potential target. In addition, the applicant stated that the initially low air density of the CNV removes most of the resistance to jet expansion, which results in a wider jet expansion. The applicant also stated that, although a graph in NUREG/CR-2913 can be used to determine the ZOI of the two-phase jet, the ZOI in the NuScale CNV is conservatively assumed to be in the forward-facing hemisphere such that any essential SSC is within the ZOI if it is located within the forward -facing hemisphere. In TR-121507-P, Revision 1, Appendix E, the applicant included a sample calculation to show how it used the NUREG/CR-2913 methodology to assess the two-phase jet impingement pressure resulting from a CVCS break. The staff finds the applicant's methodology as described above acceptable because the NUREG/CR-2913 methodology and the conservative assumption of a ZOI in the forward-facing hemisphere are appropriate for use in analyzing the two-phase jets for the NuScale plant design. The staff also noted that it had accepted the NUREG/CR-2913 methodology in previous DCAs for the analysis of two-phase jets.

FSAR Section 3.6.2.2.3, and TR-121507-P, Revision 0, Appendix E, describe the applicant's methodology for assessing the steam jet effects for the NuScale plant. In determining the jet thrust force, a thrust coefficient of 1.26 is applied. In TR-121507-P, Revision 0, Appendix E, the applicant also stated that, for breaks inside the CNV, wider jet spreading is expected to occur because the initially low air density of a CNV pressure below 7 kPa (absolute) (1 psia) removes most of the resistance to jet expansion. The applicant further stated that, as seen in the pressure contour plots included in TR-121507-P, Revision 0, Appendix F, a jet expansion **{{ }}**. The wider jet expands the ZOI but substantially reduces the pressure and the jet penetration length because the mass and energy of the jet are more widely dispersed. The applicant stated that, although a spreading half-angle of more than 60 degrees should be a reasonable approximation of an actual jet in the CNV, for assessing the steam jet pressure effects for the NuScale plant, the steam jet is conservatively assumed to expand at a 30 -degree half-angle to a downstream distance of five pipe diameters and then at 10 degrees beyond that. TR-121507-P, Revision 0, Table E-2, "CVCS Steam Jet Impingement Pressure Versus Distance," compares the CVCS steam jet impingement pressure to the jet penetration distance **{{ }}** The applicant stated that the ZOI for the steam jet in the NuScale CNV is conservatively assumed to be in the forward-facing hemisphere.

FSAR Section 3.6.2.2.3 also states that the piping arrangement in the RXB has not yet been finalized. To verify suitability of the design of the RXB, bounding HELB scenarios for MSS breaks are postulated. In addition, to ensure that the final HELB analysis results are bounded, the applicant conservatively assumed the jet impingement pressure at the potential target to be the same as that at the break exit (i.e., no reduction for spreading with distance).

Based on its review of the information described above, the staff determined that the applicant's methodology for assessing the steam jet effects is technically justified and acceptable because (1) in the NuScale CNV, the applicant conservatively assumed a steam jet spreading half-angle of 30 degrees that would result in a higher jet pressure on a potential target than the one resulting from the expected minimum 60-degree half-angle jet expansion and conservatively assumed a ZOI to be in the forward-facing hemisphere, (2) in the RXB, the applicant conservatively assumed the steam jet impingement pressure at the potential target to be the same as that at the break exit, and (3) the applicant has adequately addressed the staff's concern about the jet impingement effects related to jet plume expansion, jet pressure distribution, and the potential ZOI as identified in SRP Section 3.6.2, Appendix A.

FSAR Section 3.6.2.2.3, and TR-121507-P, Revision 0, Appendix B, address an issue identified in SRP Section 3.6.2, Appendix A, related to the potential for a jet load amplification associated with the formation of unsteadiness in free jets, especially supersonic jets, which propagate in the shear layer to induce time-varying oscillatory loads on obstacles in the flow path. The concern is that synchronization of transient waves with the shear layer vortices emanating from the jet break can lead to significant amplification of the jet pressures and forces (a form of resonance). If the dynamic response of the neighboring structure also synchronizes with the jet loading time scales, further amplification of the loading can occur as a result of the formation of a feedback loop. When the impingement surface is within 10 diameters of the jet opening and when resonance within the jet occurs, significant amplification of impingement loads might result.

In its evaluation of the potential occurrence of dynamic amplification and resonance in HELB jets for the NuScale plant design, the applicant stated that the dynamic amplification and resonance phenomenon occurs in studies in which a stable, axisymmetric jet impinged at a fixed distance perpendicular to a large, flat surface. The applicant also stated that a potential HELB jet impingement has fundamental differences from those that occur in a jet with dry, non-condensable gas issuing from a smooth, fixed nozzle. These physical differences involve instability of the discharge, irregular discharge geometry, phase changes that suppress pressure changes, misalignment of jet and impingement target surface preventing the establishment of a feedback loop, and lack of an appropriately flat surface within a sufficiently close distance. The applicant stated that if one of these criteria is not met, a resonance is implausible.

The applicant discussed multiple physical characteristics (see FSAR Section 3.6.2.2.4) of NuScale HELBs that prevent the occurrence of a resonance. For example, the break exit is distorted because of tearing as the break opens, which eliminates axi-symmetry, and self-damping effects of a two-phase jet (which is not relevant to single-phase jets where resonance has been seen). In addition, the absence of a large, flat impingement surface sufficiently close and perpendicular to the jet axis and the variation in the jet discharge angle and distance prevent the establishment of a stable feedback loop. The irregularities in the contours of the broken pipe end and the impingement target distort the outgoing jet and spread out reflected acoustic energy. Accordingly, the applicant concluded that potential dynamic amplification and resonance-induced pressure loading is not a concern for jet impingement on the NuScale plant SSCs.

Based on its review of the above information, the staff determined that the applicant's approach and conclusion are technically justified and therefore are acceptable. Specifically, the applicant has demonstrated that the conditions needed to establish resonance and dynamic amplification, as identified in the open literature, will not be present for HELBs in the NuScale Power Plant US460 standard design, and the potential dynamic amplification and resonance -induced pressure loading is not a concern for jet impingement on the Power Plant US460 standard design SSCs. Therefore, the staff finds the applicant's evaluation and approach, as described above, acceptable because the applicant has demonstrated reasonable assurance that this phenomenon will not exist for HELBs in the Power Plant US460 standard design and because it has adequately addressed the staff's concern about the potential dynamic jet amplification and resonance jet impingement effects identified in SRP Section 3.6.2, Appendix A.

3.6.2.4.1.4.3 Pipe Whip Effects

FSAR Section 3.6.2.2.2, and TR-121507-P, Revision 0, Appendix C, describe the methodology used for assessing the pipe whip effects on the nearby SSCs. The applicant's methodology determined whether a pipe has sufficient energy to whip, whether a whipping pipe can potentially impact a target, and whether the target is sufficiently robust to withstand the impact, and evaluated the consequences of an impact if the previous steps do not obviate the possibility of damage. The applicant also described the considerations applied to the evaluation of pipe whip effects for the NuScale plant design. For example, for piping that meets the criteria of break exclusion, pipe whip is not considered because the dynamic effects of ruptures are excluded. In areas where pipe ruptures are postulated to occur, the length of the whipping pipe is determined from the break location to the nearest restraint that limits the range or direction of the pipe whip. The jet thrust necessary to cause pipe whip is also determined. The calculation of thrust and jet impingement forces considers no line restrictions (e.g., a flow limiter) between the pressure source and break location, but it does consider the absence of energy reservoirs, as applicable (e.g., the high-point vent pipe in the CNV is normally isolated). If the jet thrust is insufficient to yield the pipe, pipe whip at that break location is eliminated from further consideration, although jet impingement from the postulated break is still relevant. In addition, the pipe whip could result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the vector thrust of the break force. A maximum rotation of 90 degrees is assumed about a hinge. Pipe whip occurs in the plane defined by the piping geometry and configuration and initiates pipe movement in the direction of the jet reaction. TR-121507-P, Appendix C, provides the details of the methodology described above and a sample calculation to show how the methodology is applied to the applicant's evaluation of pipe whip effects. The staff's review of the information described above determined that the applicant's methodologies for assessing the pipe whip effects, as described in FSAR Section 3.6.2.2.2, and TR-121507-P, Revision 0, Appendix C, are consistent with the pertinent staff guidance for assessing pipe whip effects in BTP 3-4, Section B, Items C(i) and C(ii), and therefore are acceptable.

3.6.2.4.1.4.4 Pipe Whip Restraints Design

As described in SRP Section 3.6.2, one of the protection methods to mitigate the pipe whip effect is to install a pipe whip restraint. FSAR Section 3.6.2.3.1.1, describes the design criteria for the pipe whip restraints for the NuScale Power Plant US460 standard design. The NuScale pipe whip design is based on energy absorption principles and considers the elastic-plastic, strain-hardening behavior of the materials used. Nonenergy-absorbing portions of the pipe whip restraints are designed to the requirements of ANSI/AISC N690 code or ASME Subsection NF. Except in cases for which calculations are performed to determine whether a plastic hinge is formed, the energy absorbed by the ruptured pipe is conservatively assumed to be zero (i.e., the thrust force developed goes directly into moving the broken pipe and is not reduced by the force required to bend the pipe). The analysis of the NuScale pipe whip restraints design is either a dynamic or static analysis that considers a dynamic factor of 2.0. In addition, an amplification factor of 1.1 is considered to account for the potential occurrence of pipe rebound upon impact on the restraint. The allowable strain in a pipe whip restraint is dependent on the type of restraint. If a crushable material such as honeycomb is used, the allowable energy absorption of the material is 80 percent of its rated energy dissipating capacity as determined by dynamic testing performed at loading rates within ±50 percent of the specified design loading rate. The staff's review of the information described above determined that NuScale's pipe whip restraint design criteria, as provided in FSAR Section 3.6.2.3.1.1, are consistent with the pertinent staff guidance for the design of pipe whip restraints in SRP Section 3.6.2, Section III, Items (2)(A) and (2)(B), and therefore are acceptable. To ensure the applicant's compliance with the applicable requirements in GDC 4 for protecting SSCs important to safety against the dynamic effects of postulated pipe ruptures, SRP Section 3.6.2, Section III, Item 2.A, provides guidance for evaluating the dynamic response of the fluid system piping when pipe ruptures are postulated. Specifically, SRP Section 3.6.2, Section III, Item 2.A, states that an analysis of the dynamic response of the pipe run or branch should be performed for each longitudinal and circumferential postulated piping break. The evaluation of postulated breaks should use the loading condition (e.g., internal pressure, temperature) of a pipe run or branch before the postulated rupture occurs. For piping that is pressurized during operation at-power, the initial condition should be greater than the contained energy at hot standby or at 102-percent power. In TR-121507-P, Table 3-3, "High Energy System Piping Characteristics for piping in the Containment Vessel, NuScale Power Module Bay, and reactor building," the applicant stated that the initial conditions assumed for dynamic response to pipe breaks are selected to bound system conditions for hot standby (for which NuScale equivalent is referred to as hot shutdown) or at 102-percent power. The staff finds that the initial conditions assumed for dynamic response to pipe breaks are consistent with the pertinent staff guidance in SRP Section 3.6.2, Section III, Item 2.A, and therefore are acceptable.

3.6.2.4.2 Pipe Rupture Hazards Analysis Report

To support the staff's review and the safety determination of the acceptability of FSAR Section 3.6.2, the applicant submitted TR-121507-P, which describes the details of the applicant's methodologies and the results of the PRHA to demonstrate its compliance with the applicable requirements in GDC 4. Specifically, TR-121507-P addresses the applicant's criteria used to identify the postulated rupture locations; the characteristics of postulated pipe ruptures, including break and crack types and sizes; the methodologies to assess the potential effects of high-energy and moderate-energy line breaks and cracks; and the design criteria and requirements to demonstrate that SSCs important to safety are designed to accommodate and protect against the effects of postulated pipe failures. The staff evaluates the above applicant PRHA methodologies and the design criteria in SER Sections 3.6.2.4.1 and 3.6.2.4.2, respectively.

FSAR Section3.6.1.3, and TR-121507-P, Revision 0, Section 2.1, "NuScale Design Features Relevant to Pipe Rupture Hazards Analysis," describe the NuScale design features relevant to the PRHA that are different from those of the design in the existing fleet of large LWRs. In particular, the applicant stated that the NuScale design is an integral, multiunit station. Up to 6 NPMs are operating at the same time and in proximity to one another; therefore, the potential for a postulated rupture in a system of one module to affect other modules must be considered. In addition, the NPM containment is operated at a vacuum. In addition, MSS and FWS piping inside the CNV meets the BTP 3-4 criteria. The size of high-energy piping is small compared to that of reactors of current design. HELBs inside the CNV are limited to a DN 50 (NPS 2) pipe size. The small containment results in congestion that makes the addition of traditional pipe whip restraints and the physical separation of essential components from break locations difficult; however, whipping pipes, in turn, have a limited range of motion before encountering an obstacle.

The applicant stated that, in the NuScale design, because of differences in both the potential piping hazard and the surrounding environment, postulated HELBs are evaluated in three discrete areas of the plant: (1) inside the containment of the NPM, (2) in the pool bay area above each NPM and under the bioshield, and (3) in the RXB. In FSAR Figure 3.6-1, "Flowchart of Methodology for Evaluation of Line Breaks," and Figure 3-1, "Flowchart of Methodology for Evaluation of Line Breaks," of TR-121507-P, Revision 0, the applicant described the process for identifying postulated rupture locations and vulnerable essential and safety-related targets by assessing the relevance and consequences of possible HELB effects (i.e., blast wave, pipe whip, and jet impingement) and the requirement for applicable load combinations. The applicant also stated that the applicable load combinations are in accordance with FSAR Section 3.9, for components and supports and with FSAR Section 3.12, "ASME BPV Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports," for piping. The applicant stated that the effects of HELB induced blast waves in the plant are considered negligible, and no damage to surrounding SSC occurs because these loads are small, and their durations are brief. In Section 5.0, "Results and conclusions," of TR-121507-P, Revision 0, the applicant summarized the evaluations and results of the NuScale PRHA analysis. The applicant stated that the application of the criteria for break exclusion leaves few locations in the CNV and none in the NPM bay requiring an evaluation of the effects of blast waves, pipe whip, jet impingement, subcompartment pressurization, and flooding. The applicant also stated that protection of the potential target SSCs is demonstrated through separation and by the robustness and qualification of the essential SSCs. The applicant stated that the evaluation of bounding high-energy and moderate-energy pipe ruptures demonstrates that the essential components in the RXB and the RXB structure are capable of withstanding the effects of postulated pipe ruptures. Based on its review of the information described above, the staff finds the applicant's PRHA in TR-121507-P, Revision 0, acceptable because the applicant has provided sufficient information to demonstrate that the PRHA methodology and criteria are in conformance with the pertinent staff guidelines in SRP Section 3.6.2 and BTP 3-4. In addition, the results presented in TR-121507-P, Revision 0, demonstrate that the NuScale Power Plant US460 standard design complies with the applicable requirements in GDC 4, such that SSCs important to safety are designed to accommodate and protect against the effects of postulated pipe failures.

Some HELB-related topics, including HELB dynamic effects (i.e., pipe whip effects) on structures (e.g., pipe whip effects on concrete structure), containment pressurization, flooding effects, and environmental qualification (EQ) of mechanical and electrical equipment, are not within the review scope of SRP Section 3.6.2 and are not addressed in this SER section. The staff evaluates these topics in SER Sections 3.8.4, 6.2.1, 3.4, and 3.11, respectively. In TR-121507-P, Revision 0, Section 2.2.4 and Appendix H, the applicant addressed the issue related to BTP 3-3, Section B, Item 1.a(1), for a postulated non-mechanistic break for MSS and FWS piping in the containment penetration area, as well as the issue related to pressurization outside containment. The staff's review of that topic is within the scope of BTP 3-3 and is addressed in SER Section 3.6.1.

The applicant's PRHA in TR-121507-P, Revision 0, addresses the effects of high-energy and moderate-energy pipe breaks and cracks in the NuScale NPM and RXB. As stated in FSAR Section 3.6.2.1.3, the final routing of piping, including placement of restraints beyond the NPM pool bay, is within the COL applicant's scope, as clarified by COL Item 3.6-1. SER Section 3.6.2.5 describes the staff's evaluation of this COL information item.

3.6.2.5 Combined License Information Items

SER Table 3.6.2-1 lists the COL information item number and description from FSAR Table 1.8-1 related to the PRHAs for their associated plant areas.

Table 3.6.2-1: NuScale COL Information Item for Section 3.6-2

Item No.	Description	FSAR Section
COL Item 3.6-1	An applicant that references the NuScale Power Plant US460 standard design will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the Reactor Building (RXB). This analysis includes an evaluation of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The as-built Pipe Rupture Hazards Analysis (PRHA) will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB, or will perform the PRHA of the high- and moderate-energy lines outside the buildings.	3.6.2.1.3

FSAR Section 3.6.2.1.3 describes the details of the COL information items. The staff finds that this COL information item adequately describes the respective actions for COL applicants to complete with regard to PRHAs for their associated plant areas. Specifically, the staff finds them acceptable because they direct a COL applicant that references the NuScale Power Plant US460 standard design to complete the routing of the applicable piping systems, to update the associated PRHAs, and to evaluate multimodule impacts in common pipe galleries and subcompartment pressurization.

3.6.2.6 Conclusion

The applicant appropriately identified or postulated pipe rupture locations and designed piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingement to provide adequate protection for the SSCs that are important to safety.

The applicant's proposed piping and restraint arrangement and applicable design considerations for high- and moderate-energy fluid systems inside and outside of containment, including the RCPB, provide adequate assurance that SSCs important to safety that are in close proximity to the postulated pipe rupture will be appropriately protected. The proposed design appropriately mitigates the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside of containment.

The staff concludes that the applicant postulated pipe ruptures appropriately, designed SSCs that are important to safety to accommodate and protect against the associated dynamic effects, and therefore met the relevant requirements of GDC 4.

3.6.3 Leak-before-Break Evaluation Procedures

3.6.3.1 Introduction

FSAR Section 3.6.3 describes the applicant's LBB evaluation procedures. As stated in GDC 4, dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of a fluid system piping rupture is extremely low under conditions consistent with the design for the piping.

3.6.3.2 Summary of Application

The applicant stated in FSAR Section 3.6.3 that for the NuScale US460 Plant, LBB methodology is not used.

ITAAC: There are no ITAAC related to LBB.

Technical Specifications: There are no GTS directly related to this area of review.

Technical Reports: There are no TRs related to LBB materials and design as LBB is not used in the SDAA.

3.6.3.3 Regulatory Basis

The following NRC regulation contains the relevant requirements for this review:

- GDC 4, as it relates to the exclusion of dynamic effects of the pipe ruptures, which are postulated in SRP Section 3.6.2.

The design basis for the piping refers to those conditions specified in the safety analysis report, as amended, and may include regulations in 10 CFR Part 50, applicable sections of the SRP, RGs, and industry standards such as the ASME BPV Code.

3.6.3.4 Technical Evaluation

The applicant did not use LBB in SDAA.

3.6.3.5 Combined License Information Items

There are no combined license information items for this area of review.

3.6.3.6 Conclusion

Based on its review, the staff finds that LBB methodology is not applied to the US460 standard design.

3.7 Seismic Design

3.7.1 Seismic Design Parameters

3.7.1.1 Introduction

Final Safety Analysis Report (FSAR) Section 3.7.1, “Seismic Design Parameters,” describes the design parameters used as input to the seismic analysis and design of the Seismic Category I structures in the NuScale US460 standard design. This section of the application discusses the following information on the seismic design parameters for the NuScale standard design:

- Design earthquake ground motion
- Percentage of critical damping values
- Supporting media for Seismic Category I structures

3.7.1.2 Summary of Application

FSAR: FSAR Section 3.7.1, describes the seismic design parameters, including the design ground motion, percentage of critical damping values, and supporting media, used as input to the seismic analysis of the NuScale Seismic Category I structures.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC associated with FSAR Section 3.7.1.

Technical Specifications: There are no Generic Technical Specification (GTS) for this area of review.

Technical Reports: NuScale Licensing Topical Report, TR-0920-71621-P-A, Revision 1, "Building Design and Analysis Methodology for Safety-Related Structures," (ML22062B056).

3.7.1.3 Regulatory Basis

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

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, GDC 2, as it requires that the structure, system, and component (SSCs) important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions and that the design bases for these SSCs reflect appropriate consideration of the most severe earthquakes that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 50, Appendix S, as it requires that, for the safe shutdown earthquake (SSE) ground motion, certain SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion associated with the SSE through design, testing, or qualification methods. The evaluation must account for soil-structure interaction (SSI) effects and the expected duration of the vibratory motion. If the operating basis earthquake (OBE) is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. The horizontal component of the SSE ground motion in the free field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.

- 10 CFR 52.137(a)(1), as it requires that an FSAR must include the site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters.

- 10 CFR 52.137(a)(20), as it requires that an FSAR must include the information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.

The guidance in Design Specific Review Standard (DSRS) Section 3.7.1, Revision 0, “Seismic Design Parameters” (ML15355A384) lists the acceptance criteria adequate to meet the above requirements and review interfaces with other DSRS sections. In addition, the following guidance provides acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.60, Revision 2, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” issued July 2014, for determining the acceptability of design response spectra for input into the seismic analysis of nuclear power plants (ML13210A432)

- RG 1.61, Revision 2, “Damping Values for Seismic Design of Nuclear Power Plants,” issued March 2023, for determining the acceptability of damping values used in the dynamic seismic analyses of Seismic Category I SSCs (ML070260029)

- DC/COL-ISG-01, “Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion in Design Certification and Combined License Applications,” dated May 19, 2008 (ML081400293)

- DC/COL-ISG-017, “Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analysis,” dated March 24, 2010 (ML100570203)

- NUREG/CR-5347, “Recommendations for Resolution of Public Comments on USI A-40, ‘Seismic Design Criteria,’” issued June 1989 (ML110030124)

- NUREG/CR-6728, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines,” issued October 2001 (ML013100232)

3.7.1.4 Technical Evaluation

The staff reviewed FSAR Section 3.7.1, against the agency’s regulatory guidance to ensure that the FSAR represents the complete scope of information related to this review topic. The staff evaluated FSAR Section 3.7.1 with regard to seismic design parameters, following the guidance in DSRS 3.7.1. The reviewed information includes (1) the design ground motions, (2) percentage of critical damping values, and (3) supporting media for Seismic Category I structures.

The evaluation of the design ground motions covers the certified seismic design response spectra (CSDRS) and the corresponding CSDRS-compatible design ground motion time histories, with seed time histories selected from actual earthquake recordings at different sites including Yermo, Capitola, Chi-Chi, Izmit, and El Centro. The design ground motions also include the certified seismic design response spectra-high frequency (CSDRS-HF) and the corresponding CSDRS-HF compatible ground motion time history, with a seed time history selected from an earthquake recording at the Lucerne site. The evaluation of the percentage of critical damping covers the system and component damping, structural damping, and soil damping. The evaluation of the supporting media for Seismic Category I structures covers the generic soil profiles and their corresponding strain compatible soil properties.

The seismic analysis of the NuScale Seismic Category I SSCs uses these seismic design parameters to develop the seismic demands used for the NuScale standard design. Meeting the DSRS Section 3.7.1 acceptance criteria ensures that the seismic design parameters in the seismic analysis of the NuScale Seismic Category I SSCs are adequately defined to form a conservative basis for the design of such SSCs to withstand the design basis seismic loadings.

This SER section presents the results of the staff’s technical evaluation of FSAR Section 3.7.1. SER Section 3.7.2 presents the staff’s evaluation of the seismic system analysis of the NuScale Seismic Category I structures and major plant systems. SER Section 3.7.3 presents the staff’s evaluation of the seismic subsystem analysis for the NuScale standard design.

3.7.1.4.1 Design Ground Motion

FSAR Section 3.7.1, describes the design ground motions developed for use as input in the seismic analysis of the NuScale standard design. The applicant stated that its Seismic Category I and II structures are designed for the CSDRS and CSDRS-HF, which represent the maximum vibratory ground motion at the generic plant site. The OBE for the NuScale Power Plant is proposed as one-third of the SSE. The applicant explained that, in accordance with 10 CFR Part 50, Appendix S, an explicit response analysis or design of the Seismic Category I SSCs for the OBE is not necessary because the OBE is set to one-third of the SSE. The staff concludes that, with the specification of the OBE as one-third of the SSE, exclusion of the seismic analysis and design for the OBE is acceptable.

3.7.1.4.2 Certified Seismic Design Response Spectra

FSAR Section 3.7.1.1.1, “Design Ground Motion Response Spectra,” applies the design response spectra, which would become the CSDRS once the NuScale FSAR is approved, as an outcrop motion at the finished grade in the free field at the foundation level of the Seismic Category I and II structures. The CSDRS is applied at three mutually orthogonal directions—two horizontal and one vertical. In FSAR Figure 3.7.1-1, “NuScale Horizontal CSDRS at 5 Percent Damping,” and Figure 3.7.1-2, “NuScale Vertical CSDRS at 5 Percent Damping,” compare the CSDRS and the RG 1.60 spectra at 5-percent damping for the horizontal and vertical directions, respectively. The CSDRS are the same in the two horizontal directions, which are identified as north-south (N-S) and east-west (E-W). The horizontal and vertical components of the CSDRS have a peak ground acceleration of 0.5g and 0.4g, respectively.

FSAR Table 3.7.1-1, “Certified Seismic Design Response Spectra Control Points at 5 Percent Damping,” provides the control points for the CSDRS at 5-percent damping. The applicant stated that the CSDRS are broad spectra that are similar in shape to the response spectra in RG 1.60. The comparison of the spectra shows that the CSDRS bound the RG 1.60 spectra anchored at 0.1g in both the horizontal and the vertical directions. Although the CSDRS and the RG 1.60 response spectra are similar, the following illustrate their differences:

- The CSDRS are not scaled from the RG 1.60 horizontal and vertical spectra to include an extended range of potential sites and experience from earthquakes.

- For the CSDRS, additional control frequency points are established below 3.5 hertz (Hz), and the control points above 3.5 Hz are shifted to higher frequencies.

- The zero-period acceleration frequency is increased from 33 Hz to 50 Hz.

This new broadband spectrum with the above characteristics, when approved, will be used as the CSDRS for the NuScale standard design. Although the CSDRS departs from the RG 1.60 guidance, the guidance provides only one example of an acceptable shape that can be used in the design of structures. The staff evaluated the applicant’s proposal and determined that the CSDRS are reasonable and described in sufficient detail for the FSAR. The use of a broadband spectral shape similar to that in RG 1.60 ensures that the resulting generic design has the potential for use at many sites, as anticipated by the applicant.

3.7.1.4.3 Certified Seismic Design Response Spectra-High Frequency

FSAR Section 3.7.1.1.1.2, “Certified Seismic Design Response Spectra-High Frequency,” describes the CSDRS-HF to include hard rock sites that may also be used for the NuScale design of Seismic Category I structures. The CSDRS-HF has a narrow frequency range below approximately 10 Hz and greater frequency range above approximately 10 Hz than the CSDRS. The CSDRS-HF is applied at three mutually orthogonal directions—two horizontal and one vertical. In FSAR Figure 3.7.1-3, “NuScale Horizontal CSDRS-HF at 5 Percent Damping,” and Figure 3.7.1-4, “NuScale Vertical CSDRS-HF at 5 Percent Damping,” the applicant compared the CSDRS and the CSDRS-HF at 5-percent damping for the horizontal and vertical directions, respectively. The CSDRS-HF are the same in the two horizontal directions (N-S and E-W).

FSAR Table 3.7.1-2, "Certified Seismic Design Response Spectra—High Frequency Control Points at 5 Percent Damping," provides the control points for the CSDRS-HF at 5-percent damping. The peak ground acceleration of the CSDRS-HF is 0.5g for both the horizontal and vertical directions.

The information and referenced figures provided by the applicant in FSAR Section 3.7.1.1, contain sufficient detail to demonstrate that the design ground motion spectra (CSDRS and CSDRS-HF) envelop the ground motion response spectra (GMRS) of most soil and hard rock sites. The applicant's approach to specifying the design ground motion spectra is consistent with the acceptance criterion in DSRS Section 3.7.1.II.1 and therefore is acceptable. The applicant demonstrated that the CSDRS bound the minimum response spectra anchored to 0.1g, as specified in 10 CFR Part 50, Appendix S. In accordance with Appendix S to 10 CFR Part 50, DSRS Section 3.7.1.II.1 states that, for an FSAR, the postulated CSDRS at the foundation level in the free field must bound the minimum required response spectrum (MRRS) anchored to 0.1g. The MRRS should be a smooth, broadband response spectrum similar to the RG 1.60 spectrum. For NuScale, the MRRS for the horizontal direction is defined as the RG 1.60 spectra anchored to 0.1g. The staff finds this acceptable because the NuScale CSDRS for the horizontal direction is a smooth, broadband spectrum that envelops the RG 1.60 response spectrum.

In summary, the staff finds the NuScale CSDRS and CSDRS-HF acceptable because both spectra (1) are smooth, broadband response spectra, (2) are specified in accordance with the guidance in DSRS Section 3.7.1 for three mutually orthogonal directions, and (3) comply with the requirement in 10 CFR Part 50, Appendix S, for enveloping the MRRS anchored at 0.1g.

3.7.1.4.4 Design Ground Motion Time Histories

FSAR Section 3.7.1.1.2, "Design Ground Motion Time History," states that the design ground motion consists of six sets of time histories (five for the CSDRS and one for the CSDRS-HF), with each set consisting of three components (the two horizontal components for the E-W direction and N-S direction and the vertical component). The associated time histories were developed to envelop the CSDRS and the CSDRS-HF in conformance with the acceptance criteria in DSRS Section 3.7.1.II.1.B, Option 1, Approach 2, Revision 0. The sections below present the staff's technical evaluation of the seed records and design ground motion time histories.

<u>Seed Records for the Design Ground Motion Time Histories</u>

The five sets of time histories used to match or envelop the CSDRS were based on the three ground motion components recorded from the magnitude 7.3 Landers, CA, earthquake (Yermo) event that occurred on June 28, 1992; the magnitude 6.9 Loma Prieta, CA, earthquake (Capitola) event that occurred on October 17, 1989; the magnitude 7.6 Chi-Chi, Taiwan, earthquake (Chi-Chi) event that occurred on September 21, 1999; the magnitude 7.4 Kocaeli, Turkey, earthquake (Izmit) event that occurred on August 17, 1999; and the magnitude 6.9 Imperial Valley, CA, earthquake (El Centro) event that occurred on May 18, 1940. The same magnitude 7.3 Landers, CA, earthquake that was recorded at the Lucerne station was also used to match the CSDRS-HF.

These actual seed records were selected to generate the design ground motion time histories based on the intensity, duration, frequency content, and epicenter distance from the recording station. The applicant also indicated that the cross-correlation coefficients between the two components of each of the modified time histories are less than 0.16; therefore, these recorded time histories are statistically independent. The total duration for each of the six-time histories is greater than 20 seconds. The strong ground motion duration for each of the modified time histories was shown to be greater than 6 seconds with a time step of 0.005 seconds.

Evaluation of CSDRS and CSDRS-HF Compatible Ground Motion Time Histories

FSAR Section 3.7.1.1.2, describes how the design time histories meet the acceptance criteria in DSRS Section 3.7.1.II.1.B, Revision 0, Option 1, Approach 2. The applicant provided the following numerical values to show how the design time histories meet the DSRS acceptance criteria in the frequency range of 0.2 Hz to 100 Hz:

- The strong motion durations, defined as the time required for the cumulative Arias Intensity to rise from 5 to 75 percent, range from 6 to 18.165 seconds in the N-S direction, 6.775 to 14.45 seconds in the E-W direction, and 6.115 to 15.7 seconds in the vertical direction, as shown in FSAR Table 3.7.1-4, "Duration of Time Histories." With the exception of the strong motion duration of 5.265 seconds for the N-S time history recorded at Station Izmit, the strong motion durations listed in Table 3.7.1-4 exceed the minimum acceptable duration of 6 seconds specified in DSRS Section 3.7.1. Regarding the Izmit N-S time history, the applicant explained that strong shaking begins slightly before the 5 percent time and continues beyond the 75 percent time, thereby meeting the intent of the minimum duration requirement. The staff reviewed the Arias intensity curve for the Izmit N-S time history provided in FSAR Figure 3.7.1-11 and observed a steep slope, indicating strong shaking occurring slightly before the 5 percent time and extending beyond the 75 percent time. This confirms the applicant's assertion that the strong motion duration for the N-S component of the Izmit time history is acceptable.

- The time increment is 0.005 seconds, which is small enough to provide a Nyquist frequency of 100 Hz.

- The absolute values of the correlation coefficients in FSAR Table 3.7.1-3, "Cross-Correlation Coefficients," which range from 0.0071 to 0.0951 (E-W/N-S), 0.0159 to 0.1162 (E-W/vertical (VT)), and 0.0141 to 0.0862 (N-S/VT), are smaller than 0.16. This shows that the acceleration time history pairs are statistically independent.

- The comparison of the six computed 5-percent-damped, compatible time histories to the CSDRS and CDSRS-HF in FSAR Table 3.7.1-5, "Comparison of Response Spectra to CSDRS and CSDRS-HF," shows the maximum difference to be 9.3 percent below target and 29.96 percent above target. No frequency point in any of the CSDRS and the CSDRS-HF compatible time histories is greater than 30 percent and more than 10 percent below the target response spectra.

- The power spectrum density of the time histories was computed. FSAR Figure 3.7.1-13a, "Power Spectral Density Curves CSDRS Compatible Time Histories," and Figure 3.7.1-13b, "Power Spectral Density Curves CSDRS-HF Compatible Time Histories," show no significant gaps in energy at any frequency over the frequency range of 0.1 to 100 Hz.

In FSAR Section 3.7.1, the applicant established its seismic design parameters of the standard design to include both the CSDRS and CSDRS-HF as its standard plant design basis. Because the applicant established both the CSDRS and CSDRS-HF as its standard site parameters, it implies that the standard seismic design uses both spectra as input to the design of all the SSCs.

In summary, the applicant used DSRS Section 3.7.1.II.1.B, Option 1, Approach 2, to envelop the NuScale CSDRS for the 5-percent damped response spectra specified for the NuScale standard design and ensured that sufficient power is contained over the entire frequency range of interest for the NuScale standard design. Based on the information provided by the applicant, the staff finds the NuScale design acceleration time histories to be acceptable because the response spectra generated from the design time histories satisfy the enveloping criteria in DSRS Section 3.7.1.II.1.B.

3.7.1.4.5 Percentage of Critical Damping Values

FSAR Section 3.7.1.2, “Percentage of Critical Damping Values,” states that the damping values used for the analysis of the Seismic Category I and II SSCs are based on RG 1.61, Revision 2, "Damping Values for Seismic Design of Nuclear Power Plants." The staff confirmed that the applicant used values of critical damping that are consistent with those in RG 1.61. The staff finds this acceptable for use in subsequent dynamic analysis.

<u>Structural Damping</u>

The applicant indicated that the NuScale Licensing Topical Report, TR-0920-71621-P-A, Revision 1, “Building Design and Analysis Methodology for Safety-Related Structures,” (ML22062B056) provides analytical models with damping values and stiffness properties based on the actual stress state of the structural members under the most critical seismic load combination. Staff evaluation of the damping values used by the applicant in analysis and design of the structural members is provided in SER Section 3.8.4.

<u>Soil Damping</u>

In FSAR Section 3.7.1.2.3, “Soil Damping,” the applicant described the dynamic properties of the soil and rock materials subject to a seismic event. The applicant stated that the shear modulus and the damping ratio, which are the dynamic properties of the soil and rock materials, are dependent on the shear strain levels induced during the shaking of an earthquake motion. Soil shear modulus decreases with the increase of soil shear strain, whereas the damping increases with the increase of the soil shear strain. The applicant used industry practices to develop the soil degradation and damping functions and provided FSAR Figure 3.7.1-17, “Soil Shear Modulus Degradation Curves,” and Figure 3.7.1-18, “Strain Dependent Soil Damping Curves,” which show the soil degradation and damping curves at different depths.

The applicant provided numerical values of the shear modulus degradation and damping ratio of the soil, gravel, and rock sites. FSAR Table 3.7.1-6, “Soil Shear Modulus Degradation and Strain-Dependent Soil Damping (0–120 ft)”; Table 3.7.1-7, “Soil Shear Modulus Degradation and Strain-Dependent Soil Damping (120 ft–1,000 ft)”; and Table 3.7.1-8, “Strain-Dependent Soil Shear Moduli and Soil Damping Ratios for Gravel and Rock,” show the tabulated values of the degradation and damping curves as a function of the shear strain. The applicant stated that the maximum soil damping is limited to 15 percent.

The staff finds the information on soil damping to be acceptable because the applicant developed soil profiles based on strain-dependent shear modulus and damping curves for different layers of the profile. The damping values are less than the prescribed limit of 15 percent. The staff finds the soil strain-dependent modulus and damping parameters to be acceptable for use in the dynamic analysis of the NuScale standard design as they are consistent with the guidance in Standard Review Plan (SRP) Section 3.7.1.II.2.

3.7.1.4.6 Supporting Media for Seismic Category I Structures

In FSAR Section 3.7.1.3, "Supporting Media for Seismic Category I Structures," the applicant described the supporting media for its Seismic Category I structures. The NuScale Seismic Category I structures consist of the RXB and CRB. The standard design considers three subgrade cases, including soft soil (Type 11), rock (Type 7), and hard rock (Type 9). FSAR Tables 3.7.1-9 through 3.7.1-11, provide the number of layers, thickness, depth, shear wave velocity, weight density, and Poisson's ratio for each layer of the three generic soil profiles, respectively.

FSAR Figure 3.7.1-19, "Shear Wave Velocities for All Soil Types," shows the shear wave velocities for the three soil profiles. The three soil profiles considered in the NuScale standard design represent a range of expected soil conditions. The SSI analysis of the NuScale Seismic Category I structures used the generic soil profiles in FSAR Tables 3.7.1-9 through 3.7.1-11.

For each soil type, the strain-compatible properties associated with each of the five CSDRS compatible time histories are averaged so that a single set of soil properties can be used per soil type. The applicant presented the average strain-compatible soil properties in FSAR Tables 3.7.1-12 and 3.7.1-13. For the CSDRS-HF, the applicant used only one set of compatible time histories; therefore, no averaging was performed. FSAR Tables 3.7.1-14, show the strain-compatible properties for the CSDRS-HF time histories for Soil Types 9. The applicant also provided figures that illustrate the strain-compatible damping for the soil types used with the five CSDRS compatible time histories and the rock types used with the single CSDRS-HF compatible time histories.

The staff reviewed the description of the supporting media for NuScale's Seismic Category I structures to ensure that the application included sufficient information. The applicant adequately described the supporting media for its Seismic Category I structures, including the depth of the three soil types over bedrock, the characteristics of the soil layering, and the soil properties. The applicant provided tables and figures that show the shear wave velocity; shear modulus; material damping, including the strain-dependent effect; and the density of the soil types as a function of depth. The staff finds the descriptive information and referenced tables and figures in FSAR Section 3.7.1.3 acceptable because (1) they contain sufficient information on the supporting media and (2) they are consistent with the acceptance criteria in DSRS Section 3.7.1.II.3.

3.7.1.5 Combined License Information Items

Table 3.7.1-1 lists the COL information item numbers and descriptions related to seismic design parameters from FSAR Table 1.8-1.

Table 3.7.1-1: NuScale COL Information Items for FSAR Section 3.7.1

Item No.	Description	FSAR Section

COL Item 3.7-1	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific safe shutdown earthquake.	3.7.1
COL Item 3.7-2	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific time histories. In addition to the above criteria for cross correlation coefficients, time step and earthquake duration, strong motion durations, comparison to response spectra and power spectra density, the applicant will also confirm that site-specific ratios V/A and AD/V^2 (A, V, D, are peak ground acceleration, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra.	3.7.1
COL Item 3.7-3	An applicant that references the NuScale Power Plant US460 standard design will include an analysis of the performance-based response spectra established at the surface and intermediate depth(s) that take into account the complexities of the subsurface layer profiles of the site and provide a technical justification for the adequacy of vertical to horizontal (V/H) spectral ratios used in establishing the site-specific foundation input response spectra and the performance-based response spectra for the vertical direction.	3.7.1
COL Item 3.7-4	An applicant that references the NuScale Power Plant US460 standard design will: • develop a site-specific strain-compatible soil profile. • confirm that the criterion for the minimum required response spectrum is satisfied. • determine whether the seismic site characteristics fall within the seismic design parameters such as soil layering assumptions used in the standard design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity.	3.7.1

3.7.1.6 Conclusion

The staff finds that the applicant has adequately addressed seismic design parameters in accordance with the acceptance criteria set forth in DSRS Section 3.7.1, and on this basis, the staff concludes that the regulatory requirements delineated in Section 3.7.1.3 of this report are satisfied.

3.7.2 Seismic System Analysis

3.7.2.1 Introduction

For the seismic design of nuclear power plants, 10 CFR Part 50, Appendix A, GDC 2 requires the design basis to reflect appropriate consideration of the most severe earthquakes that have been historically reported for a site and the surrounding area. Two levels of design earthquake ground motions are considered, the SSE and OBE. 10 CFR Part 50, Appendix S requires the nuclear power plant be designed so that, if the SSE ground motion occurs, certain SSCs will remain functional and within applicable stress, strain, and deformation limits. 10 CFR Part 50, Appendix S also requires that the seismic analysis must account for SSI effects and the expected duration of the vibratory motion. For the NuScale US460 standard design, the OBE is set at one-third of the SSE, and in accordance with 10 CFR Part 50, Appendix S, an explicit response or design analysis is not required for the OBE. This section of the SER documents the staff's evaluation of the methods used by the applicant to perform seismic analyses and their results for the Seismic Category I structures of the NuScale US460 standard design.

3.7.2.2 Summary of Application

FSAR: FSAR Section 3.7.2 provides information associated with seismic system analysis as summarized below:

The NuScale standard design includes two site-independent Seismic Category I structures that are portions of the RXB and portions of the CRB. The RXB is designed to house up to 6 installed Nuclear Power Modules (NPMs). The design-basis seismic analysis is performed with 6 NPMs in place. The applicant also discussed the effect on the RXB if a seismic event were to occur during operation with less than the full complement of 6 NPMs. Portions of the Radioactive Waste Building (RWB) are classified as non-safety related, Seismic Category III and the applicant discussed potential interaction of the Seismic Category III RWB with the Seismic Category I RXB. The RXB includes the ultimate heat sink (UHS) pool, which contains a large body of water. The UHS pool consists of the reactor pool, spent fuel pool, and refueling pool. The dry dock is assumed to be full of water for the design-basis seismic analysis. Because both the NPMs and water in the pool contribute a large amount of weight to the global mass of the RXB, they notably affect the dynamic characteristics of the building.

The applicant used linear equivalent static analysis, linear dynamic analysis, complex frequency response analysis, or nonlinear analysis method to analyze the response of structures to the design-basis earthquake ground motion accounting for the effects of soil-structure and fluid-structure interaction. The applicant also evaluated structure-soil-structure interaction (SSSI) to capture potential seismic interactions between adjacent structures (i.e., the RXB and RWB) through the medium of soil. The elements of structures, soils, and fluids are modeled using three-dimensional finite elements. The results from seismic response analysis include member forces and moments, displacements, soil pressures, and nodal acceleration time histories from which the in-structure response spectra (ISRS) are developed. The analysis is performed in each of the three orthogonal directions of the earthquake ground motion - two horizontal and one vertical.

Design of the Seismic Category I SSCs of the NuScale standard plant is based on the CSDRS shown in FSAR Figure 3.7.1-1 and Figure 3.7.1-2 and on the CSDRS-HF shown in FSAR Figure 3.7.1-3 and Figure 3.7.1-4. The seismic design of the NuScale standard plant considers a set of generic subsurface profiles ranging from soft soil to hard rock, as described in FSAR Section 3.7.1.3. The staff evaluation of the CSDRS and CSDRS-HF as well as the generic subsurface profiles used for the NuScale standard design is provided in Section 3.7.1 of this report.

ITAAC: The ITAAC associated with FSAR Section 3.7.2 are evaluated in SER Section 14.3.

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

Technical Reports:

- NuScale Licensing Topical Report, TR-0118-58005-P-A, Revision 2, “Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis” (ML20353A440)

- NuScale Licensing Topical Report, TR-0920-71621-P-A, Revision 1, "Building Design and Analysis Methodology for Safety-Related Structures," (ML22062B056)

- NuScale Technical Report, TR-121515-P, Revision 2, “NuScale Power Module Seismic Analysis” (ML25066A255 (proprietary); ML25066A254 (non-proprietary))

3.7.2.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, GDC 2, as it requires that the SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions and that the design bases for these SSCs reflect appropriate consideration of the most severe earthquakes that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 50, Appendix S, as it requires that, for the Safe Shutdown Earthquake (SSE) ground motion, certain SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion associated with the SSE through design, testing, or qualification methods. The evaluation must account for soil-structure interaction effects and the expected duration of the vibratory motion. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. The horizontal component of the SSE ground motion in the free field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.

- 10 CFR 52.137(a)(1), as it requires that an FSAR must include the site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters.

- 10 CFR 52.137(a)(20), as it requires that an FSAR must include the information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.

The guidance in DSRS Section 3.7.2, Revision 0, “Seismic System Analysis,“ (ML15355A384) lists the acceptance criteria adequate to meet the above requirements and review interfaces with other DSRS sections. In addition, the following guidance provides acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.60, Revision 2, "Design Response Spectra for Seismic Design of Nuclear Power Plants," issued July 2014

- RG 1.61, Revision 2, "Damping Values for Seismic Design of Nuclear Power Plants," issued March 2023

- RG 1.92, Revision 3, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," issued October 2012

- RG 1.122, Revision 1, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components," issued February 1978

- DC/COL-ISG-01, "Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion in Design Certification and Combined License Applications," dated May 19, 2008

- DC/COL-ISG-17, "Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analysis," dated March 24, 2010

3.7.2.4 Technical Evaluation

In this section, the staff described its evaluation of the applicant's seismic analysis for the site-independent Seismic Category I structures of the NuScale standard design. The specific areas of review include seismic analysis methods, analytical modeling for SSI effects, development of ISRS, combination of spatial and modal responses, consideration of torsional effects, analysis procedure for damping, and interaction between Seismic Category I and non-Seismic Category I structures. The staff reviewed the information in FSAR Section 3.7.2, "Seismic System Analysis" against the acceptance criteria of DSRS Section 3.7.2, "Seismic System Analysis" and the regulatory guides (RGs) and interim staff guidance referenced above. Meeting the applicable acceptance criteria provides assurance that Seismic Category I structures will be adequately designed to withstand the effects of the SSE and therefore will be able to perform their intended safety functions during and following the earthquake.

The applicant performed seismic SSI analysis accounting for the effect of fluid-structure interaction using the methodology in the NuScale Licensing Topical Report, TR-0118-58005-P-A, Revision 2, "Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis" (ML20353A440). In this methodology, the traditional SASSI-based methodology is used to create the frequency-dependent soil impedance matrices and seismic load vectors that are imported into the ANSYS model to be combined with the building and fluid substructures to perform integrated soil-structure-fluid interaction analysis.

Analysis of the Seismic Category I structures within the scope of the NuScale standard design considered two different sets of design response spectra (CSDRS and CSDRS-HF), three generic soil profiles (soft soil, rock, and hard rock), six different seed time histories (Yermo, Capitola, Chi-Chi, Izmit, El Centro, and Lucerne), and two different concrete stiffness conditions (uncracked and cracked). The analysis also used two different building models (the double building (DB) and the CRB model), and the DB model consisting of the RXB and the RWB captures the SSSI effects.

The sections below present the staff's evaluation of the seismic system analysis for the NuScale standard design. Section 3.7.1 of this report presents the staff's evaluation of the seismic design parameters, and Section 3.7.3 of this report presents the staff's evaluation of the seismic subsystem analysis.

3.7.2.4.1 Seismic Analysis Methods

FSAR Section 3.7.2.1, "Seismic Analysis Methods" describes analysis methods, computer programs, and finite element models used for the seismic analysis of Seismic Category I SSCs. The applicant stated that these SSCs are designed to withstand the effects of the SSE and are analyzed using the linear equivalent static analysis, linear dynamic analysis, complex frequency response analysis, or nonlinear analysis method. The applicant analyzed the Seismic Category I portions of the site-independent structures, the RXB and CRB, using the frequency-domain complex response analysis method discussed below.

Frequency-Domain Soil-Structure-Fluid Interaction Analysis

In FSAR Section 3.7.2.1, the applicant explained that a frequency-domain analysis methodology in the NuScale Topical Report, TR-0118-58005 is used to account for the effects of soil-structure-fluid interaction in developing the design-basis seismic demands for the SSCs in the NuScale standard design. Specifically, the applicant used in the topical report the so-called "soil library" that contains the soil impedance matrices for the excavated soil volume and the seismic load vectors associated with the free field ground motion. The soil library is then combined with the structure model and the fluid model in ANSYS for frequency-domain harmonic analysis to analyze the effects of soil-structure-fluid interaction. The topical report was submitted by NuScale and reviewed and approved by the NRC staff in December 2020, following the issuance of the NuScale US600 Design Certification. Certain aspects of the methodology described in the topical report and associated staff evaluation applicable to the US460 standard design are discussed below.

Background

A nuclear power plant may include large, complex structures with interacting soil, structure, and fluids during an earthquake. For example, the NuScale US460 standard design includes a large pool of water serving as a UHS and fluid-structure interaction as well as SSI is an important phenomenon to consider. As such, coupled soil-structure-fluid interaction effects need to be accounted for in the design of the SSCs. However, a single integrated frequency-domain analysis tool that can evaluate the effects of soil-structure-fluid interaction along with operating loads was not available. Therefore, analysis of structure with soil-structure-fluid interaction and other operating loads was typically performed using a piecewise approach involving several resource-intensive steps.

In the nuclear industry, seismic SSI analysis has been typically performed using the "System for Analysis of Soil-Structure Interaction" (SASSI) computer code. Since SASSI was first developed at the University of California at Berkeley in 1981, several SASSI versions have been issued by different entities with various added features. However, all SASSI versions are built upon the same source code as the original version. While SASSI has capabilities for handling the effect of soil-structure interaction, it does not provide an integrated analytical framework for considering the effect of fluid-structure interaction and other operating loads. Topical Report, TR-0118-58005, proposed a methodology to perform seismic analysis in the frequency domain considering interactions among the structure, soil, fluid, and major equipment in a single, integrated analysis framework. The topical report also included example problems to demonstrate the applicability and adequacy of the proposed methodology to perform seismic soil-structure-fluid interaction analysis.

Analysis Method

Elements of the proposed analysis methodology consist of substructures representing interacting entities involved in the analysis, i.e., the soil substructure, building substructure, and fluid substructure. These substructures collectively represent a coupled soil-structure-fluid interactive system analyzed for a prescribed ground motion. Different soil substructures, representing different site soil conditions, can be created and an integrated analysis can be performed for each different soil substructure without impacting other substructures. The topical report uses two computer codes to develop quantities representing these substructures: (1) SASSI is used to calculate soil impedance matrices and seismic load vectors for the soil substructure and then stores them for different soil substructures in the soil library and (2) ANSYS is used to develop stiffness, mass, and damping matrices that represent the substructures for other interacting entities (e.g., structures and fluids) involved in the integrated analysis.

An SSI problem is typically solved in the frequency domain because soil modulus and damping are frequency dependent, hence a frequency-by-frequency solution scheme for the SSI is more appropriate. However, the properties of the fluid are not frequency dependent in general and are commonly represented using acoustic elements, hence a fluid-structure interaction problem can be solved either in the time domain or in the frequency domain. The topical report integrates all the interacting entities in the frequency domain for the solution of a soil-structure-fluid interaction problem. The complex frequency response analysis is used to obtain time-domain response to transient loading such as seismic ground motion. The applicant proposed a soil library that contains a series of pre-calculated soil impedance matrices and seismic load vectors for soil substructures. The soil impedances and load vectors are frequency dependent and are calculated at each analysis frequency using the SASSI code. The excavated soil impedance matrix is developed by assembling and inverting the soil flexibility matrix for a layered half-space, and the seismic load vector is obtained as the product of the soil impedance matrix and free field ground motion at the interaction nodes.

The applicant's approach to handling a soil-structure-fluid interaction problem in the frequency domain is acceptable because the frequency-domain solution method can be applied to both the SSI and fluid-structure interaction problems. The staff also determined that the applicant's proposed soil library, which provides an efficient method for calculating and storing the excavated soil impedances and seismic load vectors, is acceptable because the parameters used in the soil library are derived from the established framework of the SASSI methodology which has been validated and widely used by the nuclear industry and evaluated and accepted by the NRC staff. The adequacy of the methodology was further validated through example problems in the topical report.

Applicability of the Topical Report Methodology to US460 Standard Design

The NRC staff previously reviewed and approved the methodology described in TR-0118-58005, with limitations and conditions. The staff's approval of the topical report was limited to the proposed analysis methodology applied to problems that satisfy the assumptions included in the topical report. These assumptions include (1) all material properties are linear elastic during the analysis, (2) the behavior of boundary conditions and constraints is linear, and (3) the seismic load is represented by vertically propagating shear and compressive waves. The staff confirmed that the seismic analysis for the US460 standard design is performed within the applicable limitations and conditions set forth in the topical report.

Therefore, the staff concludes that the frequency-domain soil-structure-fluid interaction analysis methodology approved in the topical report is applicable to the seismic analysis of SSCs in the NuScale US460 standard design.

Computer Programs

In FSAR Section 3.7.2.1.1, "Computer Programs," the applicant indicated that commercially available computer programs ANSYS, SDE SASSI, and ACS SASSI were used in the seismic analysis of the NuScale Seismic Category I SSCs. ANSYS is used to model structural elements including the reinforced concrete and steel-plate composite (SC) elements as well as fluid elements modeling the reactor pool water. SASSI is used to generate the soil library that includes soil impedance matrices and seismic load vectors. The soil library is then imported into ANSYS to be combined with the building and fluid models to perform integrated soil-structure-fluid interaction analysis.

ANSYS is a general-purpose, commercially available finite element program that has been widely scrutinized and applied by the engineering community including nuclear industry. It has been used in a variety of engineering applications including static and dynamic analysis of structural systems. ANSYS was also used to support the Topical Report, TR-0118-58005, and the staff determined the program can be used for the analysis of Seismic Category I SSCs in the NuScale standard design without further demonstration because the program is generally recognized in the public domain and has sufficient history of use to justify its applicability and adequacy.

SASSI is a computer code developed for seismic SSI analysis and has been broadly used in the nuclear industry. SASSI performs complex response analysis in the frequency domain to solve the equations of motion for the soil-structure interactive system subjected to transient loading such as the earthquake ground motion. Several different SASSI versions have been developed by different entities with added features; however, these SASSI versions share the same source code logic as the original version published in 1981. Both SDE SASSI and ACS SASSI, used in seismic analyses for the US460 standard design, were previously reviewed by the NRC staff for their applicability and technical adequacy as part of staff's licensing reviews and found them acceptable. Specifically, SDE SASSI was used to support the Topical Report, TR-0118-58005, discussed above, and ACS SASSI was used to support the combined license (COL) application for North Anna Unit 3. The NRC staff's evaluations of these computer codes are documented in its respective safety evaluation reports (SERs) (ML20353A440 for the Topical Report, TR-0118-58005, and ML16305A135 for the North Anna Unit 3 COL application).

3.7.2.4.2 Natural Frequencies and Responses

The staff reviewed the natural frequencies and responses of the structures in the NuScale standard design. FSAR Section 3.7.2.2, "Natural Frequencies and Responses" provides information on the dynamic modal properties of the models used in the seismic analysis of the Seismic Category I structures, including the natural frequencies and modal mass participation ratios. The applicant used the standalone RXB and CRB models with a fixed-base boundary condition to generate dynamic modal properties. Although the methodology used in SSI analysis is not based on traditional modal superposition, a fixed-base modal analysis of the structure used in the SSI analysis is needed to inform the analyst in selecting the frequencies of analysis and to evaluate the adequacy of other dynamic properties of the structure model used in the SSI analysis. The applicant showed that the cumulative mass participation ratios in all three directions of the design ground motion at the cutoff frequencies are sufficiently high demonstrating that important modes of vibration for the building are accounted for in the seismic SSI analysis of the building.

The applicant also provided responses, including seismically induced accelerations, displacements, forces, moments, soil pressures, and in-structure response spectra, at key locations of the RXB and CRB. These responses form the design-basis seismic demands used in the structural design of these buildings and subsystems housed in them as discussed in FSAR Sections 3.7.3, 3.8.2, 3.8.4, and 3.8.5. The staff found the type and scope of information provided in FSAR Section 3.7.2 on the dynamic modal properties and seismic responses of the Seismic Category I structures in the NuScale standard design to be acceptable because they are consistent with the acceptance criteria in DSRS Section 3.7.2.II.2.

3.7.2.4.3 Procedures Used for Analytic Modeling

In FSAR Section 3.7.2.1, "Seismic Analysis Methods" and in FSAR Section 3.7.2.3, "Procedures Used for Analysis Modeling," the applicant described the methods of analytical modeling and approaches for the analysis of Seismic Category I structures subjected to the design-basis earthquake ground motion. The staff reviewed the methods and approaches used by the applicant for their acceptability in accordance with the guidance in DSRS Section 3.7.2.II.3.

Reactor Building Model

FSAR Section 3.7.2.1.2, "Finite Element Models" states that the RXB, the reactor building crane (RBC), and pool water are modeled using solid shell (SOLSH190), shell (SHELL181), beam (BEAM188), fluid (FLUID30), surface (SURF154), and mass (MASS21) elements of ANSYS. The applicant explained that thick concrete slabs including the 8 ft-thick basemat and 2 to 3 ft-thick main floor slabs are modeled using the solid shell (SOLSH190) elements to achieve proper geometric representation in the pool region and that other floor slabs including roof slabs are modeled using shell (SHELL181) elements. FSAR Figure 3.7.2-60 shows the isometric view of the RXB ANSYS model and Figure 3.7.2-61 through Figure 3.7.2-87 show the section views of the RXB ANSYS model.

The applicant described that the RXB houses equipment for operating NPMs and provides anchorages and support for various SSCs. The overall dimensions of the building are 232 ft, 156 ft (excluding penetration shrouds), 171 ft in the east-west, north-south, and vertical directions, respectively. The RXB is deeply embedded with the basemat bottom located approximately 83 ft below grade. The grade level for the RXB is at elevation 100 ft. The east-west exterior SC walls are 5 ft thick, and the north-south exterior SC walls are 4 ft thick. The typical thickness for the structural interior SC walls is 4 ft and the primary floor slabs are 2 or 3 ft-thick reinforced concrete. The reinforced concrete basemat is 8 ft thick (and 9ft in the pool region), and the 3 ft-thick roof is comprised of reinforced concrete slab and steel girders.

A predominant feature of the RXB is the UHS that includes the spent fuel pool, refueling area pool, and the reactor pool. The normal reactor pool water depth is 53 ft. The dry dock is also assumed to be full of water and part of the UHS for the seismic analysis. The applicant performed a sensitivity study including comparison of seismic demands between the full and empty dry dock cases and the staff's evaluation is provided in SER Section 3.7.2.4.4 "Soil-Structure Interaction Analysis." The UHS pool contributes a large amount of water mass to the global mass of the RXB and this water mass influences the dynamic characteristics of the building. Water mass regions are modeled by fluid finite elements and each fluid element is defined by eight nodes having three translational degrees of freedom plus a pressure degree of freedom at each node. The fluid element is well suited for calculating hydrodynamic pressures accounting for fluid-structure interaction under the earthquake ground motion. A representative hydrodynamic pressure profile on exterior pool walls based on RXB design-basis seismic analyses is provided in FSAR Figure 3.7.2-10b.

The RBC is a bridge crane used to transport modules between the operating locations and the refueling and disassembly area. The RBC travels on rails on the top of the reactor pool walls. The RBC model is coupled to the RXB ANSYS model at interfacing nodes using constraints. For the RXB seismic analysis, the RBC does not hold the NPM load, and the crane is located in the western side of the reactor pool area because this configuration generates a larger response in the building. The RXB seismic analysis generates ISRS that are used as input to the design analysis for the RBC.

The staff reviewed the scope and level of detail of the applicant's description of the RXB and the entities housed in it and finds them sufficient for defining primary structural aspects and properties necessary to develop finite element models for seismic response analysis of the building. The staff also finds the applicant's methods for finite element modeling of the RXB and associated elements including the UHS and RBC to be acceptable because they are in conformance with the acceptance criteria in DSRS Section 3.7.2.II.3 and use elements of a generally recognized code, ANSYS.

<u>NuScale Power Module Model</u>

The NuScale Power Modules (NPMs) are partially immersed in the reactor pool and are not permanently bolted or welded to the pool floor or walls but are constrained to stay in place during and following a seismic event. The NPM base support is a steel skirt restraint comprising four built-up stainless-steel members bracing the NPM skirt in the lateral directions and an annular bearing plate supporting the NPM in the vertical direction. The other three geometrical supports are steel lug restraints placed on the bay walls near the top of the module. The NPM lugs align with a slot in the restraint and each restraint prevents movement in the direction parallel to the wall and allows the NPM to move freely in the vertical direction. The lug restraint provides only horizontal restraint in the in-plane direction of the supporting wall.

A simplified finite element model of the containment vessel (CNV) and the associated water elements representing the water within each bay is used to model the NPMs to be included in the RXB ANSYS model. Each simplified NPM model incorporates five major NPM components including top support structure, reactor pressure vessel, control rod drive mechanism (CRDM), upper reactor vessel internals (RVIs), and lower RVIs. Small components including most piping and valves, manways, instruments, pressurizer heaters, and other small internal components such as bolts are not explicitly modeled because these features do not affect the overall structural behavior of the model and removing them allows for simplified finite element meshing to be used. The applicant explained that the simplified NPM model included in the RXB ANSYS model was developed in such a way that it is dynamically compatible with the detailed NPM model used in the dynamic analysis for the mechanical design of Seismic Category I SSCs that comprise the NPM.

To validate the simplified NPM model included in the RXB ANSYS model, the applicant performed dynamic modal analysis and discussed the results from the simplified NPM model in comparison with those from the detailed NPM model in Section 4.2 of the NuScale Technical Report, TR-121515-P, Revision2, "NuScale Power Module Seismic Analysis." The staff noted that the comparison shows closely matching frequencies and mass participation ratios from the two models. Therefore, the staff concluded that the simplified NPM model captures the overall dynamic behavior of the detailed NPM model and therefore is adequate to be included in the RXB ANSYS model for the seismic response analysis of the RXB.

Control Building Model

The CRB is a reinforced concrete structure and comprises a Seismic Category I portion and a Seismic Category II portion. The main control room (MCR) is housed in the Seismic Category I portion while the Technical Support Center is housed in the Seismic Category II portion of the CRB. The Seismic Category I portion of the CRB has overall dimensions of approximately 120 ft, 55 ft, and 50 ft in the E-W (X), N-S (Y), and vertical (Z) directions, respectively. The Seismic Category I portion of the CRB consists of a 5 ft-thick basemat, 3 ft-thick exterior walls, 2 to 3 ft-thick interior walls, and 2 ft-thick floor slabs. The concrete elements are modeled using shell elements (SHELL181) in ANSYS. Unlike the RXB walls of steel composite (SC) design, the CRB walls are entirely of reinforced concrete design. FSAR Figure 3.7.2-4 shows the isometric view of the CRB ANSYS model and FSAR Figure 3.7.2-11 through Figure 3.7.2-16 show the plan and elevation views of the CRB ANSYS model.

The CRB is embedded 5 ft below grade and is modeled as a surface founded structure. However, the seismic analysis of the CRB requires the excavated soil volume to be defined to form the soil library, so the top two layers of soil are excavated and then reinserted for the CRB to sit on as illustrated in FSAR Figure 3.7.2-4. The staff considered the applicant's modeling of the CRB as a surface founded structure is reasonable because of an insignificant depth of embedment (5 ft). The staff also determined that the applicant's approach for developing the soil library is acceptable because the fictitious excavated soil volume used to form the soil library is put back (or reinserted) so that there is no net change in stiffness of the soil layers supporting the CRB. A soil library that contains information on the soil impedance and seismic load vector is needed for seismic analysis of the CRB following the methodology in the Topical Report, TR-0118-58005.

Double Building Model

The DB model is developed by combining the standalone RXB and the RWB through backfill that surrounds the two buildings. The DB model fills the excavated soil volume which is used to develop the soil library. The NPM and RBC models imported into the RXB ANSYS model are now parts of the DB model. These models are simplified versions of their more detailed versions and are used in the DB model to determine the seismic responses of the buildings subjected to the design-basis ground motion. The in-structure responses so obtained are used as input for detailed design analysis of the NPM and RBC using their detailed models.

The DB model used for the soil-separation design-basis case is built by reducing the stiffness of the engineered backfill in the top 25 ft below grade by 99 percent. This model is built using the DB model compatible with the Soil Type 7 soil library (Soil-7 library). This process results in a total of four DB models with uncracked concrete properties - DB models compatible with Soil-7 (a rock profile), Soil-9 (a hard rock profile), and Soil-11 (a soft soil profile) libraries and a DB model with soil separation and Soil-7 library. The cracking analysis is performed by first extracting the peak element forces from seismic analysis of the uncracked DB models. Then, the structural members identified as cracked are updated by changing their material properties to represent the cracked concrete to form the hybrid cracked/uncracked models as outlined in Section 4.0 of the NuScale Licensing Topical Report, TR-0920-71621-P-A, Revision 1, "Building Design and Analysis Methodology for Safety-Related Structures," (ML22062B056).

The DB models consisting of both cracked and uncracked members are grouped into two different categories, ISRS and design calculation. In developing the hybrid DB models for ISRS calculation, the damping values for the cracked reinforced concrete (RC) and SC members are specified as 7 percent and 5 percent, respectively. For the uncracked RC and SC members, the damping values are set to 4 percent and 3 percent, respectively. For design calculations, the damping values for uncracked RC and SC members are set to be the same as the cracked ones, i.e. 5 percent for SC and 7 percent for RC members. Therefore, two different hybrid models are generated for each DB model: one for ISRS calculation and the other for design calculation. The difference between the two hybrid models is the damping values assigned to uncracked members.

The staff reviewed and found the applicant's approach to hybrid models with different damping values to be acceptable because it is consistent with the guidance in RG 1.61 and the applicable provisions in ASCE Standard 4-16 and ASCE Standard 43-19. The same approach was reviewed and approved by the staff for the Topical Report, TR-0920-71621.

Conclusion on Analysis Models

The staff reviewed the applicant's methods and procedures used in finite element modeling for the seismic system analysis including structural material properties, modeling of stiffness, mass, and damping for structural members, modeling of hydrodynamic effects for the UHS pool, and finite element discretization. The staff found them acceptable because they are consistent with the acceptance criteria in DSRS Section 3.7.2.II.1 and DSRS Section 3.7.2.II.3 and applicable industry consensus standards.

Procedures for structural analysis of the RXB and CRB

FSAR Section 3.7.2.3 describes the general approach for structural analysis of the DB and CRB models, which involves the following steps: (1) create the DB model by incorporating major equipment into the RXB and then combining the RXB with the RWB; (2) perform cracking analysis of the DB and CRB models; (3) perform multiple runs of ANSYS using different combinations of the design ground motions (CSDRS and CSDRS-HF), soil profiles, material damping values, and the building models; (4) perform static analysis with uncracked models; and (5) combine the results to develop bounding design demands.

The staff reviewed the applicant's approach for developing the seismic demand and combining them with other loads to establish the structural design loads for the RXB and CRB and found them acceptable because (1) it uses the seismic design parameters evaluated in SER Section 3.7.1.4, (2) it uses the finite element models evaluated in SER Section 3.7.2.4.3.1, and (3) the seismic demand is combined with loads from other sources to develop the structural design loads for the buildings in accordance with the guidance in DSRS Section 3.8.4. A more detailed staff review on related issues is provided in SER Section 3.8.4.

3.7.2.4.4 Soil-Structure Interaction Analysis

FSAR Section 3.7.2.4, "Soil-Structure Interaction" describes SSI analysis of the Seismic Category I structures subjected to the design-basis earthquake ground motion and states that SSI analysis follows the methodology in the Topical Report, TR-0118-58005. In this methodology, the SSI analysis is performed in the frequency domain and employs the concept of a soil library. For each soil type, soil impedances and seismic loads are calculated using the SASSI code to form a soil library. The soil impedance matrix and seismic load vector are then imported into an ANSYS model for SSI analysis. The ANSYS model also contains fluid elements to capture the effects of fluid-structure interaction. When the ANSYS model is combined with the soil library developed using SASSI, it can address the seismic soil-structure-fluid interaction effects for the RXB that houses the UHS pool.

The applicant used the CSDRS and CSDRS-HF and associated time history sets, as well as the soil types evaluated in Section 3.7.1 of this SER in performing the SSI analysis for the RXB and CRB. The SSI analysis is performed to develop the ISRS and other structural design parameters including the forces and moments in SC walls, forces and moments in reinforced-concrete members, and relative displacements at selected locations. The applicant used four design-basis SSI analysis cases: Baseline-Soil-7, Baseline-Soil-9, Baseline-Soil-11, and Soil-Separation-Soil-7 cases. The three baseline cases respectively use the three soil types (Soil Type 7 for rock, Soil Type 9 for hard rock, and Soil Type 11 for soft soil) as the subsurface soil profile. The Soil-Separation-Soil-7 case is also included as a design-basis case based on the outcome of an applicant's sensitivity study on the effect of soil separation on seismic demands for the RXB, as discussed in SER Section 3.7.2.4.4.1 below. The soil-separation case uses Soil Type 7 as the subsurface soil profile. Soil separation is not considered for the CRB because the building is essentially surface-founded and therefore not embedded in soil layers.

The staff finds the applicant's SSI analyses performed for the RXB and CRB are in accordance with the guidance and acceptance criteria in DSRS Section 3.7.2.II.4 and are consistent with the methodology previously approved by the staff in Topical Report, TR-0118-58005-P-A (ML20353A440), and are therefore acceptable. The staff's safety evaluation on the applicant's sensitivity studies with respect to parameter variations in SSI analysis of the RXB is provided below.

Sensitivity Studies on Parameter Variations

In FSAR Section 3.7.2.10, "Sensitivity Studies on Soil Separation, Empty Dry Dock, and Modularity", the applicant described sensitivity studies on the RXB seismic responses for three different cases of parameter variations: an empty dry dock case, an NPM modularity case, and a soil separation case. The description and outcome of each of the sensitivity studies considered are evaluated in SER Section 3.7.2.4.4.2 to Section 3.7.2.4.4.4 below. The applicant defined the baseline (or reference) case where the uncracked DB model compatible with the Soil-7 library is used without modification, i.e. the dry dock is full of water, all six NPMs are present, and no structure-soil separation exists. Sensitivity of the structural responses to the effect of different cases of parameter variation is evaluated by comparing the selected output quantities to those from the baseline case. The output quantities compared include ISRS curves at selected sets of nodes, reactions at the NPM supports, forces and moments and demand-to-capacity ratios (DCRs) at selected section cuts.

FSAR Figure 3.7.2-47 to Figure 3.7.2-59 show the ISRS calculated at 5 percent damping for different node groups for the baseline case and three sensitivity cases. The presented ISRS are normalized with respect to the peak value for the baseline case. FSAR Table 3.7.2-11 summarizes the sensitivity ratios for the NPM support reactions calculated for different types of constraints: X-direction shear lugs, Y-direction shear lugs, X-direction basemat constraint, Y-direction basemat constraint, and Z-direction basemat constraint. FSAR Table 3.7.2-12a and Table 3.7.2-12b present the ratios of the calculated forces at section cuts from the sensitivity cases to those from the baseline case, and FSAR Table 3.7.2-13a and Table 3.7.2-13b present the calculated DCRs at section cuts from the sensitivity cases and baseline case.

Effect of Empty Dry Dock

An empty dry dock refers to the case with no water in the dry dock area. The sensitivity study compared the ISRS and structural responses of the RXB with an empty dry dock to those from the baseline case with the dry dock full of water. As shown in FSAR Figure 3.7.2-47 to Figure 3.7.2-50, the 5 percent-damped normalized ISRS calculated from the full and empty dry dock cases indicate that the response at the dry dock gate is greater when it is empty, but the difference is insignificant elsewhere. The sensitivity study shows that emptying the dry dock has local effects on ISRS that diminish rapidly with distance from the dry dock gate. As shown in FSAR Table 3.7.2-11, the sensitivity ratios for NPM support reactions indicate that the support reactions are not sensitive to the empty dry dock case, with the maximum ratio of the sensitivity-case reaction to the baseline reaction being 1.01, indicating 1 percent increase in NPM support reaction due to the empty dry dock case.

The ratios of the calculated forces at section cuts from the empty dry dock case to those from the baseline case, provided in FSAR Table 3.7.2-12a, show that the force outputs from the sensitivity case are greater than those from the baseline case only for structural components that are local to the empty dry dock area. The applicant also summarized in FSAR Table 3.7.2-13a the DCRs at section cuts from the sensitivity case and baseline case. A similar trend for the calculated force ratios to that of the DCRs is observed for the empty dry dock sensitivity case.

The staff notes that the empty dry dock case results in increase in force outputs only in a localized region around the empty dry dock. The staff also notes that COL Item 3.7-8 requires an applicant that references the NuScale US460 standard design to demonstrate that the site-specific seismic demand is bounded by the FSAR capacity for an empty dry dock condition. On this basis, the staff finds the applicant’s conclusion that the empty dry dock case does not result in significant increase in seismic demands for the RXB and need not be included as part of the design-basis is acceptable.

Effect of NPM Modularity

Modularity refers to the case with a reduced number of NPMs in the RXB. To create the most eccentric NPM responses on the pool and support walls, two NPMs, located on the north side of the pool, are removed, starting from west-most to east. The sensitivity study compared the ISRS and structural responses of the RXB from the modularity sensitivity case to those from the baseline case. As graphically shown in FSAR Figure 3.7.2-51 to Figure 3.7.2-54, the 5 percent-damped normalized ISRS calculated for the RXB with six NPMs (baseline case) and the RXB with four NPMs (modularity case) show that the effects of reducing the number of NPMs are localized with minor impacts on the ISRS outputs. As shown in FSAR Table 3.7.2-11, the sensitivity ratios for NPM support reactions indicate that the support reactions are not sensitive to the modularity case, with the maximum ratio of the sensitivity-case reaction to the baseline reaction being 1.04, indicating 4 percent increase in NPM support reactions due to the modularity sensitivity case.

The ratios of the calculated forces at section cuts from the modularity case to those from the baseline, as shown in FSAR Table 3.7.2-12a, indicate that the force outputs from the modularity case are within 10 percent of the force outputs from the baseline case. Therefore, the modularity case is not considered significant for force outputs. The applicant summarized in FSAR Table 3.7.2-13a the calculated DCRs at section cuts from the sensitivity and baseline cases. The staff observed a similar trend for the calculated force ratios in the DCRs indicating that modularity effects are minor and localized.

The staff notes that the modularity sensitivity case results in an insignificant and localized increase in force outputs and, therefore, finds that the applicant's conclusion that the modularity case needs not be included as part of the design basis is acceptable.

Effect of Potential Soil Separation

Soil separation refers to the case where there is no contact between the backfill soil and the RXB exterior walls. The model for this case is generated by reducing the Young's moduli of the soil layers in the top 25 ft to 1 percent of their original values. The sensitivity study compared the ISRS and structural responses of the RXB with soil separation to those from the baseline case with no soil separation. FSAR Figure 3.7.2-55 to Figure 3.7.2-59 show the 5 percent-damped normalized ISRS calculated from the sensitivity and baseline cases and indicate that soil separation induces significant increase in ISRS throughout the structure. FSAR Table 3.7.2-11 shows the ratios of the NPM support reactions from the sensitivity case to the baseline case and indicates a slight decrease in NPM support reaction due to soil separation.

The ratios of the calculated forces at section cuts from the soil separation case to those from the baseline case, as shown in FSAR Table 3.7.2-12b, indicate that the force outputs from soil separation are greater than those from the baseline by more than 10 percent at multiple section cuts. The staff considers this case to be significant for force outputs. The applicant provided in FSAR Table 3.7.2-13b the calculated DCRs at section cuts from the soil separation and baseline cases. The staff notes that soil separation results in increased DCRs for most of the section cuts.

Based on results from the sensitivity studies, the applicant added the soil-separation case to the design basis calculations. In FSAR Section 3.7.2.1.2.6, "Double-Building Model," the applicant explains that four DB models including those compatible with Soil-7, 9, and 11 soil libraries and the soil-separation model compatible with the Soil-7 soil library constitute the design-basis cases to calculate seismic demands for the structural design of the RXB.

Conclusion on Sensitivity Studies

Based on the above review on the sensitivity studies performed by the applicant, the staff finds that the sensitivity cases of the empty dry dock and the NPM modularity result in insignificant impact on the design-basis demands from seismic SSI analysis of the RXB. The staff, however, determined that the sensitivity case of soil separation is significant and needs to be accounted for in establishing the design-basis seismic demands for the RXB. Therefore, the staff finds the applicant's approach of including the soil separation case in the design basis cases for seismic demand calculations of the RXB is acceptable.

3.7.2.4.5 Development of In-Structure Response Spectra

The staff reviewed the methods and procedures used in developing ISRS in accordance with DSRS Section 3.7.2.II.5 and RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.” These documents provide guidance and criteria for methods acceptable to the staff for developing two horizontal and a vertical ISRS from the response time histories.

The staff reviewed FSAR Section 3.7.2.5, “Development of In-Structure Floor Response Spectra” for procedures used in developing the ISRS for Seismic Category I structures. The applicant stated that the ISRS are generated according to the procedures in RG 1.122. The applicant developed the ISRS from time histories at selected locations computed from separate SSI analyses with three directions of the input ground motion. The ISRS are obtained from SSI analyses of the DB hybrid model and CRB hybrid model for soil Types 7, 9, and 11. The DB model is also analyzed for soil type 7 with soil separation. Six input motions are used in developing the ISRS. As discussed in FSAR Section 3.7.1.1, five of them are compatible with the CSDRS and include earthquake seed time histories based on the Capitola, Chi-Chi, El Centro, Izmit, and Yermo earthquake records, and one compatible with the CSDRS-HF based on the Lucerne earthquake record. The CSDRS compatible input motions are used for Soil Types 7 and 11, and the CSDRS-HF compatible input motion is used for Soil Type 9.

The applicant used the algebraic sum of the response time histories due to each direction of the input ground motion to obtain the directionally combined response time histories, which is acceptable as discussed in SER Section 3.7.2.4.6. The ISRS of the combined response time histories are then calculated for six damping values of 2, 3, 4, 5, 7, and 10 percent. The ISRS are averaged for the five CSDRS input motions, but averaging is not performed for the CSDRS-HF input motion because there is only one input motion. The ISRS at selected nodes that belong to the same group are enveloped. For example, the ISRS of all nodes on the same floor are enveloped to obtain the ISRS for the floor. For the CSDRS, the ISRS from Soil 7, Soil 11, Soil 7-SS (soil-separation) are enveloped. After enveloping, the ISRS are smoothed, and their peaks are broadened by ±15 percent on the frequency axis in accordance with RG 1.122 to account for uncertainties in the structural frequencies due to uncertainties in the material properties of the structure and soil and due to approximations in the modeling techniques used in seismic analysis. The applicant provided ISRS at the RBC supports, NPM skirt and lug supports, and other key locations in the RXB and CRB.

The staff finds the applicant’s process for development of the ISRS from response time histories, combining the three directional response time histories at each location using the algebraic sum, computation of the ISRS at a minimum number of frequencies, and the 15-percent broadening of the peaks in the ISRS, conforms to the guidance in RG 1.122 and meet the acceptance criteria in DSRS Section 3.7.2.II.5 and, therefore, are acceptable.

3.7.2.4.6 Three Components of Earthquake Motion

The staff reviewed the method the applicant used in combining the responses from the three components (two horizontal and one vertical) of the earthquake ground motion in accordance with the guidance in DSRS Section 3.7.2.II.6 and RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” for methods acceptable to the staff for combining the three spatial components of seismic responses.

In FSAR Section 3.7.2.6, “Three Components of Earthquake Motion,” the applicant stated that the three components of the earthquake ground motion are developed as separate input time histories and the response time histories of interest for the SSCs are obtained by performing separate analyses for each of the three components of the earthquake ground motion and summing them algebraically. The staff notes that RG 1.92 allows an algebraic summation of the three directional component responses if these components of the earthquake ground motion are statistically independent. In Section 3.7.1.4.4 of this SER, the staff confirmed that the time histories of the three directional components of each input ground motion used are statistically independent.

The staff finds the applicant's method of combining the three spatial components of seismic responses using the algebraic sum to be in conformance with the guidance in RG 1.92 and meets the acceptance criteria in DSRS Section 3.7.2.II.6 and, therefore, are acceptable.

3.7.2.4.7 Combination of Modal Responses

DSRS Section 3.7.2.II.7 provides guidance for the combination of modal responses with consideration of closely spaced modes and high-frequency modes, when using the response spectrum method or the modal superposition time history method of analysis, to determine the dynamic response of a damped linear system.

In FSAR Section 3.7.2.7, "Combination of Modal Responses," the applicant stated that modal responses in seismic response analysis are combined in accordance with RG 1.92. The staff finds the applicant's method of combining modal responses to be acceptable because the method is in conformance with the guidance in RG 1.92 and meets the acceptance criteria in DSRS Section 3.7.2.II.7.

3.7.2.4.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures, Systems, and Components

The staff reviewed the methods the applicant used to assess non-Seismic Category I structures to determine whether their failure under the SSE conditions could impair the integrity of Seismic Category I SSCs, or result in incapacitating injury to control room occupants, in accordance with the guidance in DSRS Section 3.7.2.II.8.

In FSAR Section 3.7.2.8, "Interaction of Non-Seismic Category I Structures with Seismic Category I Structures," the applicant described that the nearby non-Seismic Category I structures are evaluated and concluded that there is no potential for adverse interaction with Seismic Category I SSCs during the SSE conditions. The staff notes that the nearby non-Seismic Category I structures that are adjacent to the Seismic Category I portions of the RXB and CRB include (1) the RWB that is adjacent to the RXB, and (2) the non-Seismic Category I portion of the CRB that is directly to the north of the Seismic Category I portion of the CRB. The staff also notes that the FSAR includes COL Item 3.7-7 which ensures that an applicant referencing the US460 standard design will confirm that nearby structures exposed to the site-specific SSE will not collapse and adversely affect Seismic Category I portions of the RXB and CRB.

The staff finds the applicant's assessment of potential interaction of non-Seismic Category I structures with Seismic Category I SSCs is acceptable because (1) the assessment is consistent with the acceptance criteria in DSRS Section 3.7.2.II.8 and (2) a COL information item will ensure that nearby non-Seismic Category I structures will not adversely affect the Seismic Category I portions of the RXB and CRB at a proposed site.

3.7.2.4.9 Effects of Parameter Variations on Floor Response Spectra

Staff's evaluation on the effects of parameter variations on floor response spectra is covered in Section 3.7.2.4.4.1 of this SER as part of the ISRS sensitivity studies on parameter variations.

3.7.2.4.10 Use of Constant Vertical Static Factors

DSRS Section 3.7.2.II.10 allows the use of equivalent static load factors to calculate vertical response loads for the seismic design of nuclear structures if the structure can be demonstrated to be rigid in the vertical direction.

However, FSAR Section 3.7.2.10.4, "Use of Constant Vertical Static Factors" indicates that the design of the NuScale Seismic Category I structures does not use constant vertical static factors; instead, the vertical seismic loads are directly generated from the SSI analysis of each structure. Since the applicant did not use constant vertical static factors, no further technical review of this area is needed.

3.7.2.4.11 Method Used to Account for Torsional Effects

The staff reviewed the method the applicant used to account for torsional effects in accordance with DSRS Section 3.7.2.II.11. The DSRS states that an acceptable method to account for torsional effects in the seismic analysis of Seismic Category I structures is to perform a dynamic analysis that incorporates the torsional degrees of freedom and to include the effect of accidental torsion. FSAR Section 3.7.2.11, "Accidental Torsion" stated that finite element analysis (FEA) models that included the torsional degrees of freedom are used in seismic analysis of the Seismic Category I structures. The applicant also explained that the effect of accidental torsion is accounted for in the building design by increasing the demand forces and moments by 5 percent, thus meeting the intent of the guidance in DSRS Section 3.7.2.II.11.

The staff finds the applicant's method to account for torsional effects is acceptable because (1) the dynamic analysis of Seismic Category I structures is performed using building models that included the torsional degrees of freedom and (2) the effect of accidental torsion is adequately accounted for in conformance with the acceptance criteria in DSRS Section 3.7.2.II.11.

3.7.2.4.12 Comparison of Responses

DSRS Section 3.7.2.II.12 states that if both the time history analysis method and the response spectrum analysis method are used to analyze an SSC, the peak responses obtained from these two methods should be compared to demonstrate approximate equivalency between the two methods. However, FSAR Section 3.7.2.12, "Comparison of Results," indicates that the response spectrum method is not used in the evaluation of the NuScale Seismic Category I structures and therefore a direct comparison is not applicable, which is acceptable to the staff. No further technical review of this area is needed.

3.7.2.4.13 Analysis Procedure for Damping

The staff reviewed the applicant's analysis procedure for damping in accordance with the guidance in DSRS Section 3.7.2.II.13, "Analysis Procedure for Damping." In FSAR Section 3.7.2.15, "Analysis Procedure for Damping," the applicant stated that the damping values in RG 1.61 are used in the dynamic analysis of the Seismic Category I SSCs, and, for soil and rock materials, the damping values are obtained based on the strain-compatible soil properties generated for each soil profile. The applicant further indicated that damping values for linear elastic analysis depends on the level of cracking expected in the structural elements under the design-basis ground motion as described in NuScale Topical Report, TR-0920-71621-P-A, Revision 1, “Building Design and Analysis Methodology for Safety-Related Structures."

The staff finds the applicant’s approach to determining damping values used in seismic analysis of Seismic Category I SSCs is acceptable because it is consistent with the guidance in RG 1.61 and meets the acceptance criteria in DSRS Section 3.7.2.II.13.

3.7.2.4.14 Determination of Dynamic Stability of Seismic Category I Structures

DSRS Section 3.7.2.II.14 provides guidance on determination of dynamic stability of Seismic Category I structures. In FSAR Section 3.7.2.14, “Determination of Dynamic Stability of Seismic Category I Structures,” the applicant indicated that FSAR Section 3.8.5, “Foundations” provides relevant information on this technical topic. Staff’s Section 3.8.5 of this report evaluates the dynamic stability of Seismic Category I structures.

3.7.2.5 Combined License Information Items

Table 3.7.2-1 lists the COL information item numbers and descriptions related to seismic system analysis from FSAR Table 1.8-1.

Table 3.7.2-1: NuScale COL Information Items for FSAR Section 3.7.2

Item No.	Description	FSAR Section
COL Item 3.7-5	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of soil separation. The applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra described in Section 3.7.2.	3.7.2
COL Item 3.7-6	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of non-vertically propagating seismic waves on the free-field ground motions and seismic responses of seismic Category I structures, systems, and components.	3.7.2
COL Item 3.7-7	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to a site-specific safe shutdown earthquake will not collapse and adversely affect seismic Category I portions of the Reactor Building and Control Building.	3.7.2
COL Item 3.7-8	An applicant that references the NuScale Power Plant US460 standard design will demonstrate that the site-specific seismic demand is bounded by the Final Safety Analysis Report capacity for an empty dry dock condition.	3.7.2

COL Item 3.7-9	An applicant that references the NuScale Power Plant US460 standard design will perform a soil-structure interaction analysis of the Reactor Building and the Control Building using the NuScale ANSYS models for those structures. The applicant will confirm that the site-specific seismic demands of the standard design for critical structures, systems, and components in Appendix 3B are bounded by the corresponding design certified seismic demands and, if not, the standard design for critical structures, systems, and components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands. Seismic demands investigated shall include forces, moments, deformations, in-structure response spectra, and seismic stability of the structures.	3.7.2

3.7.2.6 Conclusion

The staff finds that the applicant has adequately addressed seismic system analysis in accordance with the acceptance criteria set forth in DSRS Section 3.7.2, and on this basis, the staff concludes that the regulatory requirements delineated in Section 3.7.2.3 of this report are satisfied.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Introduction

FSAR Section 3.7.3, "Seismic Subsystem Analysis," covers the seismic analysis of Seismic Category I subsystems that are not included in main structural systems. These subsystems are reviewed in accordance with DSRS Section 3.7.3, "Seismic Subsystem Analysis." Distribution systems and equipment including their supports (e.g., cable trays, conduit, heating, ventilation, air conditioning, and piping) are reviewed in accordance with SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," and SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures."

3.7.3.2 Summary of Application

FSAR: FSAR Section 3.7.3, describes the seismic analysis methods for NuScale Seismic Category I subsystems that are not included in the main structural systems described in FSAR Section 3.7.2, "Seismic System Analysis." NuScale Seismic Category I subsystems include the NPM, fuel storage rack, RBC, and bioshields. As applicable, FSAR Section 3.7.3 references FSAR Section 3.7.1 for seismic design parameters and FSAR Section 3.7.2 for seismic system analysis.

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

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

Technical Reports: This FSAR section does not reference any TRs.

3.7.3.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, GDC 2, as it requires that the SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions and that the design bases for these SSCs reflect appropriate consideration of the most severe earthquakes that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 50, Appendix S, as it requires that, for the safe shutdown earthquake (SSE) ground motion, certain SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion associated with the SSE through design, testing, or qualification methods. The evaluation must account for soil-structure interaction effects and the expected duration of the vibratory motion. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. The horizontal component of the SSE ground motion in the free field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.

- 10 CFR 52.137(a)(1), as it requires that an FSAR must include the site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters.

- 10 CFR 52.137(a)(20), as it requires that an FSAR must include the information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.

The guidance in DSRS Section 3.7.3, Revision 0, “Seismic Subsystem Analysis,” (ML15355A384) lists the acceptance criteria adequate to meet the above requirements, as well as 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 1.61, Revision 2, “Damping Values for Seismic Design of Nuclear Power Plants,” issued March 2023

- RG 1.92, Revision 3, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” issued October 2012

- RG 1.122, Revision 1, “Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components,” issued February 1978

3.7.3.4 Technical Evaluation

Following the guidance in DSRS Section 3.7.3, Revision 0, the staff reviewed FSAR Section 3.7.3. The staff also reviewed other FSAR sections when they are referenced. If the staff identified no significant issues in those referenced FSAR sections to affect the staff's safety findings for FSAR Section 3.7.3, the SER sections that evaluate those FSAR sections are referenced. The areas of technical evaluation include: seismic analysis methods, determination of the number of earthquake cycles, procedure used for analytical modeling, basis for selection of frequencies, analysis procedure for damping, three components of design ground motion, combination of modal responses, interaction of non-Seismic Category I subsystems with Seismic Category I SSCs, multiply supported equipment and components with distinct inputs, use of equivalent vertical static factors, torsional effects of eccentric masses, and Seismic Category I buried piping, conduits, and tunnels.

3.7.3.4.1 Seismic Analysis Methods

FSAR Section 3.7.3.1, "Seismic Analysis Methods," indicates that the NuScale seismic subsystems are generally analyzed using the response spectrum analysis method or the equivalent static analysis method. The applicant indicated that the NPMs are evaluated using time history analysis method as described in FSAR Appendix 3A, "Dynamic Simulation of the NuScale Power Module."

FSAR Section 3.7.3.1.1, "Response Spectrum Analysis Method," indicates that the response spectrum analysis method is used to determine seismic response parameters for an SSC based on the ISRS developed from the seismic analysis of the buildings as discussed in FSAR Section 3.7.2. The modal response for each mode of the SSC is determined by accelerating each mode with the spectral acceleration corresponding to the frequency of that mode. The representative maximum response of interest for design is then obtained by combining the corresponding maximum individual modal responses.

FSAR Section 3.7.3.1.2, "Equivalent Static Load Method," indicates that the equivalent static method is used for the analysis of simple SSCs if dynamic analysis is not warranted. The equivalent static load is the product of the mass of the SSC times the constant static factor of 1.5 times the peak spectral acceleration of the applicable in-structure response spectra.

The staff finds the applicant's approaches to the response spectrum analysis method and equivalent static load method for seismic subsystems analysis are acceptable because they are consistent with common industry practices and in conformance with the acceptance criteria in DSRS Section 3.7.3.II.1. The staff evaluation of the time-history seismic analysis method for the NPMs is provided in SER Section 3.9.2.

3.7.3.4.2 Determination of the Number of Earthquake Cycles

FSAR Section 3.7.3.2, "Determination of Number of Earthquake Cycles," indicates that the fatigue analysis of seismic subsystems, components, and equipment considers two SSE events with 10 maximum stress cycles (20 full cycles of maximum SSE stress range in total). It also allows an alternative method in which the number of fractional vibratory cycles equivalent to 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to Institute of Electrical and Electronics Engineers (IEEE) Standard 344-2013, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Plants."

The staff finds that the FSAR specification of these two methods is consistent with the acceptance criteria in DSRS Section 3.7.3.II.2, for the case in which the OBE is defined as less than or equal to one-third of the SSE. The OBE for the NuScale standard design is specified as one-third of the CSDRS, as evaluated in SER Section 3.7.1. Therefore, the staff finds the methods for determining the number of earthquake cycles acceptable. SER Section 3.9.2 provides the staff evaluation of the piping and components related to the number of earthquake cycles.

3.7.3.4.3 Procedure Used for Analytical Modeling

FSAR Section 3.7.3.3, “Procedures Used for Analytical Modeling,” describes criteria used to determine whether a component or structure will be analyzed as a subsystem. This approach is consistent with DSRS Acceptance Criterion 3.7.3.II.3, which directly references DSRS Acceptance Criterion 3.7.2.II.3.

FSAR Section 3.7.3.3 indicates that the RXB weight is 337,000 kips, and a subsystem can be decoupled if its weight is less than 1 percent of the RXB weight, or 3370 kips. The larger subsystems, the NPM and RBC, weigh approximately 2,000 kips and 3000 kips, respectively, and thus could be decoupled. However, the applicant coupled both the NPM and RBC models in the RXB model, which would provide more accurate analysis results for their seismic response. The fuel storage racks are assumed to have a weight of 400 kips each, and each bioshield is less than 230 kips and, therefore, these SSCs are decoupled.

Distribution systems, such as cable trays, piping, heating, ventilation, air conditioning, and individual components will not have significant weight. Hence, these systems satisfy the acceptance criteria in DSRS 3.7.3.II.3 for subsystem decoupling. FSAR Section 3.7.3.3 specifically addresses four subsystems: NPM, Fuel Storage Racks, RBC, and Bioshields. The staff evaluated these subsystems as follows:

<u>NPM</u>

Each NPM is a subsystem. FSAR Appendix 3A summarizes the seismic analysis of the NPMs. The RXB seismic model includes the detailed NPM model. The RXB model is then analyzed for seismic SSI to establish the seismic demands. Results from the RXB analysis include in-structure response time histories and ISRS at each NPM support location and the pool walls and floor surrounding the NPM. These results are then used as the seismic input for the NPM seismic analysis.

<u>Fuel Storage Racks</u>

NuScale deferred the design and evaluation of the fuel storage racks to the applicant that references the NuScale US460 standard design in COL Item 9.1-2. The staff evaluation on the fuel storage racks of the US460 standard design is provided in SER Section 9.1.2.

<u>Reactor Build Crane</u>

A simplified RBC model, consisting of beams, masses, and link elements, is incorporated into the RXB seismic model as discussed in FSAR Section 3.7.2. Detailed analysis and design of the RBC is discussed in FSAR Section 9.1.5 and is evaluated by the staff in SER Section 9.1.5.

<u>Bioshields</u>

In FSAR Section 3.7.3.3.1, “Bioshields,” the applicant described the analysis and design of the bioshields. The bioshields are classified as non-safety related, not risk-significant, Seismic Category II components that provide an additional radiological barrier to reduce dose rates in the RXB and support personnel access. Bioshields are removed while an NPM is being detached and refueled.

The bioshield has horizontal and vertical components that are two separate pieces. The horizontal component is a 24 inch-thick reinforced-concrete slab encased by a stainless-steel liner. The horizontal bioshield is attached to the bay walls using square-tube post with bolting brackets. This feature allows for the horizontal bioshield to be temporarily removed and placed on top of the adjacent bioshield for refueling and maintenance. The vertical component is a square stainless-steel tube-framing system with radiation paneling consisting of borated high-density polyethylene (HDPE). The vertical bioshield is supported at the top and is also constrained to the end of the pool bay walls by seismic restraints that resist horizontal motion. The vertical bioshield is removed and stored during maintenance and refueling.

The bioshield is a non-Seismic Category I component and it must not fail or impair the integrity of nearby Seismic Category I SSCs due to adverse seismic interactions. The applicant states that the bioshield is analyzed and designed to prevent its failure under the SSE. The applicant developed ISRS with 4 percent damping at the top of the bay walls for the design of the bioshield. FSAR Figure 3.7.3-5 shows the enveloped ISRS based on the RXB seismic analysis cases with the CSDRS input. The applicant provided COL Item 3.7-10 that requires an applicant that references the NuScale US460 standard design to demonstrate that the bioshield components and connections can withstand the bioshield loads and appropriate load factors.

The staff reviewed the applicant's description of the bioshield design and evaluation provided in FSAR Section 3.7.3.3.1 and found them acceptable because (1) the bioshield is designed such that it does not fail and impair the integrity of nearby Seismic Category I SSCs under the SSE, thus meeting the acceptance criteria in DSRS Section 3.7.3.II.8, (2) the applicant used enveloped ISRS as the input for bioshield design analysis that are conservatively determined based on the RXB seismic analysis cases, and (3) the applicant provided a COL item that ensures the bioshield withstands the plant-specific bioshield loads.

3.7.3.4.4 Basis for Selection of Frequencies

In FSAR Section 3.7.3.4, "Basis for Selection of Frequencies," the applicant indicated that, in order to avoid resonance, components are designed so that the fundamental frequencies of the components are either less than one-half or more than twice the dominant frequencies of the support structure. The applicant also indicated that the equipment is tested or analyzed to demonstrate that it is adequate in consideration of the fundamental frequencies of the equipment and support structure. The staff finds the applicant's basis for the selection of frequencies acceptable because it is consistent with the acceptance criteria in DSRS Section 3.7.3.II.4.

3.7.3.4.5 Analysis Procedure for Damping

FSAR Section 3.7.3.5, "Analysis Procedures for Damping," indicates that the analysis procedure used to account for the damping in subsystems is consistent with FSAR Section 3.7.1.2, "Percentage of Critical Damping Values" and FSAR Section 3.7.2.15, "Analysis Procedure for Damping," for seismic systems. The staff finds this approach acceptable because it is consistent with the acceptance criteria in DSRS Section 3.7.3.II.5. The staff evaluated FSAR Section 3.7.1.2 and Section 3.7.2.15 in SER Section 3.7.1.4 and Section 3.7.2.4, respectively. The staff evaluated component modal damping of the piping systems in SER Section 3.12.

3.7.3.4.6 Three Components of Design Ground Motion

In FSAR Section 3.7.3.6, “Three Components of Earthquake Motion,” the applicant indicated that seismic responses resulting from the analysis of subsystems in response to three components of the earthquake ground motion are combined in the same manner as the seismic response resulting from the analysis of building structures, as specified in FSAR Section 3.7.2.6. The staff finds this approach acceptable because it is consistent with DSRS Acceptance Criterion 3.7.3.II.6, which directly references DSRS Acceptance Criterion 3.7.2.II.6. The staff evaluated FSAR Section 3.7.2.6 in Section 3.7.2 of this report.

3.7.3.4.7 Combination of Modal Response

In FSAR Section 3.7.3.7, “Combination of Modal Responses,” the applicant indicated that in response to the spectrum analysis of subsystems, the square root of the sum of the squares (SRSS) method is used to combine the modal responses when the modal frequencies are well separated; otherwise, the modal responses are combined in accordance with guidance in RG 1.92, Revision 3, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.” The staff finds that the approach is acceptable because it is consistent with DSRS Acceptance Criterion 3.7.3.II.7 and follows the NRC guidance in RG 1.92, Revision 3.

3.7.3.4.8 Interaction of Non-Seismic Category I Subsystems with Seismic Category I SSCs

In FSAR Section 3.7.3.8, “Interaction of Non-Seismic Category I Subsystems with Seismic Category I Structures, Systems, and Components,” the applicant stated that when non-Seismic Category I subsystems (or portions thereof) could adversely affect Seismic Category I SSCs, the subsystems are categorized as Seismic Category II and analyzed following the methodology discussed in FSAR Section 3.7.3.1. The staff finds this approach acceptable because it is consistent with the acceptance criteria in DSRS Section 3.7.3.II.8.

The applicant also stated that for non-Seismic Category I subsystems attached to Seismic Category I SSCs, the modeling of the Seismic Category I SSCs includes the dynamic effects of the non-Seismic Category I subsystems. The attached non-Seismic Category I subsystems, up to the first anchor beyond the interface, are designed so that the CSDRS does not cause any failure in the Seismic Category I SSCs. As defined in FSAR Section 3.7.1, for the NuScale US460 standard design, the CSDRS consists of two sets of spectra, identified as CSDRS and CSDRS-HF. The staff finds this approach acceptable because the applicant’s approach meets the acceptance criteria in DSRS Section 3.7.3.II.8.

3.7.3.4.9 Multiple-Supported Equipment and Components with Distinct Inputs

In FSAR Section 3.7.3.9, “Multiple-Supported Equipment and Components with Distinct Input,” the applicant indicated that both the uniform support motion (USM) method and the independent support motion (ISM) method are used to address multiply supported equipment and components.

The applicant explained that equipment and components may be supported at several points by either a single structure or separate structures and motions of the primary structure at each of the support points may be different. A suitable approach for analyzing equipment supported at two or more locations is to define a uniform response spectrum that envelopes individual response spectra at support locations. The uniform response spectrum is applied at all locations to calculate the maximum inertial responses of the equipment, which is referred to as the USM method. In the ISM method, structural support points that are attached to a rigid floor or structure are considered as one group of supports. After the individual group responses are determined for each input direction, they are combined by the absolute sum method. For the ISM method, the applicant followed the guidance in NUREG-1061, Volume 4, “Evaluation of Other Loads and Load Combinations,” dated December 1984.

The staff finds the applicant’s approaches for handling the multiply supported equipment and components using the USM method and ISM method acceptable because both methods are endorsed as acceptable in DSRS Section 3.7.3.II.9.

3.7.3.4.10 Use of Equivalent Vertical Static Factors

In FSAR Section 3.7.3.10, “Use of Equivalent Vertical Static Factors,” the applicant stated that the equivalent vertical static factors are not used in the design of the Seismic Category I and II structures. The applicant further stated that the vertical seismic loads are generated from the SSI analysis. Since the applicant did not use equivalent vertical static factors, no further technical evaluation of this area is needed.

3.7.3.4.11 Torsional Effect of Eccentric Masses

In FSAR Section 3.7.3.11, “Torsional Effects of Eccentric Masses,” the applicant stated that the subsystem analysis includes the torsional effect of significant eccentric masses connected to the subsystem. For a rigid component with natural frequency greater than 50 Hz, the lumped mass is modeled at the center of gravity of the component with a rigid link to the appropriate point in the subsystem. Also, for flexible components, the subsystem model is expanded to include an appropriate model of the component. The staff finds the applicant’s approach for torsional effect of eccentric masses acceptable because it is consistent with the acceptance criteria in DSRS Section 3.7.3.II.11.

3.7.3.4.12 Seismic Category I Buried Piping, Conduits, and Tunnels

In FSAR Section 3.7.3.12, “Buried Seismic Category I Piping, Conduits, and Tunnels,” the applicant explained that there is a Seismic Category I underground reinforced-concrete duct bank that contains conduits connecting the RXB and CRB. The applicant stated that the reinforced concrete design of the duct bank and applicable load combinations are based on ACI 349-13. The applicant further stated that the duct bank seismic analysis under the safe-shutdown earthquake was performed in accordance with ASCE 4-16. The staff finds the applicant’s seismic analysis and structural design for the Seismic Category I duct bank is acceptable because they are performed in accordance with acceptable industry standards and meet the acceptance criteria in DSRS Section 3.7.3.II.12.

3.7.3.4.13 Methods for Seismic Analysis of Seismic Category I Concrete Dams

The applicant stated that the NuScale US460 standard design does not include or require the presence of a dam. Therefore, no further technical evaluation of this area is required.

3.7.3.4.14 Methods for Seismic Analysis of Aboveground Tanks

The applicant stated that the NuScale US460 standard design does not include Seismic Category I aboveground tanks. Therefore, no further technical evaluation of this area is required.

3.7.3.5 Combined License Information Items

Table 3.7.3-1 lists the COL information item numbers and descriptions related to seismic subsystem analysis from FSAR Table 1.8-1.

Table 3.7.3-1: NuScale COL Information Item for FSAR Section 3.7.3

Item No.	Description	FSAR Section
COL Item 3.7-10	An applicant that references the NuScale Power Plant US460 standard design will determine the means and methods of lifting the bioshield. An applicant will demonstrate that bioshield components and connections can withstand the bioshield loads and appropriate load factors.	3.7.3

3.7.3.6 Conclusion

The staff finds that the applicant has adequately addressed seismic system analysis in accordance with the acceptance criteria set forth in DSRS Section 3.7.3, and on this basis, the staff concludes that the regulatory requirements delineated in Section 3.7.3.3 of this report are satisfied.

3.7.4 Seismic Instrumentation

3.7.4.1 Introduction

This SER section presents the instrumentation system for measuring the effects of an earthquake. 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” requires a timely shutdown of a nuclear power plant if vibratory ground motion exceeding that of the OBE occurs or if significant plant damage occurs. To achieve this goal, seismic instrumentation should be installed in the free field and within Seismic Category I structures to measure effects of an earthquake. The data from the nuclear power plant’s free-field seismic instrumentation, coupled with information obtained from a plant walkdown, are used to make the initial determination of whether the plant must be shut down.

NuScale SDAA FSAR Section 3.7.4 presents the instrumentation system for measuring the effects of an earthquake.

3.7.4.2 Summary of Application

SDAA Part 2 (FSAR): FSAR Section 3.7.4.1 states that the NuScale design requires a deviation from the guidance in RG 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," in that seismic instrumentation is not installed inside containment because the containments (of six power modules) are flooded as part of the refueling process. Instead of locating seismic instrumentation inside containment, instrumentation will be located in the RXB.

FSAR Section 3.7.4.2 describes the exact sensor locations to ensure the site, RXB and CRB are adequately instrumented for a seismic event:

- One free-field strong motion accelerator (FFSMA) is a downhole instrument located at the foundation level as close as directly below the free-field ground surface FFSMA as practical.

- One strong-motion accelerometer located in the RXB on the basemat in the northwest boric acid storage room.

- One SMA located in the RXB on the basemat in the northeast vestibule room.

- One SMA located in the RXB in the northwest utilities area room.

- One SMA located on the RXB roof.

- One SMA located in the CRB on the basemat in the northeast corridor room.

- One SMA located in the CRB in the MCR.

FSAR Section 3.7.4.3 states that the SMS provides Seismic Category I annunciation in the MCR. Separately, the SMS provides information to the MCR via the plant control system (PCS).

FSAR Section 3.7.4.4 provides comparison with guidance and states that conformance with RG 1.166 is site-specific.

FSAR Section 3.7.4.5 states that the SMS is expected to be operable during all modes of plant operation, including periods of plant shutdown.

FSAR Section 3.7.4.6 specifies that SMS program implementation will be discussed during COL application (COL Item 3.7-11).

ITAAC: There are no ITAAC associated with this area of review.

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

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

3.7.4.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” requires seismic instrumentation. Suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.

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

- RG 1.12, Revision 3, “Nuclear Power Plant Instrumentation for Earthquakes”

- RG 1.166, Revision 1 “Pre-Earthquake Planning, Shutdown and Restart of a Nuclear Power Plant Following an Earthquake”

3.7.4.4 Technical Evaluation

When an earthquake occurs ground motion data are recorded by seismic monitoring system (SMS). SMS includes the sensors, wiring between the sensors, the control cabinet, and the instrumentation in the control cabinet. The controller processes the data and provides alarm notification to the MCR via the PCS. Because the PCS is not a Seismic Category I system, additional Seismic Category I annunciation equipment is located in the MCR to alert operators of a seismic event. This annunciation is part of the SMS.

Seismic sensors will be located in the free-field, RXB and the CRBs at locations that have been modeled as mass points in the building dynamic analysis so that the measured motion can be directly compared with the design spectra.

The staff reviewed the SDAA and evaluated the completeness and adequacy of technical requirements to the placement and operability of seismic monitoring system, and finds it satisfactory.

3.7.4.5 Combined License Information Items

SER Table 3.7.4-1 lists COL information item numbers and descriptions related to seismic instrumentation from FSAR Table 1.8-1.

Table 3.7.4-1: NuScale COL Information Item for FSAR Section 3.7.4

Item No.	Description	FSAR Section
COL Item 3.7-11	A COL applicant that references the NuScale Power Plant US460 standard design will prepare site-specific procedures for seismic instrumentation maintenance and post-earthquake activities. Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operations and shutdown. The procedures for post-earthquake activities must provide sufficient information to determine if the level of earthquake ground motion requiring shutdown has been exceeded and appropriate corrective actions to be taken if needed.	3.7.4.6

3.7.4.6 Conclusion

Based on the review of the FSAR Section 3.7.4, the staff finds that the applicant provided complete and adequate technical requirements for the placement and operability of SMS suitable to record seismic response of nuclear power plant features important to safety after an earthquake. The staff, therefore, concludes that the seismic instrumentation proposed by the applicant, as supported by COL Item 3.7-11, complies with the requirements of 10 CFR Part 50, Appendix S.

3.8 Design of Category I Structures

The NuScale Power Plant US460 standard design includes two Seismic Category I structures, portions of the CRB and portions of the RXB. Section 1.2, “General Plant Description,” a general description of the US460 standard design. FSAR Figure 1.2-1 presents the layout of a typical NuScale US460 Power Plant.

3.8.1 Concrete Containment

The NuScale Power Plant US460 standard design does not use a concrete containment. Therefore, this section does not apply to the NuScale Power Plant US460 standard design because the NuScale design uses a steel containment.

3.8.2 Steel Containment

3.8.2.1 Introduction

In Section 3.8.2.1, “Description of Containment,” of the FSAR, the applicant describes the CNV as an integral part of the NPM located in the RXB. As shown in FSAR Figure 6.2-1, “Containment System,” of the FSAR, the CNV houses the reactor pressure vessel (RPV), reactor coolant system (RCS), and associated SSC. The CNV support skirt rests at the top of RXB foundation and upper CNV is laterally supported by three (3) support lugs. Further, the NPM, and thus the CNV is partially immersed in the reactor pool water to enable decay heat removal during postulated design-basis events.

3.8.2.2 Summary of Application

The applicant describes the CNV as a metal containment, Subsection NE, Class MC pressure vessel of the American Society of Mechanical Engineers (ASME) Code, Section III, that undergoes design, analysis, fabrication, inspection, testing, and stamping as an ASME Subsection NB, Class 1 pressure vessel.

The summary of FSAR Section 3.8.2 is provided below:

In FSAR Section 3.8.2.1, “Description of Containment,” the applicant provides the physical description and the primary functions of the CNV. That includes the CNV configuration descriptions, supports, access/manways, penetrations (piping, electrical, emergency core cooling, etc.), the welded attachments (lateral and vertical support to the RPV, the CNV-RPV support ledge, the RPV-CNV support ledge, etc.), horizontal and vertical shims for fit-up purposes.

In FSAR Section 3.8.2.2, “Applicable Codes, Standards, and Specifications,” the applicant describes the codes, standards, and specifications meeting acceptance criteria in Design Specific Review Standard (DSRS); specifically, compliance to the CNV structure and skirt support to the requirements of ASME Code.

In FSAR Section 3.8.2.3, “Loads and Load Combinations,” the applicant describes that the CNV pressure retaining components’ stresses and fatigue evaluations are performed in accordance with the Subsection NB of ASME Code, Section III. However, the applicant considers that the load combinations of RPV are also applicable to the CNV since the characteristics of the fabrication, inspection, and testing requirements of RPV and CNV structures are comparable by meeting requirements of Class 1 vessel in Subsection NB and NF of ASME Code, Section III, respectively.

In FSAR Section 3.8.2.4, “Design and Analysis Procedures,” the applicant describes that the CNV and support designs and analyses conform to the requirements of Subsection NB of ASME Code, Section III. The applicant uses the combinations of standard textbook hand calculations for simple structures and ANSYS general purpose finite element program to determine stress in CNV. The applicant evaluates for buckling, or elastic instability that results in collapse, as part of the limit load analysis, by using ASME Code Case N-759-2 (e.g., buckling of the torispherical lower head).

In FSAR Section 3.8.2.5, “Structural Acceptance Criteria,” the applicant describes the structural behavior of the CNV complies with the Subsection NB of ASME Code, Section III and RG 1.57

In FSAR Section 3.8.2.6, “Materials, Quality Control, and Special Construction Techniques,” the applicant provides the CNV engineered safety feature components list and for their material specifications in Tables 6.1-1 and 6.1-2 in the FSAR.

In FSAR Section 3.8.2.7, “Testing and Inservice Inspection Requirements,” the applicant describes the examinations, testing and inservice inspection (ISI) requirements with respect to compliance with the ASME Code for fabrication and preservice examinations of the CNV and the other components relied on for containment integrity. The applicant identifies the requirements in ASME Code, Section III, Subsection NB and Section XI, using examination methods in ASME Code, Section V. The applicant describes the preoperational and periodic design pressure leakage test in Section 6.2, “Containment Systems,” of the FSAR. The applicant also provided TR-123952-NP, Revision 1, referenced in Section 6.2.8 of the FSAR, describing the requirements of the Containment Leakage Integrity Program (CLIP) where the leakage integrity of CNV will be assured by local leak rate testing (Type B and Type C) per the requirements of 10 CFR Part 50, Appendix J. In this technical report, the applicant describes the exemption from the Type A integrated leak rate testing of 10 CFR Part 50, Appendix J due to significant challenges that render the test either invalid or infeasible.

The testing, inspection and design criteria provide sufficient leakage integrity assurance for the CNV.

ITAAC: ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: The applicant provided the TS associated with Section 3.8.2, in Section 6.2, “Containment Systems.”

Technical Reports: The following NuScale TRs apply to the CNV:

- TR-123952, Revision 1, "NuScale Containment Leakage Integrity Assurance,"
- TR-121516, Revision 1, "Containment Vessel Ultimate Pressure Integrity,"
- TR-121517, Revision 1, "NuScale Power Module Short-Term Transient Analysis."

3.8.2.3 Regulatory Basis

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

- General Design Criterion (GDC) 1 - The CNV is subject to the design, manufacturing, and operating quality assurance requirements in the Quality Assurance Program Description.

- GDC 2 - Seismic design to withstand the effects of a safe shutdown earthquake (SSE) regarding the CNV is met by using the guidance provided in Regulatory Guide (RG) 1.29, "Seismic Design Classification for Nuclear Power Plants," Revision 5.

- GDC 4 -The CNV is designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

- GDC 16 - The CNV is designed to provide a leak-tight barrier and to contain the CNV design pressure during design-basis events.

- GDC 50 - The CNV is designed to ensure the component, access openings, penetrations, and containment heat removal systems have the capability to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA.

- GDC 53 - The CNV is designed with provisions to permit inspection and testing for periodic verification that the CNV remains within the limits defined by the design-basis.

- 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," as it relates to the capability of the containment to resist those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding.

- 10 CFR 50.55a requires that (1) SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, (2) containments, systems, and components of nuclear power reactors meet the requirements of the ASME Code, and (3) RGs 1.84 and 1.147 provide guidance related to NRC-approved ASME Code cases that may be applied to the design, fabrication, erection, construction, testing, and inspection of containments, systems, and components. Compliance with 10 CFR 50.55a also requires that examination of steel containments be performed in accordance with the requirements of Subsection IWE of the ASME Code, Section XI.

DSRS Section 3.8.2 lists the acceptance criteria to meet the above requirements, as well as 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 1.7, Revision 3, "Control of Combustible Gas Concentrations in Containment," issued March 2007.

- RG 1.57, Revision 2, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," issued May 2013.

- RG 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007.

- RG 1.216, Revision 0, “Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure,” issued August 2010.

3.8.2.4 Technical Evaluation

The staff reviewed Section 3.8.2 and Chapter 6 of the FSAR against the NRC regulatory guidance to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function. DSRS Section 3.8.2 identifies seven specific DSRS acceptance criteria to meet the relevant requirements of the NRC’s regulations listed in DSRS Section 3.8.2.II and included in SER Section 3.8.2.3 above.

DSRS Section 3.8.2 provides guidelines for the staff to use in reviewing the technical areas related to the design of the steel portion of the containment that is not backed by concrete, based on 10 CFR 50.55a; GDC 1, 2, 4, 16, and 50; 10 CFR Part 50, Appendix B; and 10 CFR 50.44. The staff used the guidance in DSRS Section 3.8.2 to review Section 3.8.2 of the FSAR. A summary of the application is discussed in SER Section 3.8.2.2 above.

DSRS Section 3.8.2 identifies seven specific acceptance criteria to meet the relevant requirements of the NRC’s regulations listed in DSRS Section 3.8.2.II, in particular, the review focused on (1) a description of the containment, (2) applicable codes, standards, and specifications, (3) loads and load combinations, (4) design and analysis procedures, (5) structural acceptance criteria, (6) materials, quality control, and special construction techniques, and (7) testing and inservice surveillance programs.

3.8.2.4.1 Description of Steel Containment

The applicant identifies the CNV as a Class MC, constructed and stamped as Class 1, pressure vessel that complies with the requirements of Section III of the ASME Code. As addressed in SER Section 3.8.2.3, the applicant describes that the design of the CNV complies with the regulatory requirements of GDC 1, 2, 4, 16, 50 and 53 in 10 CFR Part 50, Appendix A. The applicant describes that the boundaries of jurisdiction for the CNV are per the requirements in Subarticle NE-1130 of the ASME Code, Section III. The applicant defines the nonstructural attachments as non-pressure retaining and do not contribute to support of the CNV. However, all of the structural and nonstructural attachments at the surface of the CNV shell and the welds between the attachments and the CNV are considered part of the vessel.

The applicant described that the CNV is a shop-fabricated vessel with corrosion-resistant stainless-steel materials. The staff confirmed by reviewing document EC-124581, Revision A, “ASME Code Evaluation of the Lower CNV,” and EQ-146988, Revision 0, “ASME Design Specification for CNV,” that the pressure boundary materials for the upper portion of CNV are SA-336, F6NM, the bottom portion of CNV are F6NM and SA-965, Grade FXM-19, and the materials for the support skirt is SA-182, Grade F304. The CNV flange of the upper and lower CNV assemblies uses the same seal design (double seal and test port arrangement) as the RPV.

In Subsection 3.8.2.1 of the FSAR, and in Section 2.2, “General Description,” of EQ-146988, Revision 0, “ASME Design Specification of CNV,” the applicant describes the primary functions of the CNV as follows:

1. The ultimate barrier against uncontrolled release of radioactivity and radiological contaminants to the environment.

2. Passive heat transfer from coolant inventory inside the CNV through the CNV wall to the UHS during emergency core cooling system (ECCS) operation. Additionally, the CNV supports emergency core cooling by passive retention of coolant inventory during ECCS operation.

3. Nozzles and penetrations to allow transmittal of signals from SSC inside the CNV.

4. Nozzles and penetrations to allow for flow into and out of the CNV.

5. Access ports entryway into the CNV and access ports for potential maintenance of components within the CNV, such as CRDM, steam generator (SG), main steam, and pressurizer heater.

6. Structural support to SSC located inside or attached to the CNV.

In FSAR Section 3.8.2.1, the applicant describes the external boundary condition of the CNV that includes welded lateral three lug restraints located on the upper shell of the CNV, as shown in Figures 3B-36 “Plan View Layout of NPM Lug Restraint Configurations,” and 3B-37 “Plan View of Typical NuScale Power Module Bay with Lug Restraints,” in the FSAR, and integrally welded skirt support to the bottom head of CNV with four built-up stainless-steel passive seismic supports bracing from the skirt to the bay walls, as shown in Figures 3B-33 “Plan View of Lower NPM Bay with Skirt Restraint ,” and 3B-34 “NuScale Power Module Section View at Skirt Support.” **{{**

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In FSAR Section 3.8.2.1, the applicant describes that the upper and lower decay heat removal system (DHRS) condenser supports are welded to the CNV shell. The applicant states that there are four SG access ports, two pressurizer access ports, and two manways on the CNV shell.

In FSAR Table 6.2-4, the applicant lists the containment penetrations and states which penetrations are used for each process system fluids or gases. There are 45 penetrations in the CNV. Figures 6.2-2a, “Containment Vessel Assembly,” and 6.2-2b, “Containment Vessel

Assembly," in the FSAR show the CNV top head and side penetrations and listing the penetrations with nozzle numbers. The applicant describes types of penetrations as follows:

- Fluid system penetrations.
- Instrument seal assemblies and electrical penetration assemblies.
- ECCS valve actuator assembly penetrations.
- Access port penetrations.

During the audit, the staff requested engineering documents for review to enhance understanding of the engineering methodologies and analysis details used in qualifying the CNV structure against the applicable codes and standards. The staff noted that some of the documents presented during the audit are in preliminary stages (e.g., EC-140852, Revision B, "ASME Code Evaluation of CNV Head Feedwater Nozzle Region," EQ-105619, Revision 0, "ASME Design Specification for Containment Vessel and Top Support Structure") were updated with editorial changes. The staff also noted that the engineering document, EC-124581, Revision A, "ASME Code Evaluation of the Lower CNV," will be further revised to provide additional information without the impact on the current analysis. The staff performed reviews of these documents to better understand the methodologies and results utilized in the design of the steel containment, as well as to inform the staff's safety findings. The staff finds that these engineering supporting documents provide sufficient information of engineering methodologies and analysis details used in qualifying the CNV structure. In addition, the staff finds that the applicant provided sufficient information describing the CNV in the FSAR, and that the FSAR description complies with the acceptance criteria identified in DSRS 3.8.2.II.1.

3.8.2.4.2 Applicable Design Codes, Standards, and Specifications

The staff reviewed the codes, standards, and specifications in Sections 3.8.2.2 and 3.8.2.4 of the FSAR, against the list in DSRS Section 3.8.2.II.2. The applicant describes that the CNV ASME, Class MC component is designed, constructed, and stamped as an ASME Code Class 1 vessel in accordance with ASME Code, Section III, Subsection NB, except that overpressure protection is in accordance with Article NE-7000 instead of Article NB-7000 of the ASME Code, Section III. The staff also audited the design specifications of the CNV, CNV support, and TSS provided in document EQ-105619, Revision 0, "ASME Design Specification for Containment Vessel and top Support Structures."

The applicant classifies the CNV support skirt and lugs as an ASME, Section III, Class MC support, constructed to Class 1, and conforms to the requirement of ASME Section III, Subsection NF.

In FSAR Chapter 6, Section B, the applicant also describes that the fabrication requirements imposed on the construction of the CNV are consistent with requirement of 10 CFR 50.55a.

The staff finds the requirements of CNV design codes, standards, and specifications are acceptable because they comply with the acceptance criteria identified in DSRS 3.8.2.II.2, and the staff finds that this satisfies the applicable requirements of 10 CFR 50.55a and the criterion of GDC 1.

3.8.2.4.3 Loading Criteria, Including Loads and Load Combinations

In FSAR Section 3.8.2.3, the applicant describes that the stresses and fatigue for the CNV pressure retaining components were evaluated in accordance with Subsection NB of the ASME Code, Section III. The applicant also lists all the design loads for the CNV with detailed descriptions. The applicant provides the ASME Code Design, Service Level (Level A, Level B, Level C, Level D) load combinations in Tables 3.8.2-2, 3.8.2-3, and 3.8.4-5, of the FSAR for the pressure retaining items, Class 1 supports, and Class 2 supports for CNV, CNV bolts, CNV bolted connections, and supports, respectively. The applicant describes that the load combinations for the CNV design were performed to the requirements of ASME, Section III, Subsection NB. The applicant describes that the load combinations for the Class 1 RPV are more applicable than using the load combinations in RG 1.57 due to the increased quality of the design, fabrication, inspection, and testing required by ASME Section III, Subsection NB for a Class 1 vessel. Furthermore, the applicant concludes that the load combinations in RG 1.57 are intended for structures designed, fabricated, inspected, and tested to ASME Section III, Subsection NE requirements.

The applicant also provided justification by comparing the requirements of the ASME Code for both an NB, Class 1 vessel and an NE, Class MC vessel, and the staff summarized two examples as follows: (1) the welds in an NB, Class 1 vessel, are required to have volumetric and either liquid penetrant or magnetic particle inspections performed per ASME Code, Section III, Subarticle NB-5200; however, the welds in an NE, Class MC vessel, are only required to have a fully radiographed inspection per Subarticle NE-5200. (2) The ASME, Section III, Article NB-6000 hydrostatic test pressure is approximately 14 percent greater than the requirement in Article NE-6000, because Paragraph NB-6221 specifies a minimum hydrostatic test pressure of 1.25 times the design pressure while Paragraph NE-6221 specifies a minimum hydrostatic test pressure of 1.1 times the design pressure.

The staff agrees that the vessel load combinations and allowable limits differ from containment structures because design, fabrication, inspection and testing requirements for vessels are more restrictive, which allows higher design limits in ASME Section III, Subsection NB.

GDC 2 requires that SSCs important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena, including earthquakes.

The staff reviewed the structural modeling, input motion, major assumptions, acceptance criteria, fluid structural interaction considerations, mass distribution, damping values, dominant frequency and mode shape plots, and gap/impact modeling and finds that the analysis was performed in accordance with the guidance in SRP Section 3.9.2.

FSAR Section 3.8.2.3, states that the load combinations meet the requirements of NCA-2141(b) in the ASME Code, Section III, and considers the guidance in RG 1.57. The staff reviewed the load combinations given in Tables 3.8.2-2 through 3.8.2-5 of the FSAR and finds that the loads and load combinations comply with those identified in the Acceptance Criteria of DSRS Section 3.8.2.II.3. The staff also finds that this satisfies the criteria in GDCs 2, 4, and 16.

3.8.2.4.4 Design and Analysis Procedures

In FSAR Sections 3.8.2.4 and 6.2.1.1.2, the applicant provides an overview of the design and analysis requirements for the CNV and CNV supports. In FSAR Sections 3.8.2.1.2 and

3.8.2.2.2, the applicant describes that the CNV is an ASME, Class MC (steel) containment whose design, analysis, fabrication, inspection, testing, and stamping conform to the ASME Code, Class 1 pressure vessel requirements in accordance with Section III, Subsection NB as permitted by NCA-2134(c). In FSAR Table 3.8.2-1 “Design and Operating Parameters,” and Table 6.2-1, “Containment Design and Operating Parameters,” of the FSAR, the applicant provides the design and operation parameters of internal/external pressure and temperatures. Further, in FSAR Section 3.8.2.4, the applicant described that the CNV design and analyses (stress and fatigue usage) conform to the requirements of ASME Code, Section III, Subarticle NB-3200, the CNV support design and analysis conform to the requirements of Subarticle NF-3200, the fabrication conforms to the requirements of Article NB-4000 and Article NF-4000, and the nondestructive examination of pressure retaining and integrally attached materials meet the requirements of Article NB-5000 and Article NF-5000. The applicant also describes that the overpressure protection is performed in accordance with Article NE-7000 in ASME, Section III.

In FSAR Sections 3.8.2.4.6 and 6.2.7, the applicant describes the radiation effects in the CNV and concludes that the lower CNV shell is made of austenitic stainless steel which is resistant to neutron embrittlement.

During the audit, the staff requested numerous documents that provide the bases of construction of the CNV per the requirements of Section III of the ASME Code. The staff reviewed EQ-146988, Revision 0,” ASME Design Specification for CNV,” to confirm that the methodologies presented are consistent with the DSRS Acceptance Criterion of 3.8.2. II. This document provides the ASME design specifications requirement for the construction of the CNV. Section 2.6, Table 2-1 of the document provides the ASME classifications of piping penetrations, Section 2.7 of the document provides the ASME Code jurisdictional boundaries of the components of CNV based on the classification of the components that includes the CVN penetrations, welded supports, CNV support skirt, CNV shipping lug and top support structure attachments. Section 3.0, Table 3-1 of the document provides design parameters and values on the CNV. Section 4.0, Tables 4-1 and 4-2 in the document, provide load combinations for ASME stress analysis of CNV and for Class 1 supports, respectively. In Section 3.1 of the document, the applicant describes that the design life of CNV shell is 60 years.

The staff finds the requirements of the CNV design acceptable because the structural acceptance criteria comply with those identified in DSRS Acceptance Criterion of 3.8.2.II.4, in that the total stresses and loads are defined in accordance with Section III of the ASME Code.

3.8.2.4.5 Analysis Procedures

The applicant describes that the detailed analyses of ASME Code primary stresses for the CNV use a combination of standard textbook hand calculations for simple structures, such as nozzles, and ANSYS general purpose finite element program for more complex geometry, such as the CNV top head. Other ASME Code evaluations were performed using ANSYS. Buckling of the torispherical lower head was evaluated using ASME Code Case N-759-2. The staff notes that limit analyses to determine lower bound limit buckling loads may be employed in lieu of Code Case N-759-2. Evaluation of buckling, or elastic instability that results in collapse, is considered as part of the limit load analysis.

The applicant performed stress analyses using the load combinations defined in Section 3.8.2.3 and the allowable limits in accordance with ASME Code, Section III, Subarticles NB-3200 and NF-3200 for the CNV and CNV support, respectively. The allowable limits are based on the mean metal temperature for the applicable Service Level or a conservative higher temperature (i.e., design temperature).

The applicant describes that the computer code verification, validation, configuration control, error reporting, and resolution are performed according to the quality assurance requirements of Chapter 17 of the FSAR.

In summary, the staff observed that the analytical calculations of CNV are in various stages of completion (some calculations were final, and some were not final calculations) and determined that the analytical calculations comply with the structural acceptance criteria identified in DSRS Acceptance Criterion of 3.8.2.II.5 and are in accordance with Section III of the ASME Code. Furthermore, the staff observed that any deficiencies were either identified in the corrective action program or were planned to be addressed when the calculations were to be made final during the ITAAC closure process. The staff finds that this satisfies the criteria of GDC 50.

3.8.2.4.6 Containment Vessel Stress Analysis

In FSAR Section 3.8.2.4.1, "Containment Vessel Stress Analysis," the applicant describes the analytical evaluations of the CNV to maintain the integrity of the pressure retaining function for the loads and load combinations described in FSAR Table 3.8.2-2. The applicant describes that stress and fatigue results are evaluated in accordance with limits provided in Subarticle NB-3200 of the ASME Code, Section III.

The applicant describes that the minimum wall thicknesses for nozzles on the CNV shell, nozzle reinforcement, and limits of reinforcement along the CNV wall and normal to the CNV wall are performed in accordance with Subarticle NB-3300 of ASME Code, Section III. The applicant uses the design by analytical analysis requirements of Subarticle NB-3200 as permitted by Paragraph NB-3331(c) if the requirements of Subarticle NB-3300 are not satisfied.

During the audit, the staff requested numerous documents that provided the basis of construction of the CNV per the requirements of Section III of the ASME Code. The staff reviewed the documents and confirmed that the methodologies and the results presented are consistent with DSRS Acceptance Criterion of 3.8.2.II.4 and 5. The staff summary of its review of selected documents as related to the qualification of the CNV are as follows:

The applicant provides the design specifications of the CNV, CNV support and TSS in EQ-105619, Revision 0, "ASME Design Specification for Containment Vessel and top Support Structures." The applicant classifies the CNV as ASME Section III, Class MC. As permitted by NCA-2134(c), the applicant describes that the CNV is constructed and stamped as an ASME BPVC, Section III, Class 1 vessel in accordance with ASME BPVC, Section III, Subsection NB except that for the overpressure protection shall be in accordance with NE-7000.

The scope of the preliminary document, EC-124581, Revision A, "ASME Code Evaluation of the Lower CNV," is to provide analytical analysis of the lower sections of CNV, namely, lower shell, transition shell, core region shell, lower head of the CNC and the CNV support skirt. **{{**

}}. Based on the review, the staff finds that this preliminary analytical evaluation of the lower sections of CNV is acceptable since the results are within the allowable design limits in Subsections NB-3000, NF-3000 and Appendix XIII of Section III the ASME Code.

The scope of the document EC-140851, Revision B, “ASME Code Evaluation of CNV Head Feedwater Nozzle Region,” is to evaluate the analytical analysis of the CNV upper head containing the feedwater (FW) nozzle. **{{**

}}. Based on its review, the staff finds that this preliminary analytical evaluation of the CNV Head Feedwater Nozzle Region is acceptable since the results are within the allowable design limits in Subsections NB-3000 and Appendix XIII of the ASME Code, Section III.

The scope of the document EC-128779, Revision 0, “CNV Sizing Calculation,” is to determine the minimum pressure thickness of the CNV shell walls, nozzles, safe-ends and bolted cover plates using the design conditions (temperature, pressures, etc.), material properties, geometry of components. **{{**

}}, the staff finds that the design calculations meet the requirements of Article NB-3324.1 of the ASME Code, Section III and are, therefore, acceptable.

The applicant evaluates the CNV top head shell on the effects of loading from the piping and electrical penetrations. The applicant checks buckling of the CNV for the Service Level D seismic event using the requirement of ASME Code Case N-759-2. The applicant also checks buckling on the inside knuckle regions of the top head and bottom head due to internal pressure causing compression in the knuckle regions that using hand calculation based on Equation 4.3-19 from Section VIII of ASME Code, Division 2. The staff finds that the top head shell and bucking checks are consistent with the acceptance criteria of DSRS 3.8.2.II.4.B, and therefore is acceptable.

The analytical evaluations for the CNV were partially available for staff review. Consequently, for its safety review, the staff also used information from the Chapter 3 SER (ML20205L491) for NuScale Design Certification Application (DCA), Revision 5 (ML20225A071). The staff noted that the changes to the geometry of the CNV and their integral components in SDAA are similar to those in the DCA. Therefore, the staff concluded that the external loading conditions on the SDAA CNV would be bounded by those in the DCA due to having smaller mass participation from the SDAA RXB structure and pool water.

The staff also noted that the primary loading conditions on the SDAA CNV are from the internal pressure, penetrations and support attachments. As discussed in Section 3.8.2.4.5 of this report, the applicant conservatively determines the ultimate internal pressure capacity of the CNV for a beyond design-basis LOCA is well over the internal design pressure.

Therefore, based on its review of the available and selected engineering documents, the staff finds that the applicant’s analysis results for the CNV are acceptable because they meet the Subsections NB and NF of Section III of ASME Code and are consistent with the acceptance criteria in DSRS 3.8.2.II.4 and 5.

3.8.2.4.7 Containment Vessel Lateral Support Lugs Analysis

In FSAR Section 3.8.2.4.2, “Containment Vessel Lateral Support Lugs,” the applicant describes the analytical stress evaluations of CNV lateral support lugs performed for the loads and load combinations described in FSAR Section 3.8.2.3 and Table 3.8.2-5. The applicant describes that the lateral support lugs are attached to the CNV upper shell as defined by Paragraph NB-1132.1(a) of Section III of the ASME Code. The lateral support lugs are laterally constrained by the NPM lugs attached to the NPM bay walls. The applicant provides the material specification of CNV support lug as UNS N06690 in Table 5.5.1, “Material for CNV,” of EQ-146988, “ASME Design Specification for Containment Vessel.” The applicant describes that the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subarticle NB-3200 of Section III of the ASME Code.

The applicant provides Figure 6.2-2a, “Containment Vessel Assembly,” in the FSAR showing that the CNV lateral lugs are located near the top of the CNV. The staff also reviewed the following drawings: SL-SPD1-00-M-GA-F010-05101, Revision 0, “Reactor Building General Section B-B,” and SL-SPD1-00-M-GA-F010-30001, Revision 0, “Reactor Building General Arrangement Plan View Elevation 85’-0”,” to gather configurational understanding of the CNV lugs. The staff finds the information provided in FSAR Section 3.8.2.4.2 is acceptable because the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subarticle NB-3200 of Section III of the ASME Code and is consistent with the acceptance criteria in DSRS Section 3.8.2.II.4

3.8.2.4.8 Containment Vessel Lower Support Analysis

In FSAR Section 3.8.2.4.3, “Containment Vessel Lower Support,” the applicant describes the analytical stress evaluations of the CNV skirt support performed for the loads and load combinations described in Section 3.8.2.3 and Table 3.8.2-5 of the FSAR. The applicant describes that the CNV support skirt is integrally welded to the bottom of the CNV lower head and provides vertical restraint, by bearing on the reactor pool floor, and horizontal restraint, by contact with a metal ring called the passive skirt support, which is attached to the reactor pool floor. The applicant classifies the CNV support skirt as an ASME Section III, Class MC Support. However, the applicant describes the design specifications of the CNV support skirt in document EQ-105619, Revision 0, and as permitted by NCA-2134(d), the CNV skirt support will be constructed as an ASME Section III, Class 1 support in accordance with ASME Section III, Subsection NF requirements. The applicant describes that the stress and fatigue evaluations of the CNV support skirt are performed in accordance with Subarticle NF-3200 and the support skirt is constructed in accordance with the requirements of Article NF-4000 of Section III of ASME Code.

FSAR Figure 3B-33 shows the general configuration of CNV lower support located on top of foundation. The staff also reviewed the following general arrangement drawings of SL-SPD1-00-M-GA-F010-05101, Revision 0, “Reactor Building General Section B-B,” and SL-SPD1-00-M-GA-F010-10001, Revision 0, “Reactor Building General Arrangement Plan View Elevation 25’-0”,” to gain configurational understanding of the CNV lower support. The staff finds the information provided in FSAR Section 3.8.2.4.3 acceptable because the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subsection NF-3000 of Section III of the ASME Code and are consistent with the acceptance criteria in DSRS Section 3.8.2.II.4

3.8.2.4.9 Containment Vessel Reactor Pressure Vessel Supports Analysis

In FSAR Section 3.8.2.4.4, “Containment Vessel Reactor Pressure Vessel Supports,” the applicant describes the analytical stress evaluations of the CNV to RPV support for the loads and load combinations described in FSAR Section 3.8.2.3 and FSAR Table 3.8.2-5. In FSAR Table 6.1-2, the applicant provides the material specifications of CNV-RPV support ledge shell lug and RPV support ledge/gussets as SA-168, UNS N06690.

The applicant identifies the CNV-RPV support as Subsection NF, Class 1. The applicant describes that the RPV is laterally and vertically supported to the CNV at four locations by ledges that are integrally welded to the CNV inner and RPV outer shell surfaces. The applicant also describes that vertical lift off is prevented by a threaded pin and collar at the CNV-RPV ledge supports and the seismic connection at the lower RPV-CNV interface is only for lateral support in seismic events.

The applicant describes that the stress and fatigue results of the RPV-CNV support are evaluated in accordance with the limits in Subarticle NB-3200 of Section III of the ASME Code.

The staff finds that the information provided in FSAR Section 3.8.2.4.4 is acceptable because the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subarticle NF-3000 of Section III of the ASME Code and are consistent with the acceptance criteria in DSRS Section 3.8.2.II.4.

3.8.2.4.10 Containment Vessel Ultimate Capacity Analysis

In FSAR Section 3.8.2.4.5, "Containment Vessel Ultimate Capacity," the applicant describes determination of the ultimate pressure capacity of the CNV to the guidance in Appendix A of NUREG/CR-6906 and the failure criteria of the CNV are based on guidance in RG 1.216. The applicant refers to the Technical Report, TR-121516, "CNV Ultimate Pressure Integrity," (ML19158A382) describing the methodology, ultimate pressure, and method of failures for the CNV internal pressure capacity for a beyond design-basis LOCA.

The applicant determination of the ultimate pressure capacity is by meeting one of the following failure criteria:

A. A maximum global membrane strain away from discontinuities of 1.5 percent is reached.

B. Loss of bolt preload occurs at any bolted CNV opening.

C. Buckling occurs in the knuckle of the upper or lower CNV head due to internal pressure.

D. A flange gap that exceeds the calculated allowable values is reached at the outer O-ring of any bolted CNV opening.

E. Solution divergence occurs.

The applicant uses multiple finite element models and analyses to evaluate the bolted connections, shell regions away from concentrations, and buckling of the knuckle regions in the heads. The applicant determines the ultimate internal pressure capacity of the CNV, considering a conservative temperature level of 600 degrees Fahrenheit for the material properties, where the failure is at the CNV manway port due to the CNV shell pressure dilation that promotes gaps of the manway port opening. This ultimate internal pressure is above the internal design pressure of 1,200 psia as provided in Tables 3.8.2-1 and 6.2-1 of the FSAR.

Based on its review, the staff finds the applicant's design evaluation results acceptable because they are consistent with acceptance criteria in DSRS 3.8.2.II.4.F, and therefore, satisfy the criteria of GDC 50 and 10 CFR 50.44.

3.8.2.4.11 Containment Vessel Radiation Exposure Effects

In FSAR Section 3.8.2.4.6, "Containment Vessel Radiation Exposure Effects," the applicant describes the CNV radiation exposure effects. The applicant refers to FSAR Section 6.2 on the discussions on the effects of irradiation embrittlement of lower CNV materials, which concludes that loss of fracture toughness is negligible at the beltline SA-965 Grade FXM-19 base metal and associated weld metal from neutron irradiation during the design lifetime. Section 8.2.6, "Irradiation Embrittlement of Lower Containment Vessel," in TR-123952-NP, the applicant states that the peak 57 effective full-power years fluence is 0.0075 dpa for the lower CNV beltline base metal and 0.0035 dpa for the lower CNV beltline welds and concludes that loss of fracture toughness in the lower CNV beltline SA-965 Grade FXM-19 base metal or associated weld metal from neutron irradiation during the design lifetime is negligible as well as these peak fluence values are tiny fractions of NUREG/CR-7027 threshold fluence for irradiation embrittlement.

The staff performed an independent review of NUREG/CR-7027 and determined that the existing fracture toughness data on austenitic stainless steels irradiated in light-water reactors indicate little or no loss of fracture toughness below an exposure of about 0.5 dpa. The existing data for welds also suggest that ~0.3 dpa may be considered a threshold neutron dose below which irradiation has little or no effect on fracture toughness of stainless steels welds. Therefore, the staff confirms that the applicant's conclusion that *"loss of fracture toughness in the lower CNV beltline SA-965 Grade FXM-19 base metal or associated weld metal from neutron irradiation during the design lifetime is negligible,"* due to its distance from the core region and is consistent with the acceptance criteria in 3.8.2.II.4.C, therefore the staff finds this acceptable.

3.8.2.4.12 Containment Vessel Cyclic Fatigue Analysis

In FSAR Section 3.8.2.4.7, "Containment Vessel Cyclic Fatigue," the applicant performs the fatigue analysis for the CNV including for the Class 1 reactor coolant pressure boundary (RCPB) nozzles penetrations (reactor cooling systems, pressurizer spray and reactor pressure vessel high point degasification) in accordance with the ASME Code, Section III, Paragraph NB-3200. Applicable cyclic, dynamic, pressure, and thermal transient loads and load combinations, discussed in Section 3.8.2.3, are considered in the fatigue evaluation. For the CNV process fluid penetrations classified as ASME Code Class 1, the fatigue analysis considers effects of the pressurized-water reactor (PWR) environment in accordance with RG 1.207 and NUREG/CR-6909. Section 3.7.3 of this report discusses operating basis earthquake seismic loads and analysis.

The staff finds that the calculations are in various stages of completion (some calculations were final with assumptions that needed to be verified, and some that were not final calculations) but determined that the calculations were in accordance with Section III of the ASME Code, and that any deficiencies were either identified in the corrective action program or were planned to be addressed when the calculations are made final during the ITAAC closure process. The staff finds that this satisfies GDC 50.

3.8.2.4.13 Structural Acceptance Criteria

In FSAR Section 3.8.2.5 the applicant describes the CNV structural integrity acceptance criteria limits, which are developed in accordance with Subarticles NB-3200 and NF-3200 of ASME Code, Section III, for plate-type and shell-type supports for the CNV support. In FSAR Tables 3.8.2-2, Table 3.8.2-3, and Table 3.8.2-5, the applicant tabulates the plant events, service levels load combinations and the ASME Code allowable stress limits for CNV pressure retaining items, Class 1 Supports, and Class 2 Supports for the CNV, and the associated bolts, bolted connections and supports. The applicant describes that the CNV is fabricated, installed, and tested according to Subsections NB and NF of ASME Code, Section III. The applicant refers to TR-123952, Revision 0, and is also provided in Section C of Section 6.2, of FSAR, describing CLIP. The CLIP provides assurance that leakage integrity of containment is maintained, and that containment leakage does not exceed allowable leakage rate values per the requirements of GDC 52 and performance of the preoperational and periodic integrated leak rate testing per the requirements of 10 CFR Part 50, Appendix J.

The staff reviewed FSAR Table 3.8.2-2 through Table 3.8.2-5, and finds them to be acceptable because the structural acceptance criteria comply with those identified in DSRS 3.8.2.II.5 and the loads and stress are defined in accordance with Section III of the ASME Code.

3.8.2.4.14 Materials, Quality Control Programs, and Special Construction Techniques

In FSAR Section 3.8.2.6 and Section 6.1.1.1, the applicant describes the CNV materials, which conform to the requirements of Subarticle NB-2000 and NF2000 in the ASME Code. The CNV fabrication conforms to the requirements of Subarticles NB-4000 and NF-4000 in the ASME Code. The CNV uses no special construction techniques. The quality control program involving materials, welding procedures, and nondestructive examination of welds conforms to Subarticles NB-2000, NB-4000, and NB-5000 in the ASME Code. FSAR Tables 6.1-1 and 6.1-2 list the materials used for fabrication of the CNV and attachments.

The staff reviewed FSAR Section 3.8.2.6, Section 6.1.1.1 and Tables 3.8.2-2 through Table 3.8.2-5, and finds to be acceptable because the materials, quality control and special construction techniques acceptance criteria comply with those identified in DSRS Section 3.8.2.I.6 and are in accordance with Section III of the ASME Code.

3.8.2.4.15 Testing and Inservice Inspection Programs

FSAR Section 3.8.2.7 describes the testing and ISI requirements for the CNV.

The applicant describes that nondestructive examination of the CNV pressure retaining and integrally attached materials meet the requirements of Article NB-5000 in the ASME Code, Section III, and NF-5000 using nondestructive examination methods of Section V and inservice inspections of Section XI.

The applicant describes the hydrostatic test of the CNV, performed after fabrication, in accordance with ASME Section III, Paragraph NB-6000 by pressurizing it to a minimum of 25 percent over design pressure of 1,200 psia. The acceptance criterion for the test is that there are no indications of leakage.

The applicant describes that nondestructive examination of the CNV after fabrication includes the following preservice examinations that are performed after hydrostatic testing but before code stamping:

- General visual examinations for pressure retaining surfaces above the reactor pool level in accordance with Paragraph IWE-2200,
- VT-3 visual examinations for pressure retaining surfaces below the reactor pool level in accordance with Paragraph IWE-2200,
- VT-1 visual examinations for pressure retaining bolting in accordance with Paragraph IWE-2200,
- Volumetric examinations for select welds in support of the break exclusion zone requirement in accordance with augmented requirements,

- Volumetric examinations for the: (a) CNV upper head to CNV upper seismic support shell, (b) CNV lower shell to CNV lower transition shell, and (c) CNV lower core shell to CNV lower head circumferential vessel welds in accordance with augmented requirements.

The applicant also refers to FSAR Section 6.2 that describes inservice inspections of the CNV. As described above, the applicant also refers to TR-123952, Revision 1, that describes the CLIP that provide adequate assessments of overall CNV leakage rates by performing Type B and Type C tests of 10 CFR Part 50, Appendix J.

The applicant also describes that Subsection IWE of the ASME Code, Section XI, requires for Class MC structures, 80 percent of one side of the pressure retaining boundary of the vessel be accessible for either direct or remote visual examinations for the life of the plant.

The staff reviewed FSAR Section 3.8.2.7 and finds that it is in accordance with the guidance in DSRS Section 3.8.2.II.7. The staff finds that this satisfies the criteria in GDC 53. SER Section 6.2.6 discusses containment leakage testing and gives the staff evaluation of compliance with 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”

3.8.2.5 Combine License Information Items

There are no COL information items for this area of review.

3.8.2.6 Conclusion

The staff finds that the applicant has adequately addressed the design of the steel containment in the FSAR, as supplemented by the documents presented during the audit, in accordance with the acceptance criteria set forth in DSRS Section 3.8.2. On this basis, the staff concludes that the design of the steel containment is acceptable and meets the relevant requirements of 10 CFR 50.44, 10 CFR 50.55a, and GDC 1, 2, 4, 16, 50, and 53.

3.8.3 Concrete and Steel Internal Structures of Steel Containments

The NPM does not use internal structures (compartments, pedestals, or walls). SER Section 3.8.2 gives the staff’s evaluation of connections between the CNV and the reactor vessel.

3.8.4 Seismic Category I Structures

3.8.4.1 Introduction

This section describes the review of areas relating to the structural design of Seismic Category I structures other than the containment, namely, the RXB and CRB. DSRS Section 3.8.4, “Other Seismic Category I Structures,” provides guidelines and acceptance criteria for reviewing issues related to the design of Seismic Category I structures other than the containment.

The Seismic Category I structures are portions of the RXB and portions of the CRB. These buildings are site-independent and designed for the CSDRS and the certified seismic design response criteria - high frequency (CSDRS-HF) described in Section 3.7.1. The static and seismic analyses of the Seismic Category I structures are performed using ANSYS (Reference 3.8.4-1). The ANSYS software used for performing seismic analysis of SC-I SSCs conforms with the requirements for computer software as per the NuScale Quality Assurance Program Description (QAPD) (Reference 3.7.1-12).

3.8.4.2 Summary of Application

The applicant describes that the static and seismic analyses are performed using the ANSYS finite element computer code, which conforms to the applicant's requirements for computer software under the NuScale QAPD. The applicant also describes that the Seismic Category I buildings are site-independent and designed for the CSDRS and the CSDRS-HF as described in Section 3.7.1.

In FSAR Section 3.8.4.1, "Description of the Structures," the applicant describes the physical description and the primary functions of the Seismic Category I structures, primarily RXB, CRB and other structures that also includes RXB components. The applicant also issues a COL Item 3.8-1 for the design of the reactor flange tool.

In FSAR Section 3.8.4.2, "Applicable Codes, Standards, and Specifications," the applicant describes the codes, standards, and specifications meeting acceptance criteria in DSRS, specifically, compliance with ACI 349, AISC N690, ASCE 7 and ASCE 4.

In FSAR Section 3.8.4.3, "Loads and Load Combinations," the applicant refers to Table 6-2 and Table 8-2 in TR-0920-71621-P-A, Revision 1, "Building Design and Analysis Methodology for Safety-Related Structures," (ML20353A404) (TR) of SC walls and RC structures, respectively, for the loading combinations for the structural design and analysis of the Seismic Category I portions of RXB and CRB.

In FSAR Section 3.8.4.4, "Structural Modeling and Analysis Procedures," the applicant describes that the methodology of structural design and analysis of the Seismic Category I portions of RXB and CRB provided in TR-0920-71621-P-A, Revision 1 and the calculated DCR values at selected critical sections are summarized in FSAR Appendix 3B.

In FSAR Section 3.8.4.5, "Structural Design and Acceptance Criteria," the applicant describes the acceptance criteria in TR-0920-71621-P-A, Revision 1.

In FSAR Section 3.8.4.6, "Materials, Quality Control and Special Construction Techniques," the applicant describes the material properties in Table 3.8.4-3 for the structural design and analysis.

In FSAR Section 3.8.4.7, "Testing and Inservice Inspection Requirements," the applicant only requires quality control performances for concrete members and SC wall per the requirements of ACI 349 and AISC N690 and issued a COL Item 3.8-2 for a site-specific program for monitoring and maintenance of the Seismic Category I structures.

In FSAR Section 3.8.4.8, "Evaluation of Design for Site-Specific Acceptability," the applicant describes the evaluation of design and analysis in Section 3.8.4 could be acceptable if site-specific parameters were to be shown less than provided in Table 2.0-1 of the FSAR and forces experienced at the critical sections in Seismic Category I structures under the site-specific earthquake are less than that provided in the FSAR and supporting reports.

In FSAR Appendix 3B, “Design Reports and Critical Section Details,” the applicant summarizes the structural design and analysis with of the RXB and CRB that includes selection criteria for the critical sections, checking for the structural integrity of critical sections DCR values under load combinations.

ITAAC: ITAAC are evaluated in SER Section 14.3.

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

Technical Reports: There are no Technical Reports for FSAR Section 3.8.4.

3.8.4.3 Regulatory Basis

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

- 10 CFR 50.55a(1)(i)(E)(17) and GDC 1, as they relate to SSCs being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 2, as it relates to the design of structures important to safety being capable to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform their safety functions, the design bases for these structures should reflect as appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

- GDC 4, as it relates to appropriately protecting structures important to safety against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC 5, “Sharing of Structures, Systems, and Components,” as it relates to not sharing structures important to safety among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

- 10 CFR Part 50, Appendix B, as it relates to the QA criteria for safety-related SSCs of nuclear power plants.

The guidance in DSRS Section 3.8.4 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other DSRS and SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.69, Revision 1, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” issued May 2009.

- RG 1.91, Revision 2, “Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants,” issued April 2013.

- RG 1.115, Revision 2, “Protection Against Turbine Missiles,” issued January 2012

- RG 1.142, Revision 2, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” issued November 2001.

- RG 1.143, Revision 2, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” issued November 2001.

- RG 1.160, Revision 3, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” issued May 2012.

- RG 1.199, “Anchoring Components and Structural Supports in Concrete,” issued November 2003.

- RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” issued October 2011.

3.8.4.4 Technical Evaluation

The staff reviewed FSAR Section 3.8.4 in accordance with the DSRS Section 3.8.4. DSRS Section 3.8.4 describes the acceptance criteria to meet the relevant requirements of the NRC’s regulations pertaining to the structural design of Seismic Category I structures other than the containment. The summary of the application is discussed in SER Section 3.8.2.2 above.

3.8.4.4.1 Description of the Structures

In FSAR Sections 1.2.2.1, 3.7.2.1.2.1, 3.8.4.1.1 and Appendix 3B, the applicant provides general information related to the RXB. The RXB is deeply embedded with a center of gravity below the site grade elevation. The overall dimensions of the RXB are 231.5 ft, 155.5 ft, 171 ft in the east–west (X), north–south (Y), and vertical (Z) directions, respectively, and consists of SC walls, RC basemat and slabs and are designed to withstand the effects of natural phenomena. The thickness of the main structural interior and exterior SC walls is 4 ft. The RC floor slabs are either 2 ft. or 3-ft thick. The thickness of the east and west exterior SC walls is 5 ft. The thickness of the north and south exterior walls is 4 ft. The thickness of the basemat foundation is 8 ft. The thickness of roof slab is 3 ft.

The primary feature of the RXB is the pool located at the center of the building designed to be the UHS for the NPMs. The pool consists of the spent fuel pool and the refueling pool housing up to six NPMs. The normal depth of the reactor pool water is maintained at 53 ft. RXB includes the following components: bioshields, RXB pool liner, equipment door and dry dock gate. The design properties of the critical SC wall and RC slab sections of the RXB are provided in Tables 3B-1 and 3B-2 of the FSAR.

In FSAR Section 3.8.4.1.10, of the FSAR, the applicant describes that the modular construction techniques will be used to construct the Seismic Category I RXB SC walls. Modular construction techniques increase the efficiency of construction and productivity because the steel portions of the SC walls are fabricated off-site in a controlled environment. Furthermore, construction of the formwork at the site is not required and reinforcement is not needed.

In FSAR Sections 1.2.2.2, 3.7.1.2.5, 3.8.4.1.2 and Appendix 3B, the applicant describes the CRB. The CRB is an RC structure comprised of Seismic Category I, Category II and Category III (per FSAR Table 3.2-1) sections. The overall dimensions of the CRB are 120 ft, 55 ft, and 50 ft in the east–west (X), north–south (Y), and vertical (Z) directions, respectively, and consists of RC basemat, walls and slabs and are designed to withstand the effects of natural phenomena. The primary function of the CRB is to house the MCR and the technical support center. The critical wall and slab sections of the CRB are provided in Tables 3B-3 and 3B-4, of the FSAR.

The staff reviewed the descriptions of structures in FSAR Sections 1.2.2, 3.7.2, 3.8.4 and Appendix 3B, including general arrangement drawings with plan and section views of the structures, overall structural dimensions, floor and wall thicknesses, floor elevations, and steel reinforcement configurations. The staff's review found the level of detail with respect to the description of structures is sufficient for defining the primary structural aspects and elements that are relied upon to perform the safety-related functions of these structures. Specifically, based on the structural descriptions provided in the FSAR, the staff was able to identify the structural load path for the transfer of loads from the roof to the basemat of the structures. Further, the staff was able to identify enough dimensions to develop the dynamic models for the seismic analyses of the structures and establish the relationship between adjacent structures. Additionally, the staff found the structural descriptions contained sufficient details to confirm the consistency of the structural design aspects (e.g., structural member capacities and reinforcement configuration) in the design descriptions with the reference design codes. Based on the above, the staff concludes that the descriptions of structures in the FSAR are acceptable.

3.8.4.4.2 Applicable Codes, Standards, and Specifications

FSAR Section 3.8.4.2 lists the codes, standards, and specifications applicable to the Seismic Category I portions of the RXB and CRB. The staff reviewed the list of codes, standards, and specifications to confirm that the criteria used in the analysis, design, and construction of the RXB and CRB are consistent with the established criteria, codes, standards, and specifications acceptable to the staff. DSRS Section 3.8.4.II.2 lists the codes, standards, and specifications acceptable to the staff.

Based on the applicant's use of codes, standards, and specifications consistent with DSRS Section 3.8.4.II.2, and the conservative implementation of AISC N690-12 as described above, the staff concludes that the information in FSAR Section 3.8.4.2 on applicable codes, standards, and specifications for the other Seismic Category I structures of the NuScale design is acceptable.

3.8.4.4.3 Loads and Load Combinations

In FSAR Section 3.8.4.3, the applicant describes the loads and load combinations for the RXB and CRB structural design conform to the ACI 349-13, endorsed by RG 1.142, and AISC N690-18, Appendix N9, endorsed by RG 1.243, as the basis for the loads and load combinations. In Sections 3.8.4.3.1 through 3.8.4.3.22 of the FSAR, the applicant provides symbols and detail description of the applicable loads. The applicant refers to Table 6-2 and Table 8-2 in TR-0920-71621-P-A, Revision 1, for the load combinations of the SC walls and RC structures.

The staff reviewed Table 6-2 and Table 8-2 load combinations in the TR-0920-71621-P-A, Revision 1, and determined them to be consistent with Chapter NB2.5 of Specification N680-18 for the load resistance factor design (LRFD) method and ACI 349-13. For the differences in load factors between RG 1.142 and Table 8-2 in TR-0920-71621-P-A, Revision 1, the applicant selected the load factors in ACI 349-13 instead of the revised values in RG 1.142 because the values in ACI 349-13 more closely align with the recent design codes for developing load combinations.

The load combinations listed in ACI 349-13, and corresponding labels used in the calculations, are used in the design of the RC members and SC walls of the RXB and are provided in FSAR Tables 3B-6 and 3B-7, respectively. Further, DSRS provides the guidance and acceptance criteria specifically related to the design and evaluation of structural steel and concrete structures in the NuScale nuclear power plants. DSRS 3.8.4 refers to the applicable codes and standards, loads and load combinations, design and analysis procedures, structural acceptance criteria, and materials and special construction techniques. The acceptance criteria in DSRS 3.8.4-II states the structural acceptance criteria refers to the applicable codes of AISC N690-1994 and ACI 349-13 for the SC walls and concrete structures with additional criteria provided in RG 1.142. DSRS 3.8.4, Section II.4.J, also provides reference to guidance contained in NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants," and other applicable industry documents related to the use of modular construction methods. Use of these guides and specifications in TR-0920-71621-P-A, Revision 1, provide assurance and impose specific restrictions to ensure that the SC walls and concrete structures will perform their intended safety function with the identified loads and their load combinations, and therefore acceptable to the staff.

In FSAR Appendix 3B, Section 3B.1.3, the applicant determines the cracking states due to in-plane-demands using seismic combination of "D + 0.8L + E_{ss}," where a label of "CrkEs," was assigned for this load combination case in the calculations. Further, the staff agrees with this seismic load combination, since the maximum differential pressure load, Pa, fluid pressure load, F, soil pressure load, H, and thermal loads during normal and abnormal conditions, T_o, and T_a, are expected to have a major effect on the out of plane flexure of walls and slabs but only a minor effect in their in-plane direction.

The load combination in Equation 3B-1 is only used to evaluate the state of cracking in walls and slabs of the seismic force resisting system. The full load combinations in ACI 349 or AISC N690-18 are used to obtain the member forces for design.

In accordance with COL Item 3.6-1, the COL applicant will address final piping layout, analysis, and additional protection features as necessary. Based on the applicant's generic evaluation, the staff's review, and the site-specific verifications to be performed by the COL applicant, the staff finds the applicant's consideration of Yj, Yr, and Ym concentrated local load effects in the RXB design to be acceptable.

The load and load combinations follow the requirements of ASIC N690-18, Appendix N9, and 360-16, and consistent with the structural acceptance criteria in NRC NuScale DSRS Section 3.8.4, DSRS 3.7.2. DSRS 3.8.4 provides guidance regarding basic specifications for concrete and steel structures in compliance with NRC regulations and cites certain RGs and industry consensus codes and standards, specifically ACI 349-13 and AISC N690-18, Appendix N9 that are acceptable to the staff. The staff reviewed and compared the loads and load combinations presented in the FSAR with the referenced codes. The staff's review found that the load definitions and load combinations conform with the referenced codes and therefore acceptable.

3.8.4.4.4 Design and Analysis Procedures

In FSAR Section 3.8.4.4, and Appendix 3B, the applicant provides an overview of the design and analysis requirements for the RXB and CRB. The applicant describes that the structural integrities of RXB and CRB are to be maintained, and the safety-related SSC remain operable during and following an earthquake represented by the CSDRS or the certified CSDRS-HF. Specifically, Appendix 3B addresses twelve critical sections in the RXB and 5 in the CRB that were selected because they (1) perform a safety-critical function, (2) are subjected to large stress demands, (3) are considered difficult to design or construct, or (4) are representative of the structural design.

3.8.4.4.5 Analysis Procedures

The applicant performed static and seismic analyses with the ACS SASSI, SHAKE2000 and ANSYS computer codes to determine the structural response to non-seismic and seismic loads, including the fluid-structure interaction effects. Additionally, the applicant performed thermal and pressurization analyses with ANSYS. Consistent with the acceptance criteria in DSRS Section 3.8.4.II.4, the staff determined the use of these computer programs to be acceptable because these programs are recognized and have a sufficient history of use in the nuclear industry to justify their applicability. Section 3.7.2 of this SER provides specific details with respect to the staff's review of the modeling and analysis performed with ACS SASSI and SHAKE2000.

ANSYS is a general purpose commercially available finite element program that has been widely used and accepted by the engineering community. It is used in a variety of structural applications including both linear and nonlinear static and dynamic analysis. Further, the engineering community has accepted ANSYS and used it in nuclear applications to obtain results that are also acceptable to the staff. The applicant performed the analyses with ANSYS, which consistent with the acceptance criteria in DSRS Section 3.8.4.II.4, the staff determined the use of these computer programs to be acceptable because these programs are recognized in the public domain and have a sufficient history of use to justify their applicability.

The design and analysis methodologies to determine the in-structure response spectra, effective stiffness, damping ratio, SC walls and connections and RC structures for Seismic Category I and Category II structures are described in the TR-0920-71621-P-A-R1 (TR). The staff reviewed the TR to confirm that the methodologies presented in the TR are consistent with the established criteria acceptable to the staff, as presented in Sections 3.7.2 and 3.8.4 of the DSRS.

The seismic load-resisting RC members and SC wall sections are checked for potential cracking from the CSDRS motion using the maximum force calculated in a structural member during the entire time-history. The method used to determine the effective stiffness and damping of the RC and SC members are recommended by ASCE/SEI 4-16, ASCE/SEI 43-19, and AISC N690-18 and are considered acceptable by the staff.

All cracked RC members are assigned the effective stiffness values and corresponding damping, expressed as fraction of critical damping, in accordance with ASCE/SEI 4-16 (Tables 3-1 and 3-2) and Table 3-1 of ASCE/SEI 43-19 and consistent with DSRS

Section 3.7.2. All SC members are assigned the effective stiffness values as provided in Specification N690-18 and the damping values provided in ASCE/SEI 43-19.

As presented in the TR, the effective stiffness of an SC wall for both operational and thermal conditions is specified in Section N9.2.2 of AISC N690-18, and the out of plane flexural stiffness is calculated based on the stiffnesses of the cracked concrete infill and the faceplates using Equation A-N9-8 from Section N9.2.2 of Specification N690-18. The effective in-plane shear stiffness per unit width of the SC wall for operating conditions depends on the ratio of the average in-plane required shear strength, S_{rxy} and the concrete cracking threshold, S_{cr}, and is calculated by the trilinear relationship given by Equations A-N9-9 through A-N9-14 of Section N9.2.2 of AISC N690-18. The effective in-plane shear stiffness per unit width of the SC wall for accidental thermal conditions is determined using Equation A-N9-12 assuming cracked concrete. The threshold for crack developing in concrete, S_{cr}, is given by Equation A-N9-10.

This proposed methodology uses three different materials to represent the widely different surrounding media: a soft soil profile (Type 11), a rock profile (Type 7), and a hard rock profile (Type 9).

The analysis starts with the structure subjected to a CSDRS motion (demand) with the structural members having uncracked material properties and with response level- (RL) 1 damping values, as permitted by ASCE/SEI 43-19. This harmonic analysis is repeated for the three soil types considered. The seismic load-resisting RC members and SC wall sections are checked for the potential for cracking from the CSDRS motion using the maximum force calculated in a structural member during the entire time-history considering the most critical seismic load combination. The state of cracking of a member is calculated by the DCR.

The ISRS at a given location of a structural member is generated from the harmonic analysis with the updated stiffness and damping properties for each of the CSDRS motions by algebraic summation of the acceleration time history in each direction from the input motion in the three orthogonal X, Y, and Z directions. The ISRS is calculated at 2, 3, 4, 5, 7, and 10 percent of the critical damping. The average ISRS is calculated from the results obtained for each CSDRS motion used. The peak of the ISRS is broadened by ±15 percent following RG 1.122 (ML003739367) to account for the uncertainties in the structural frequencies. The ISRS, calculated at all nodes of a structural member, is enveloped by repeating the analysis for the three soil types selected, with the final ISRS selected that envelope the ISRS determined for each of the soil type.

In FSAR Section 3.8.4.3.22.2, the applicant describes the structural analysis performed for six NPMs in their respective bays, even though the operations with fewer than six NPMs are allowed. However, the applicant describes that the dynamic effects on the building with fewer than six NPMs would be similar compared to when all six NPMs are in place. Consideration of all six NPMs is conservative given that the weight of all six NPMs in the pool would result in a relatively higher demand on the structural members of the RXB and, therefore, is acceptable.

In FSAR Section 3.8.4.4, and Appendix 3B, the applicant describes the structural modeling and analysis procedures for the RXB and CRB as well as associated components. The applicant uses the ANSYS structural analysis software for design and analysis of Seismic Category I and II structures. The applicant developed three-dimensional DB models of RXB and CRB, and CRB with major components using the following elements in ANSYS: solid shell (SOLSH190), shell (SHELL181), beam (BEAM188), fluid (FLUID30), surface (SURF154), mass (MASS21), soil (MATRIX50), and surface elements (TARGE170 and CONT174). The applicant also used the thermal shell elements. SHELL131, for thermal analysis of the RXB SC and RC walls and slabs. Figure 3.7.2-60 through Figure 3.7.2-87 in the FSAR, show isometric and section views of the RXB ANSYS model identifying the primary element types with colors. FSAR Figure 3.7.2-4 shows isometric view of the CRB ANSYS model. The applicant used the hybrid DB (RXB and RWB) and CRB models for design of the critical sections. The applicant used results from the ANSYS analysis to determine the structural response for static and dynamic loads, and post-processing of the analysis results. In addition to Soil 7, 9, and 11, soil 7 with soil separation modeling capability are considered in design calculations. In this context, soil 7 with soil separation capability is treated as a new soil type.

The staff finds the design-basis demands, which envelops analysis results considering a range of key structural and site parameters, and further site-specific verifications to be performed by the COL applicant to be conservative and acceptable.

3.8.4.4.6 Design Procedures

In FSAR Appendix 3B, the applicant describes the structural design and analysis of the RXB and CRB. Appendix 3B provides detail descriptions of the analysis and design of selected critical sections of the SC walls and RC sections to withstand the design-basis demands. The applicant describes the critical sections as part of a structure which are selected using the following criteria: (1) perform a safety-critical function, (2) are subjected to large stress demand, (3) are considered difficult to design or construct, or (4) are representative of the structural design. The applicant lists the critical sections of the RXB and CRB and their design properties in Tables 3B-1 and 3B-2, and in Tables 3B-3 and 3B-4 of the FSAR, respectively. FSAR Table 3B-8 and Table 3B-9 provide the cracked states based on load combinations for SC walls and RC members, respectively for the RXB. For each load combination and action, the applicant calculates the DCR values by dividing the total demand by the capacity of the critical sections.

The staff’s review confirmed that the design conditions and calculated maximum DCR values of the critical sections as listed in FSAR Appendix 3B, Tables 3B-10, 3B-12, 3B-14 and 3B-20 for the RXB and in Tables 3B-23 through 3B-33 for the CRB are primarily equal to or less than 1.0. The DCR values are calculated and assessed following both element-based and panel section-based approaches, and SC wall regions where the element-based approach results in high DCR values are reevaluated using panel section-based approach. The applicant summarizes the maximum DCR values at critical section of SC Walls on RXB in Table 3B-10 showing the DCR counter plots in Figures 3B-17 through Figure 3B-22 and determines that there are three (3) localized areas at SC Wall RX1, RXE and Pool Wall where the DCR values exceed 1.0 by no more than 5% for design conditions of cMS (1.05 and 1.04) and Vy (1.02), respectively. The applicant describes that these localized exceedances occur at joint regions and reentrant corners and do not impact the overall structural design and safety of the walls. Because the effect of additional localized reinforcement is not included in the DCR calculations with the limitations of finite element modeling of challenging geometries, with elastic properties, providing high levels of localized stress levels. On this basis, the staff concludes that the DCR values of the selected critical sections of the RXB and CRB are within the limits specified by AISC N690-18 and are, therefore, acceptable.

3.8.4.4.7 Design Checks

Although the staff performed independent reviews of all the design checks for the selected critical sections of RXB and CRB addressed in the FSAR, a summary of the evaluations of the design checks limited only to the sections in the RXB is described in this report.

3.8.4.4.8 RXB SC Wall Design Checks

In FSAR Section 3B.2.2.3, the applicant describes the procedures used to calculate the DCR values for each design condition at each of the finite elements of the critical SC wall sections for all active load combinations. The applicant lists the maximum DCR values in FSAR Appendix 3B, Table 3B-10 for the SC wall of RX1, RX4, RX4.3, RX 4.6, RXB, RXE and Pool Walls.

Based on its review, the staff finds the applicant’s design evaluation results are acceptable because the RXB SC walls retain greater capacity than the demands. As concluded above, the staff also agrees with the applicant’s conclusion that the three (3) localized DCR exceedances of no more than 5% would not affect the overall structural design and safety of RXB SC Walls. Based on the seismic demands and demonstration of adequate RXB SC Wall capacities, the staff concludes that the RXB SC walls are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.9 Reinforced Concrete Slabs Design Checks in RXB

In FSAR Appendix 3B, Section 3B.2.3.1, the applicant identifies the critical sections from the member of RC slabs of the basemat, floor slab at elevation 100 ft, and the roof slab of the RXB. The applicant also identifies the initial design properties of the critical sections of RXB in Table 3B-2.

In FSAR Appendix 3B, Section 3B2.3.2, the applicant calculates the strength required at the critical sections of the RC slabs for the load combination of Load Combination 6 (LC6), as described in the FSAR Equation 3B-15, for Soil Type 7 (a rock profile). The peak contour plots of “Combined Demands for Load Combination LC6_p (force unit kip/ft and moment unit kip-in./ft),” of all elements for the basemat slab, floor slab at elevation 100 ft, and the roof slab under the LC6 demand load combination are shown in Figures 3B-24 through 3B-26. The element-based contour plots confirm that the selected critical sections match well with the locations where the demand values are the largest. Based on the contour plots, additional critical sections are selected for analysis under LC6.

In FSAR Appendix 3B, Section 3B2.3.3, the applicant describes the design calculations of the RC members under demand values at the selected critical sections are shown and labeled in Figures 3B-27 through 3B-29 for the basemat slab, floor slab at elevation 100 ft and the roof slab. Using the initial design properties from Table 3B-1, the applicant calculates design values for out of plane demands (axial force - out of plane moment and axial force - out of plane shear).

The applicant also describes that the DCR values for in-plane shear conditions at the critical sections associated with the maximum allowed in-plane shear capacity are calculated for the governing load combination and soil type by taking the ratio of the required in-plane shear reinforcement, r_o, to the maximum in-plane shear reinforcement, r_{o_max}. The applicant describes that the additional required in-plane shear reinforcement are added to the longitudinal reinforcement. The final design properties of critical sections of RC members in RXB and the maximum DCR values are provided in Table 3B-14.

The applicant provides the summary of calculated DCR values at the critical sections of RC members for out of plane design conditions in FSAR Appendix 3B, Table 3B-13 and in-plane-shear design condition in Table 3B-13 that the DCR values are less than 1.0. The applicant provides the final design properties of critical sections of RC members in RXB and the maximum DCR values in Table 3B-14 that refers to Figures 3B-30 through 3B-31a providing reinforcement layouts for the basemat slab, floor slab at elevation 100 ft and the roof slab.

Based on its review, the staff finds the applicant's design evaluation results acceptable because the DCR results for the reinforced concrete slabs in RXB listed in FSAR Appendix 3B, Tables 3B-12, 3B-13, and 3B-14, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity, the staff concludes that the reinforced concrete slabs in RXB are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.10 NuScale Power Module Skirt Support

In FSAR Section 3B.2.4.1, the applicant describes NPM skirt restraint providing lateral support at the base of the NPM. Figure 3B-33 shows a plan view of the typical NPM skirt support, the annular bearing plate, and lateral skirt restraints.

The applicant describes that the two evaluations: (1) the vertical analysis evaluates the annular bearing plate and (2) the lateral analysis evaluates the lateral braces based on the load path, starting with combined axial and bending of the braces, local bearing on the braces, evaluation of the brace connections to the SC walls, and local evaluation of the SC walls.

The applicant describes that the acceptance criteria of structural steel components of NPM skirt support skirt restraint conform to the provisions of AISC N690-18. The LRFD load combinations using the governing equation of NB2-6 of AISC N690-18 are used for evaluation of the NPM skirt support skirt restraint.

The applicant tabulates the calculated DCR values of the components of NPM skirt restraints in Table 3B-15 for vertical and lateral analyses.

Based on its review, the staff finds the applicant's design evaluation results acceptable because the DCR values for the NPM skirt restraint components listed in Table 3B-15, of the FSAR, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity as described above, the staff concludes that the lug restraints are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.11 NuScale Power Module Lug Restraints

In FSAR Section 3B.2.4.2, the applicant describes that the three-lug restraint are Seismic Category I components where the configurations is extending from the bay SC walls with the wedge-jacks preventing the lateral movement of the NPM. FSAR Appendix 3B, Figures 3B-36 and 3B-37 show the plan layout of lug restraint configurations in six NPM bays and components of the typical lug restraint in one NPM bay, respectively. When the wedge-jacks are in retracted position, there is a gap to allow the NPM to be removed/reinstalled by the RBC.

The applicant describes that the NPM lug restraint and associated components are evaluated for circumferential, radial, and vertical design seismic force from NPM lug. The design seismic loads are: Circumferential: 1,500 kips, Radial: +/- 112.5 kips and Vertical: +/- 112.5 kips. The applicant also notes that the boundary conditions NPM lug restraint is only subject to dead load and seismic loads at accident temperature. And the LRFD load combinations in Section NB2 of AISC N690-18 are used for evaluation of the NPM seismic lug restraint components. The applicant describes that the acceptance criteria of structural steel components of NPM lug restraint conform to the provisions of AISC N690-18. In FSAR Appendix 3B, Table 3B-16, the applicant lists the calculated DCR value for the components in load path for the NPM lug restraint.

Based on its review, the staff finds the applicant’s design evaluation results acceptable because the DCR values for the NPM Lug restrain components in the load path, listed in FSAR Table 3B-16, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity as described above, the staff concludes that the lug restraints are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.12 Reactor Building Crane Corbel

In FSAR Appendix 3B, Section 3B.2.4.2, the applicant describes the RBC corbel, as shown in Figures 3B-23, 3B-38 and 3B-39, are the two continuous stiffened ledges attached to the RXB and RXD SC walls design at elevation 145 feet 6 inches to support moving point loads of RBC. The components of RBC corbel attached to the SC wall module are made from American Society for Testing and Materials (ASTM) A572 Grade 55 material. The loads evaluation of the RBC corbel is split into three sections: downward load analysis, upward load analysis, and lateral analysis. The LRFD load combinations of the governing equation of NB2-6 of AISC N690-18 correspond to the extreme environmental load combination, and are used for the structural integrity evaluations. In FSAR Table 3B-7, the applicant lists the calculated DCR values for the components of RBC corbel.

Based on its review, the staff finds the applicant’s design evaluation results acceptable because the DCR values of RBC corbel components in the load path listed in Table 3B-16, of the FSAR, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity as described above, the staff concludes that the RBC corbel components are designed to retain their structural integrity when subjected to the design-basis demands, which is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.13 Structural Acceptance Criteria

The applicant describes that Seismic Category I structural steel and SC wall components are designed to AISC N690-18 and Seismic Category I SC members are designed to ACI 349-13. In FSAR Section 3.8.4.5, the applicant refers to TR-0920-71621-P-A, Revision 1, providing the structural design and acceptance criteria for the seismic Category RXB and CRB. The applicant also describes that the acceptance criteria in FSAR Appendix 3B provide with structural design evaluation results for selected critical sections of the RXB and CRB checked by calculating the DCR values, for each load combination and action, by dividing the total demand by the capacity, at the critical sections that the DCR value must be less than one for acceptable design.

The staff reviewed the structural acceptance criteria in FSAR Section 3.8.4.5 for application to the Seismic Category I structures, SC walls and RC members. The staff found the use of these structural acceptance criteria to be in accordance with the guidance given in DSRS Section 3.8.4.II.5 and, with respect to the updated criteria in AISC N690-18 and ACI 349-13 to be implemented conservatively as described in Section 3.8.4.2 of this report. On this basis, the staff finds the information in FSAR Sections 3.8.4.5 and Appendix 3B on the structural acceptance criteria to be acceptable.

3.8.4.4.14 Materials, Quality Control, and Special Construction Techniques

In FSAR Sections 3.8.4.6.1, the applicant refers to FSAR Table 3.8.4-3 for the principal construction materials for structures including SC walls are concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. In FSAR Appendix 3B, Sections 3B.2.1.2, 3B.2.5.2 and 3B.3.1.2, the applicant provides the structural material requirements for RXB, SC Wall to RC slab connections and CRB, respectively. In FSAR Section 3.8.4.6.1.1, the applicant describes the structural concrete, used in the Seismic Category I RXB and CRB, conforms to ACI 349-13 as applicable, ACI 318-08 and for the SC walls per the requirements of AISC N690-18 and, as applicable, AISC 360-16.

In FSAR Section 3.8.4.6.1.1, the applicant provides the following engineering requirements for the concrete:

- The compressive strengths of concrete (f_c) are 5,000 psi, typical, and 7,000 psi, for the RXB roof and floor slabs, as tabulated in Table 3.8.4-3 of the FSAR.

- Concrete mixes are designed in accordance with ACI 211.1, “Selecting Proportions for Normal-Density and High Density-Concrete – Guide.”

- Cement conforms to the requirements of ASTM C150, “Standard Specification for Portland Cement.”

- Aggregates conform to the requirements of ASTM C33, “Standard Specification for Concrete Aggregates.” Further, ASTM C1260, “Standard Test Method for Potential Alkali Reactivity of Aggregates,” and C1293, “Standard Test Method for Determination of Length Change of Concrete Due to Alkali-Silica Reaction,” are used in testing aggregates for potential alkali-silica reactivity. Concrete with potentially reactive aggregates uses low-alkali cement.

- Air-entraining, chemical, and fly ash and pozzolan admixtures, if used, conform to the requirements of ASTM C260, “Standard Specification for Air-Entraining Admixtures for Concrete,” C494, “Standard Specification for Chemical Admixtures for Concrete,” and C618, “Standard Specification for Coal Fly Ash and Raw or Calcined Natural Pozzolan for Use in Concrete,” respectively.

- Water and ice for mixing are clean, with a total solids content of not more than 2,000 parts per million.

Further, in addition to ACI 349-13, FSAR Section 3.8.4.6.1.1 addresses codes and standards used for concrete construction, including placement, inspection, and testing. These include:

- ACI 301, “Specifications for Structural Concrete for Buildings,”
- ACI 304R, “Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete,”
- ACI 305.1, “Specification for Hot-Weather Concreting,”
- ACI 306.1, “Specification for Cold-Weather Concreting,”
- ACI 347, “Recommended Practice for Concrete Formwork,”
- ACI SP-2, “Manual of Concrete Inspection,”
- ASTM C94, “Specification for Ready-Mixed Concrete.”

In FSAR Section 3.8.4.6.1.2, the applicant describes the reinforcing steel conforms to ASTM-designation A615 grade 60 or A706 grade 60. Concrete reinforcement is emplaced in accordance with ACI 349. Reinforcing development length and splice length are calculated by ACI 349-specified formulas. Welded wire fabric for concrete reinforcement conforms to ASTM A185 (plain wire) or ASTM A497 (deformed wire).

In FSAR Section 3.8.4.6.1.2, the applicant refers to TR-0920-71621-P-A, Revision 1, for the SC wall requirements.

In FSAR Section 3.8.4.6.1.4, the applicant provides the following list of engineering requirements for the connections:

- Steel bolts conform to either ASTM A307, high-strength ASTM A490, or ASTM A325.
- Material. Steel studs meet the requirements of ASTM A108 and American Welding Society D1.1/D1.1M, “Structural Welding Code-Steel.”
- Anchor bolts are of type ASTM F1554 36 ksi or 55 ksi yield-strength material or ASTM F1554 105 ksi yield-strength or higher strength material.
- Welding electrodes are E70XX, unless otherwise noted on drawings, or are within the specification for ASTM A36 steel and E308L-16 or equivalent for ASTM A240-type 304-L stainless steel.

In FSAR Section 3.8.4.6.1.4, the applicant describes that grating is welded and galvanized steel, "Metal Bar Type,” conforming to ANSI/NAAMM MBG 531-00, “Metal Bar Grading Manual,” and ANSI/NAAMM MBG 532-00, “Heavy Duty Metal Bar Grating Manual.” Further, the applicant describes that there are no safety-related reinforced masonry walls in Seismic Category I structures.

In FSAR Section 3.8.4.6.2, the applicant refers to Chapter 17, for the details of the Quality Assurance Program.

The staff's review confirmed that the material specifications discussed above are within the scope of the primary design codes; that is, ACI 349-13 and AISC N690-18 or other referenced codes and standards are within the scope of the primary design codes as well as is consistent with DSRS Section 3.8.4.II.6. Therefore, the staff finds these certifications to be acceptable.

3.8.4.4.15 Testing and Inservice Surveillance Requirements

FSAR Section 3.8.4.7, "Testing and Inservice Inspection Requirements," states that there is no testing or inservice surveillance beyond the quality control tests performed during construction, which is in accordance with ACI 349-13 and AISC N690-18. Further, the applicant issues a COL Item 3.8-2 for a COL applicant that references the NuScale Power Plant US460 will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," as discussed in RG 1.160. Monitoring is to include below-grade walls, ground water chemistry, if needed, base settlements, and differential displacements. The staff finds the above-described testing or inservice surveillance and program for monitoring and maintenance to be consistent with DSRS Section 3.8.4.II.7 and therefore acceptable.

Further, FSAR Table 1.9-2 shows that the COL applicant is responsible for the water control structures and associated ISI and surveillance programs, in accordance with RG 1.127. The use of RG 1.127 for addressing the site-specific inspection and surveillance programs is consistent with DSRS Section 3.8.4.II.7 and is therefore acceptable.

3.8.4.5 Combined License Information Items

Table 3.8.4-1 lists COL information item numbers and descriptions related to the structural design of Seismic Category I structures, other than containment, from, Table 1.8-2.

Table 3.8.4-1: NuScale COL Information Items for FSAR Section 3.8.4

Item No.	Description	FSAR Section
COL Item 3.8-1	An applicant that references the NuScale Power Plant US460 standard design will provide the design of the reactor flange tool.	3.8.4.1.5.4
COL Item 3.8-2	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below-grade walls, groundwater chemistry if	3.8.4.7

	needed, base settlements, and differential displacements.	

3.8.4.6 Conclusion

The staff finds that the criteria used in the analysis and design of NuScale's Seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime conform with established criteria, codes, standards, and specifications and are therefore acceptable to the NRC staff. On this basis, the staff concludes that the design of NuScale's Seismic Category I structures other than containment (addressed in SER Section 3.8.2) is acceptable and meets the relevant requirements described in Section 3.8.4.3 of this SER.

3.8.5 Foundations

3.8.5.1 Introduction

This section documents the staff's review of areas related to the structural design of Seismic Category I foundations for the RXB and CRB. DSRS Section 3.8.5, "Foundations," provides guidelines and acceptance criteria for reviewing issues related to the foundations of all Seismic Category I structures.

3.8.5.2 Summary of Application

FSAR Sections 3.8.4, 3.8.5 and Appendix 3B, "Design Reports and Critical Section Details," provide information on the structural design and analysis of the Seismic Category I RXB and CRB structures and foundations.

The applicant described structures and foundations; applicable codes, standards, and specifications; design and analysis procedures; loads and load combinations; structural acceptance criteria; settlement; thermal loads; construction loads; leak detection; materials, quality control, and special construction techniques; and testing and inservice inspection (ISI) requirements. The applicant also described COL information items related to structural design aspects of Seismic Category I structures.

The applicant indicated that the Seismic Category I structures (other than the containment) are portions of the RXB and the CRB. Both buildings designed based upon generic soil profiles and FSAR Section 2.0 enveloping site parameters are site independent. The applicant stated that the CRB is located northwest of the RXB and that there is an underground ductbank between the two buildings. The applicant performed the static and seismic analyses using ANSYS finite element analysis software.

FSAR Appendix 3B provides a design report for critical sections. In accordance with Appendix 3B, the applicant selected the critical sections based on whether they (1) perform a safety-critical function, (2) are subjected to large stress demands, (3) are considered difficult to design or construct, or (4) are considered to represent the structural design.

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.

3.8.5.3 Regulatory Basis

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

- GDC 1, as it relates to safety-related structures being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 2, as it relates to the design of the safety-related structures being capable to withstand the most severe natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, and the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

- GDC 4, as it relates to appropriately protecting safety-related structures against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC 5, as it relates to not sharing safety-related structures among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

- 10 CFR Part 50, Appendix B, as it relates to the QA criteria for nuclear power plants.

The guidance in DSRS Sections 3.8.4 and 3.8.5 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other DSRS sections. In addition, the following guidance documents provide acceptance criteria that confirm the above requirements have been adequately addressed:

- RG 1.206, as it provides the basis for evaluating the description of structures to be included in a DC or a COL application.

- RG 1.142, as it describes methods and procedures for the analysis, design, construction, testing, and evaluation of safety related nuclear concrete structures (excluding concrete reactor vessels and concrete containments) that comply with NRC regulations.

3.8.5.4 Technical Evaluation

The staff reviewed FSAR Section 3.8.5, in accordance with DSRS Section 3.8.5. DSRS Section 3.8.5 describes acceptance criteria to meet the relevant requirements of the NRC’s regulations pertaining to foundations of all Seismic Category I structures. Consistent with DSRS Section 3.8.5, the staff reviewed (1) the description of the foundations, (2) applicable codes, standards, and specifications, (3) loads and load combinations, (4) design and analysis procedures, (5) structural acceptance criteria, (6) materials, quality control, and special construction techniques, and (7) testing and inservice surveillance requirements. The staff also reviewed applicable COL information items.

3.8.5.4.1 Description of Foundations

The staff reviewed the descriptions of the foundations to ensure that they contain sufficient information to define the primary structural aspects and elements that are relied upon to perform the safety-related functions of these structures. The primary function of a foundation is to transmit the loads imposed by the superstructure to the underlying supporting media, rock, or soil. The staff's review also ensures that the foundation design meets the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and that is in accordance with DSRS Acceptance Criterion 3.8.5.II.1.

FSAR, Section 3.8.5.1, "Description of Foundations," describes the physical and functional characteristics of the reinforced concrete basemats of the RXB and CRB for the NuScale US460 Power Plant. The applicant identified the RXB and CRB as Seismic Category I. FSAR Tables 3B-11, 3B-14 and 3B-22 and FSAR Figures 3B-30, 3B-32, and 3B-51 contain the information of steel reinforcement for the RXB and CRB basemats.

The applicant described the RXB basemat dimensions as 70.1 m (230 ft) by 47.2 m (155 ft), with a minimum thickness of 2.44 m (8 ft). The applicant indicated that the foundation top of concrete (TOC) elevation is 7.6 m (25 ft), except for the refueling pool area which has a TOC elevation of approximately 7.9 m (26 ft), **{{ }}**. FSAR Tables 3B-11 and 3B-14 and FSAR Figures 3B-30 and 3B-32 provide the information of steel reinforcement and reinforcement layout for the RXB basemat. Typical longitudinal reinforcement of the RXB basemat consists of four layers of #11 bars centered at 30 cm (12 in) each way on top and bottom surfaces, and typical shear reinforcement consists of #4 ties centered at 30 cm (12 in) each way.

The applicant described the CRB basemat dimensions as 36.6 m (120 ft) by 16.8 m (55 ft), with a thickness of 1.5 m (5 ft). The applicant indicated that the **{{ }}**. FSAR Table 3B-22 and Figure 3B-51 provide the information of steel reinforcement for the CRB basemat. Typical longitudinal reinforcement of the CRB basemat consists of two layers of #11 bars centered at 30 cm (12 in) each way on top and bottom surfaces, and typical shear reinforcement consists of #3 ties centered at 30 cm (12 in) each way.

The staff reviewed the descriptions of the foundations for RXB and CRB buildings to ensure that they contain sufficient information to define the primary structural aspects and elements that are relied upon to perform the safety-related functions of these structures. The primary function of a foundation is to transmit the loads imposed by the superstructure to the underlying supporting media, rock, or soil. The applicant's description meets the applicable requirements in 10 CFR Part 50, Appendix B, General Design Criterion (GDC) 1 thus in accordance with the guidance in DSRS Acceptance Criterion 3.8.5.II.1.

3.8.5.4.2 Applicable Codes, Standards, and Specifications

FSAR Section 3.8.5.2 refers to FSAR Section 3.8.4 for the codes, standards, and specifications used to design and construct the RXB and CRB structures and foundations. FSAR Section 1.9 presents the regulatory guides applicable to design and construction of the Seismic Category I portions of the RXB and CRB. The applicant indicated that they would use the latest endorsed edition of the ASTM standards at the time of the construction.

Section 3.8.4 of this SER documents the staff conclusion and review of the applicable codes, standards, and specifications used for the structures and foundations.

3.8.5.4.3 Design and Analysis Procedures

DSRS Section 3.8.5 provides review guidance pertaining to the design and analysis procedures of foundations. FSAR Tables 3B-11, 3B-14, 3B-22 and FSAR Figures 3B-30, 3B-32, and 3B-51 provide the steel reinforcement patterns for the RXB and CRB foundations based on the structural analyses and calculations; and FSAR Section 3.8.5.3 "Design and Analysis Procedures" describes the RXB and CRB stability analysis model. FSAR Appendix 3B summarizes the structural design and analysis of the RXB and CRB. The applicant also addressed the capacity of sections, forces and moments at critical locations, and design checks, boundary conditions for each foundation model, soil stiffness conditions, and settlement evaluations. DSRS Section 3.8.5.II.4 provides review guidance on the evaluation of stiff and soft spots in the foundation soil to maximize the bending moments used in the design of mat foundations. In FSAR, Section 3.8.5.3.3 and Table 1.8-1 "Combined License Information Items", the applicant provided COL Item 3.8-3 for an applicant that references the NuScale Power Plant US460 standard design to identify local "stiff and soft spots" in the foundation soil and to address these in the design of foundations, as necessary.

The applicant also employed the ANSYS computational software to generate finite element models (FEMs) simulating the structural response under static and dynamic loads as described in FSAR Sections 3.7 and 3.8 for the design and analysis of the RXB and CRB foundations as appropriate. The foundations were modeled using solidshell (SOLSH190) elements for RXB and shell (SHELL181) elements for CRB, respectively. The soils were modeled using Soil Type 7, 9, and 11, along with Soil Type 7 with soil separation (soil separation case applies to the RXB only) in analysis, as applicable.

Based on the review, the staff determined that the applicant provided an appropriate level of information for the design and analysis procedure used for the Seismic Category I foundations. The staff also determined the use of the ANSYS FEM to design and analyze the RXB and CRB foundations to be acceptable because the ANSYS computer code is widely recognized in the industry and has sufficient history of use to demonstrate its suitability. The staff concludes that the applicant meets the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.1 Reactor Building Stability Analysis Model Description

FSAR Section 3.8.5.3.1 provides the uplift, sliding and overturning stability analysis model description for the RXB. The applicant used a static force equilibrium method to develop equations for each of the stability cases to determine the factor of safety (FOS). FSAR Table 3.8.5-2a contains the parameters used for the RXB stability analyses. The applicant took results from the RXB portion of the double building harmonic analysis and post-processed them. From this analysis, the applicant took transfer functions in three directions for each soil type of interest and extracted, interpolated, and convolved with input seismic motions to retrieve a time history of resultant forces and moments at the center of the RXB basemat. Then the applicant used these time histories to form the basis of the demand forces and are compared against resisting forces to establish the FOS.

The applicant indicated that the RXB has a center of gravity more than 20 ft below site grade elevation and that in accordance with ASCE 43-19 for a deeply embedded structure with a center of gravity below the site grade elevation on each perimeter wall, demonstration of sliding and overturing stability is not required. However, the applicant decided to perform a completeness calculation to demonstrate the FOS values are greater than or equal to required 1.1.

Based on the review, the staff finds the applicant's approach acceptable for the stability analyses model of the RXB. The staff determined that the applicant provided sufficient information to describe the modelling used for the RXB stability analyses and that the applicant's description meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.2 Control Building Stability Analysis Model Description

FSAR Section 3.8.5.3.2 provides the uplift, sliding and overturning stability analysis model description for the CRB conservatively assumed as a surface-founded structure. FSAR Table 3.8.5-6 contains the parameters used for the CRB stability analyses.

The applicant indicated that it performed a linear elastic analysis using a force equilibrium method similar to the RXB. The applicant further conducted a nonlinear transient analysis because the FOS calculated from the linear elastic analysis of load combination with the seismic demand did not meet an acceptable FOS required for sliding and overturning. The applicant considered a hybrid cracked case in its nonlinear model for load combination with the seismic demand. The applicant indicated that the non-linearity stems from the interface between the CRB base and the underlying soil, which is modeled as a frictional surface that allows both sliding and gap formation. The applicant used the Hilber-Huges-Taylor (HHT) implicit time integration method for the transient analysis and verified the convergence solution by repeating the analysis for selected cases with reduced time steps.

The applicant included COL Item 3.8-3 so an applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address them in the design, as necessary.

Based on the review, the staff finds the applicant's approach acceptable for the stability analyses model of the CRB. The staff determined that the applicant provided sufficient information to describe the modelling used for the CRB stability analyses and that applicant's description meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.4 Loads and Load Combinations

The staff reviewed loads and load combinations used for the foundations to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with the guidance in DSRS Acceptance Criterion 3.8.5.II.3.

FSAR Section 3.8.4 presents information for loads and load combinations used for the design of RXB and CRB, including the design of the foundations.

- FSAR, Section 3.8.4.3.3, "Earth Pressure(H)," describes that the embedded exterior walls of the buildings are subjected to lateral soil pressure loads induced by two types of loads, static soil pressure and soil-structure-interaction dynamic soil pressure. The staff noted that applicant correctly described the lateral soil pressure loads for both static and dynamic cases on embedded structures, including RXB.

- FSAR, Section 3.8.4.3.3, “Earth Pressure(H),” explains that the buoyant force is the upward pressure exerted on the bottom of the foundation during a saturated condition. The staff noted that applicant correctly described the buoyant force as equal to the volume of the building below grade multiplied by the density of water.

- FSAR, Section 3.8.4.3.22, “Other Loads,” describes construction loads, and operation with less than 6 NPMs:

- FSAR Section 3.8.5.9, “Construction Loads,” describes the construction loads on the basemats of the RXB and CRB. The RXB basemat will be poured in a very short time, and the main loads (the pool water, the NPMs) will be added after RXB construction is completed. The staff do not identify concerns about construction-induced settlement for the RXB and CRB basemats.

- The applicant performed a study to evaluate the dynamic effects of an earthquake when operating with less than 6 NPMs. FSAR Section 3.7.2.10, Section 3.7.2.10, “Sensitivity Studies on Soil Separation, Empty Dry Dock, and Modularity,” and Section 3.7.2.10.2, “Sensitivity Study Results,” report that the difference in results between operation with 6 NPMs and operation with fewer NPMs in place is small and within the capacity of the building design.

3.8.5.4.4.1 Stability Load Combinations

The applicant considered four load combinations for the assessment of stability for flotation, uplift, sliding, and overturning for RXB and CRB:

A. $D + H + W$
B. $D + H + E_S$
C. $D + H + (W_t \text{ OR } W_h)$
D. $D + F' + E_S$

The applicant defined the dead load of a structure as “D”; the weight and pressure of soils as “H”; the operating basis wind load as "W"; loads effects from SSE as “E_S”; loads generated by the design-basis tornado as “W_t”; loads generated by the design basis hurricane as “W_h”; and the buoyant force as “F’”.

The loads and load combinations used for the design of the RXB and CRB, including the design of the foundations, are discussed in FSAR Section 3.8.4. The applicant indicated that the OBE is established as one-third of the SSE. Therefore, in accordance with DSRS Acceptance Criterion 3.8.5.II.4. The OBE is not a design-basis ground motion for Seismic Category I structures and no specific analysis is required.

The applicant indicated that for the RXB, load combinations A and C do not need to be analyzed because wind loads in these combinations are bounded by the SSE in combination B. For the CRB, the applicant analyzed for load combination C and load combination D because the hurricane reactions have the highest amplitude. The applicant indicated that it did not consider soil weight and pressure around the basemat because the base of the CRB is shallow (1.5 m (5 ft)).

Thus, the applicant concluded that the load combinations B and D, as described above are bounding for the stability assessment for the RXB and load combinations C and D, as described above, for the CRB structures.

The staff reviewed the load combinations considered by the applicant against the DSRS acceptance criterion 3.8.5.II.3 and concludes that the load combinations used to check against sliding and overturning attributable to earthquakes, winds, tornados hurricanes and against flotation are bounding for the stability assessment. The stability load combinations are acceptable because they are in accordance with DSRS Acceptance Criterion 3.8.5.II.3.

3.8.5.4.4.2 Lateral Soil Force and Seismic Loads

FSAR Section 3.8.5.4.1, "Lateral Soil Force and Seismic Loads," states that the RXB is an embedded structure; therefore, surrounding soil imposes lateral soil pressures to the embedded structure. The applicant indicated that the CRB is not embedded in the soil, therefore the exterior walls are not subject to static and dynamic lateral soil pressure loads.

FSAR Table 3.8.5-6 provides input evaluation parameters for CRB including static coefficient of friction (CoF) of 0.58 and Kinetic CoF of 0.5 between concrete and underlying soil, which sets the basis on required minimum static CoF and kinetic CoF (Refer to Table 2.0-1, "Site Parameters," in ML24215A044). FSAR Section 2.5.4 indicates that the friction is defined between the concrete and clean gravel, gravel-sand mixture, or coarse sand with a friction angle of 30 degrees.

FSAR Section 3.8.4.3.3 describes the values of total maximum lateral soil pressure on walls and FSAR Section 3.8.5.4.1 provides the equation to determine the total lateral static effective soil forces on walls.

FSAR Table 3.8.5-2a lists the surcharge load as 12 kPa (250 psf), which is used in the design calculations for the RXB embedded walls subjected to static lateral soil pressure.

FSAR Section 3.8.5.4.1 calculates the lateral soil forces for the RXB. The forces on the RXB walls are calculated as 223,505 kN (50,246 kips) for the north and south walls and 150,127 kN (33,750kips) for the east and west walls. The staff performed an independent check of the calculations and determined that the applicant calculated the forces properly.

Based on the review of the parameters and independent check of the calculations, the staff concludes that the applicant correctly calculated the lateral soil forces and pressure and the seismic base reactions for RXB and CRB. The applicant also met DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.4.3 Effective Vertical Load

FSAR Section 3.8.5.4.2 describes the effective vertical load. The effective vertical load is an important stabilizing force for stability evaluations of the buildings. The applicant calculated the effective dead weights of the RXB and CRB by subtracting the dead weight of the buildings from the buoyancy forces and lists them as 630.8 MN (141,800 kips) and 86.7 MN (19,492 kips), respectively.

Based on the review, the staff finds the applicant's approach in accordance with DSRS Acceptance Criterion 3.8.5.II.4 since it used an acceptable method for determining the effective vertical load of RXB and CRB by subtracting the buoyancy loads from the total weight of the buildings.

3.8.5.4.4.4 Friction-Resistant Loads

FSAR Section 3.8.5.4.3 describes the friction-resistant loads. The friction-resistant loads consist of (1) total sliding frictional resistance on the foundation surface from effective vertical load and (2) friction forces resulting from at-rest earth pressures. Frictional resistance loads are considered to stabilize the structure against floating, sliding, and overturning loads since the RXB is a deeply embedded structure.

FSAR Section 3.8.5.4.3.1 describes the passive and active earth pressures and corresponding friction force. The applicant calculated the passive earth pressure coefficient K_p and passive pressure force acting on each wall; and the active earth pressure coefficient K_a and active pressure force acting on each wall.

FSAR Sections 3.8.5.4.3.2 and 3.8.5.4.3.3 describe overturning moment resistance in east-west direction and in north-south direction, respectively. FSAR Figure 3.8.5-4 and Figure 3.8.5-5 provide illustration of RXB for the east-west and north-south overturning moments with their associated moment arms, respectively.

FSAR Section 3.8.5.4.3.4 describes how factors of safety against flotation, sliding, and overturning are derived for RXB. FSAR Figure 3.8.5-1 through Figure 3.8.5-5 provide free body diagrams of the forces at play when establishing each FOS for the equation derived in FSAR Section 3.8.5.4.3.4 for flotation, sliding, or overturning. Several terms related to demand come from the seismic results covered in FSAR Section 3.7.2.

The staff reviewed the information on how to determine friction-resistant loads and the approaches to derive the factors of safety for the RXB. Based on the review, the staff finds the applicant’s description in the FSAR acceptable for describing the friction-resistant loads for RXB because (1) adequate consideration has been given to each frictional resistance scenario, including resistance loads for flotation, sliding, or overturning, and (2) the equations of factors of safety against flotation, sliding, and overturning are reasonably established based on resistance capacities and load demands. The applicant’s description also met DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.5 Results Compared with Structural Acceptance Criteria

The staff reviewed the structural acceptance criteria used for the foundations to ensure they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.1 Reactor Building Stability

In FSAR Section 3.8.5.5, the applicant used a static force equilibrium method to determine factor of safety for the RXB against overturning, sliding, and uplift.

The applicant calculated FOS for RXB uplift for a flooding event acting simultaneously with the maximum vertical seismic force. The staff noted that the applicant computed a FOS of 1.25, which complies with the minimum FOS of 1.1 required.

For the RXB sliding stability calculation, the applicant introduced a scaling factor, Ci, related to passive and active pressure from soil. This factor is iterated upon until the minimum factor of safety across all fields investigated is equal to the acceptable value of 1.1. The FSAR Table 3.8.5-13 contains the RXB sliding factors of safety for every seismic/soil configuration. The staff noted that the applicant used a passive and active factor, ci of 0.29, resulting in an FOS of 1.45, which complies with the minimum FOS of 1.1 required.

The applicant calculated the RXB overturning stability at every time step in each seismic event time history and evaluated each edge of the RXB basemat separately. FSAR Table 3.8.5-14 contains the RXB overturning moment factor of safety for every seismic/soil configuration. The staff noted that the applicant's resulting minimum FOS was 1.1 for a soil type with a passive and active pressure factor, Ci of 0.29, which complies with the minimum FOS of 1.1 required.

The staff reviewed FSAR Table 3.8.5-3 which contains a summary of the applicant's calculated factor of safety for the RXB against uplift (flotation), sliding and overturning. The staff noted that all results comply with the required minimum factor of safety of 1.1. Based on the review and comparison of the applicant stability analysis results with the structural acceptance criteria the staff finds that the RXB stability analysis meets applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.2 Control Building Stability

FSAR Section 3.8.5.5.2 provides the uplift, sliding, and overturning stability evaluation of the CRB. The applicant performed the evaluation using a linear elastic analysis and a nonlinear analysis.

The applicant calculated the CRB FOS for uplift using the resistance force of the building dead weight; and buoyancy and the peak vertical forces of the base reaction calculated from the hurricane and seismic analyses as driving forces. The staff reviewed FSAR Table 3.8.5-15 which contains the CRB uplift calculated FOS and noted that all FOS complies with the minimum FOS of 1.1 required.

The applicant performed a nonlinear transient analysis for the calculation of FOS of sliding because for the stability analysis, performed with the force equilibrium method, the load combination with the seismic demand does not meet the minimum factor of safety of 1.1 against sliding. The CRB sliding FOS time histories are shown in FSAR Figure 3.8.5-16a, Figure 3.8.5-16b, and Figure 3.8.5-16c for seismic events of interest with the Soil Type 7, Soil Type11, and Soil Type 9, respectively. The applicant tabulated the sliding result in FSAR Table 3.8.5-17, where the maximum absolute sliding from the nonlinear transient analysis is 33 mm (1.3 inches). The applicant considered this value to be acceptable given the level of conservatism in the analyses and the distance of the CRB to the nearby SC-II structure. Based on its review, the staff finds the applicant's approach acceptable because the applicant performed detailed nonlinear sliding analyses which would provide more realistic results for CRB. The staff also reviewed the tabulated results in FSAR Table 3.8.5-17 and confirmed that the results would not cause structural damage to CRB structure due to its sliding.

For the CRB overturning stability the applicant indicated that overturning is not a concern under hurricane load. The applicant stated that the overturning stability is further analyzed through nonlinear transient analyses for the load combination with seismic load.

The staff reviewed response to audit question 3.8.5-8 (ML24346A142) and the applicant referenced report EC-103147 “Stability Analysis of the SC-I Category Control Building Structure” where the applicant provided results of the stability analysis of the SC-I structures including the overturning stability analyzed through a non-linear transient analysis for all load combinations. The staff confirmed that the non-linear transient results for overturing stability are negligible as shown in FSAR Table 3.8.5-17 with a maximum non-linear transient result of 0.8 mm (0.03 inches) for vertical displacement.

Based on the review and comparison of the applicant stability analysis results with the structural acceptance criteria the staff finds that the CRB stability analysis meets applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.3 Average Bearing Pressure Approach

FSAR Section 3.8.5.6, “Bearing Pressure Approach,” describes the average bearing pressure results and the bearing pressures along the edges. The applicant calculated the mean RXB bearing pressures from SOLID185 elements forming the soil layers under the basemats and by dividing the sum of nodal forces in vertical direction under the basemats by the corresponding areas. The applicant calculated mean bearing pressures under CRB similarly but using nodes at the solid-structure interface beneath the basemat as the CRB basemat used contact elements to connect the soil and the backfill. The total maximum dynamic bearing pressure (static+ dynamic) values are 33.6 ksf for the RXB and 25.1 ksf for the CRB. The applicant used three rows of elements to define the areas for the calculation of toe pressure. FSAR Tables 3.8.5-4 and 3.8.5-5 list the average bearing and toe pressure values under the RXB and CRB basemats, respectively.

Based on its review, the staff finds the applicant’s approach acceptable, since it is appropriately formulated as described in DSRS Acceptance Criterion 3.8.5.II.4.N.

3.8.5.4.5.4 Settlement

FSAR Section 3.8.5.7, “Settlement,” describes the foundation settlements, including the approach and results. The applicant used a large-scale ANSYS FEM to determine the effect of foundation differential movements of the RXB and RWB comprising of uncracked and cracked structural members, referred to as hybrid static double building (DB) model. The applicant used Soil Type 11, the soft soil profile from the soil libraries, to maximize the effect of the differential movements and further reduced the stiffness of soil by 50 percent to amplify the effect of differential movements or settlements. The applicant applied the 50 percent reduction in soil stiffness to the areas below the basemats and extended it to the entire free-field soil model. The staff reviewed responses to audit questions 3.8.5-9 (ML24215A041) and 3.8.5-10 (ML24215A043). As part of the responses, the applicant referenced various calculation reports that the staff reviewed. During its review of reference EC-112976-0 “Differential Settlement Analysis of the Double Building Model” the staff confirmed that the applicant performed calculations that considered a 50-percent reduction of the soft-soil profile and performed spot checks on the calculations and methodology used. The staff noted that 2.4 m (8.0 ft) thick RXB mat foundation and 1.5 m (5.0 ft) thick CRB mat foundation are modeled with single layer of solid-shell (SOLSH190) elements and of shell (SHELL181) elements, respectively. To address staff audit concerns regarding the modelling adequacy for use of single layer of solid-shell or shell element for basemat, the applicant performed a mesh-density evaluation for the RXB and

CRB foundation to test the impact of a multi-layered solid-shell element SOLSH190 in their basemat deformation. The applicant modeled mat foundations from one to four elements throughout the thickness. The staff reviewed reference EC-151256 “RXB and CRB Basemat Element Selection and Convergence Evaluation” and noted that the difference in vertical displacement between the RXB static models was less than one percentage difference for the CRB statics models. The staff noted that the calculated vertical displacement was not significantly affected by adding multi layers of elements, and that the differential settlement result is not expected to be significantly affected, considering that under the same load and soil conditions, the vertical settlement will be greater than or equal to the differential settlement.

The applicant calculated differential settlement using the static load combination of dead, live, hydrostatic, and effective earth pressure. In addition, out of conservative considerations, the applicant ignored buoyancy forces in settlement analysis and defining the load combination as follows:

$$U = D + F + L + H$$

where U is total load, D is dead load, F is the hydrostatic loads that stem from the RXB pool, L is live load, and H is the effective earth pressure with surcharge load excluding hydrostatic loads. Because the CRB is surface founded the load combination used for its analysis is D + L. The applicant modeled the lateral effective earth pressure acting on the RXB.

FSAR Table 2.0-1 provides the selected site parameters appropriate for the design. FSAR Tables 3.8.5-7 and 3.8.5-8 list displacement, differential settlement values, and tilt in inches per 50 ft for a set of nodes selected on the RXB basemat and FSAR Tables 3.8.5-11 and 3.8.5-12 list applicable settlement results for the CRB. The maximum vertical displacement for the RXB and CRB are 37.6 mm (1.48 in) and 19.6 mm (0.77 in), respectively.

Based on its review of the information submitted by the applicant, the staff finds the applicant’s approach is acceptable because the staff confirmed that the applicant performed calculations that considered a 50-percent reduction of the soft-soil profile (Soil Type 11) stiffness values to conservatively determine the static demand forces for the RXB and CRB foundation designs and determine the maximum differential settlements within each building basemat. The staff confirmed that all settlement values resulting from the analysis are bounded by FSAR Table 2.0-1 which provides the selected site parameters appropriate for the design. The applicant’s responses and evaluations also meet DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.5.5 Thermal Loads

The staff reviewed FSAR Section 3.8.5.8 to ensure that it meets the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and is in accordance with DSRS Acceptance Criterion 3.8.5.II.4.B.

In FSAR Section 3.8.5.8 the applicant stated that in design of reinforced concrete members including basemat, the design demands from thermal effects due to accident temperature were not directly included in design load combinations. Instead, the applicant considered thermal effects by calculating the capacity of concrete sections by limiting the “usable” axial and bending strains to the allowable strains reduced by the thermal strains. The applicant indicated that the thermal forces and moments are greatly reduced or completely relieved with the progress of concrete cracking and reinforcement yielding.

Based on its review, the staff finds the applicant's description of thermal loads acceptable, since they are self-relieving because of concrete cracking and reinforcement yielding. Concrete cracks act as release points for the built-up stress, therefore reducing the magnitude of internal forces and moments. If thermal forces cause significant stress, the reinforcement can yield, absorbing some of the stress and reducing the overall forces and moments in the structure. Thus, the application meets DSRS Acceptance Criterion 3.8.5.II.4.B and is therefore acceptable.

3.8.5.4.5.6 Construction Loads

The staff reviewed the construction loads induced by the proposed construction sequence and by the differential settlements of the soil under and to the sides of the structures for the foundations to ensure they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.3.

In FSAR Section 3.8.5.9, "Construction Loads," the applicant stated that the main loads (the reactor pool and the NPMs) will be added after the RXB construction is completed. Therefore, the applicant did not consider construction-induced settlement. Accordingly, the RXB basemat design did not consider the loads induced by construction. The CRB basemat is smaller than the RXB basemat, and the concrete will be poured after the RXB basemat in the construction sequence.

The staff finds that the applicant's description stating that the main loads will be added after the completion of the RXB construction is acceptable. The staff also agrees that any loads induced by the construction sequence will be negligible since the main loads will be added after the completion of the RXB construction. Similarly, loads induced by the construction sequence will be negligible in the design of the CRB basemat because it is smaller than the RXB basemat, and the loads will be added after the completion of the CRB construction. Therefore, the staff finds the applicant's conclusions acceptable since the main loads will be added after the completion of RXB and CRB construction, and thus the effects of construction loads are not a concern, which meets DSRS Acceptance Criterion 3.8.5.II.4.M.

3.8.5.4.5.7 Leak Detection

The staff reviewed the design details that prevents and monitor potential leakage from the pool and potential leakage into the RXB from ground water to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5; and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.4.O.

FSAR Section 3.8.5.10, "Leak Detection," describes the leak detection of pool and ground water into the RXB walls and foundation. Ground water has the potential to leak through the RXB exterior walls through microscopic concrete cracks at a very slow rate of less than 3.8 liters (1 gallon) per day. The applicant concluded that this leak would not be enough to cause an interior flood in any of the rooms that share an exterior wall. However, the plant's concrete maintenance specifications and dewatering system surrounding the RXB would effectively reduce ground water leakage.

FSAR Section 3.8.5.10, states, "A leak chase system is provided in the RXB basemat to detect any leakage from the reactor pool." FSAR Section 9.1.3. describes the pool leakage detection system and SER Section 9.1.3 provides the staff evaluation on the pool leakage detection system.

3.8.5.4.6 Materials, Quality Control, and Special Construction Techniques

The staff reviewed the material, quality control, and special construction techniques used for the foundations to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with the guidance in DSRS Acceptance Criterion 3.8.5.II.6.

In FSAR Section 3.8.4.6, the applicant describes the materials, quality control, and special construction techniques for the RXB and CRB, including the foundations. The staff reviewed the material, quality control, and special construction techniques in FSAR, Section 3.8.4.6, regarding their application to the RXB and CRB foundations. FSAR, Section 3.8.4.6, describes the principal construction materials for Seismic Category I structures as concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. FSAR Table 3.8.4-3, provides the material properties for materials considered for structural design and indicates that the minimum typical compressive strength of concrete is 34 MPa (5,000 psi) and 48.3 MPa (7,000psi) for the RXB roof slab and floor slabs. FSAR, Section 3.8.4.6.1.1, also states that the concrete ingredients are cement, aggregates, admixtures, and water. FSAR Sections 3.8.4.6 and 3.8.4.6.1.1 provide the applicable industrial codes and standards and RGs that the materials and quality control shall satisfy, and they specifically refer to ACI 349, ACI 301, and RG 1.142 for the design of Seismic Category I structures.

FSAR Section 3.8.4.6.1.2, states that the steel reinforcing bar material conforms to A615 Grade 60 or A706, Grade 60.

The staff finds the use of these materials, quality control, and special construction techniques in the design and construction of the foundations of the RXB and CRB to be in accordance with DSRS Acceptance Criterion 3.8.5.6. In SER Section 3.8.4, the staff evaluates the adequacy of materials, quality control, and special construction techniques of Seismic Category I structures in accordance with ACI 349 and RG 1.142. On this basis, the staff finds the material, quality control, and special construction techniques in FSAR Section 3.8.5.6, to be acceptable.

3.8.5.4.7 Testing and Inservice Inspection Requirements

The staff reviewed the testing and inservice surveillance requirements used for the foundations to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.7.

FSAR Section 3.8.5.12 refers to FSAR Section 3.8.4.7, for a description of the testing and inservice inspection requirements for the RXB and CRB foundations. The applicant stated that there is no testing or inservice surveillance beyond the quality control tests performed during construction, which is in accordance with ACI 349, and AISC N690. In FSAR Section 3.8.4-7 the applicant included COL Item 3.8-2, which states that an applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with 10 CFR 50.65 as discussed in RG 1.160, where monitoring is to include below-grade walls; ground water chemistry, if needed; base settlements; and differential displacements.

The staff reviewed FSAR, Sections 3.8.5-12 and 3.8.4.7, and concludes that the testing and in service surveillance requirements used for foundations are in accordance with 10 CFR 50.65 and RG 1.160, as addressed in DSRS Section 3.8.5.

3.8.5.5 Combined License Information Items

SER Table 3.8.5-1 lists COL information item numbers and descriptions related to the structural design of Seismic Category I foundations for RXB and CRB.

Table 3.8.5-1: NuScale COL Information Items for Section 3.8.5

Item No.	Description	FSAR Section
COL Item 3.8-2	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements, and differential displacements.	3.8.4.7
COL Item 3.8-3	An applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.	3.8.5.3.3

3.8.5.6 Conclusion

The staff concludes that the NuScale Power Plant US460 standard design's RXB and CRB foundations is acceptable and meets the regulatory requirements described in Section 3.8.5.3 of this SER.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Introduction

This section reviews design transients and methods of analysis for Seismic Category I components and supports, including both those designed as ASME BPV Code, Section III, Division 1, Class 1, 2, 3, or core supports and those not covered by the ASME BPV Code. This section also reviews the computer programs used in the design and analyses of Seismic Category I components and their supports.

The staff reviewed relevant information the FSAR, in accordance with SRP Section 3.9.1, Revision 4, "Special Topics for Mechanical Components," issued December 2016. The applicant's FSAR submittal for special topics for mechanical components is acceptable if the submittal meets the requirements, codes and standards, and regulatory guidance on the methods of analysis for Seismic Category I components and supports, including those designated as ASME BPV Code, Section III, Class 1, 2, 3, Subsection NG core support structures, Subsection NF for supports and those not covered by the ASME BPV Code. The staff reviewed the FSAR to ensure the applicant provided information on design transients for ASME BPV Code Class 1 and core support components and supports. Specific areas that the staff reviewed include the following:

- transients used in the design and fatigue analyses of all ASME BPV Code Class 1 and core support components, supports, and reactor internals

- identification and description of computer programs to be used in analyses of Seismic Category I ASME BPV Code and non-ASME BPV Code items

- the environmental conditions to which all safety-related components will be exposed over the life of the plant

3.9.1.2 Summary of Application

FSAR: FSAR Section 3.9.1.1, “Design Transients,” describes the design transients for each of five service or test conditions defined in ASME BPV Code, Section III, and the frequencies (number of cycles) for each transient assumed in the ASME BPV Code design and fatigue analyses of RCS Class 1 components, Auxiliary Class 1 components, RCS component supports, and reactor internals. The number of cycles assumed for each design transient was based on a 60-year design life. The ASME BPV Code, Section III, service level conditions the applicant considered include the following:

- Level A Service Conditions—(normal conditions)
- Level B Service Conditions—(upset conditions, incidents of moderate frequency)
- Level C Service Conditions—(emergency conditions, infrequent incidents)
- Level D Service Conditions—(faulted conditions, limiting faults)
- Testing Conditions—(primary-side, secondary-side, and containment hydrostatic tests)

FSAR Section 3.9.1.1, does not cover the seismic loading and other mechanical loading on each component. FSAR Section 3.9.3, “ASME BPV Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures,” describes seismic loading and other mechanical loading.

FSAR Section 3.9.1.2, “Computer Programs Used in Analyses,” identifies the computer programs that are used for static, dynamic, and hydraulic transient analyses of mechanical system components.

FSAR Section 3.9.1.3, “Experimental Stress Analysis,” states that experimental stress analysis is not used for the NuScale design.

FSAR Section 3.9.1.4, “Considerations for the Evaluation of Service Level D Condition,” indicates that analytical methods used to evaluate stresses for Seismic Category I systems and components subjected to Service Level D condition loading are described in Section 3.9.3.

ITAAC: There are no ITAAC associated with 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.

3.9.1.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, GDC 1, which 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

- 10 CFR Part 50, Appendix A, GDC 2, which requires, in part, that SSCs important to safety be designed to withstand seismic events without loss of capability to perform their safety functions

- 10 CFR Part 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary,” which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- 10 CFR Part 50, Appendix A, GDC 15, “Reactor Coolant System Design,” which requires that the RCS and associated auxiliary, control, and protection systems be designed 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)

- 10 CFR Part 50, Appendix B, Section III, as it relates to quality of design control

- 10 CFR Part 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics

The guidance in SRP Section 3.9.1 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections. In addition, the following guidance document provides acceptance criteria used to confirm that the requirements of 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR Part 50, Appendix B, have been adequately addressed:

- NUREG/CR-1677, “Piping Benchmark Problems,” Volumes I and II, issued August 1980

3.9.1.4 Technical Evaluation

The staff reviewed FSAR Section 3.9.1, “Special Topics for Mechanical Components,” in accordance with SRP Section 3.9.1.

Test conditions include primary side hydrostatic test, secondary side hydrostatic test, and containment hydrostatic test. On the basis that all these tests comply with ASME BPV Section III, the staff finds this acceptable.

3.9.1.4.1 Design Transients

The staff reviewed FSAR Section 3.9.1.1 to ensure that it meets the relevant requirements of GDC 1, 2, 14, and 15 and 10 CFR Part 50, Appendix S, in regard to including a complete list of transients to be used in the design and fatigue analysis of ASME BPV Code Class 1 and core support components, supports, and reactor internals within the RCPB. The design transients define thermal hydraulic conditions (i.e., pressure, temperature, and flow) for the NPM. Bounding thermal hydraulic design transients are defined for components of the RCPB. FSAR Table 3.9-1, “Summary of Design Transients,” lists the design transients by ASME service level and includes the number of occurrences or cycles for each design transient based on a plant life of 60 years.

The staff reviewed heatup transient and identified that three different heatup rates were presented. The lowest heatup rate was used for the stress calculation input and pressure and temperature limits report (PTLR) input. The highest heatup rate was presented in design bases documents. NuScale stated in the FSAR, "The maximum transient heatup rate is consistent with the thermal limitations of the module heatup system as described in Table 9.3.4-1." This FSAR statement defines the value for the heatup rate. The staff also identified that stress calculations will be revised with the correct heatup rate consistent with ITAAC and PTLR will be revised by the COL applicant as discussed in SER Chapter 5. The staff finds this acceptable.

FSAR Section 3.9.3 gives load combinations and their acceptance criteria for mechanical components and associated supports, and FSAR Section 3.12 gives the same for piping systems. The Service Level A and B transients are representative events that are expected to occur during plant operation. These transients are frequent enough to be evaluated for component cyclic behavior and equipment fatigue life, and the analyzed conditions are based on a conservative estimate of the frequency and magnitude of temperature and pressure changes. FSAR Table 3.9-1 contains the description of events and occurrence cycles.

GDC 1 requires, in part, that SSCs important to safety be designed to high quality standards commensurate with the safety function performed. Therefore, the transient conditions selected for equipment design evaluation are based on the conservative estimates of the magnitude and frequency of temperature and pressure transients resulting from various operating conditions that may occur in the plant.

GDC 14 and 15 require, in part, that certain SSCs be designed with sufficient margin to withstand postulated transients anticipated during the design life of the plant. In accordance with SRP Section 3.9.1, Section III.2, the staff compared information on similar and previously licensed applications with that in the FSAR. The staff finds that the transients identified by NuScale are similar to previously licensed LWRs.

In accordance with GDC 14 and 15, SSCs important to safety must be designed to have a low probability of abnormal leakage and to withstand operational occurrences (i.e., postulated transients anticipated during the design life of the plant).

The staff already evaluated in US600 DCA the design transients described in FSAR Section 3.9.1.1 and Table 3.9-1 meeting requirements of GDC1, 14 and 15 and documented in Section 3.9.1 of DCA FSER. The staff compared the transients of DCA with the transients of SDAA. The same type transients apply to SDAA. The design transient related to Density Wave Oscillation (DWO) to evaluate the mechanical components' integrity is addressed in Section 3.9.2 of this report. SER Section 5.4.1.4. addressed the thermal hydraulic aspect of the DWO transient.

3.9.1.4.2 Computer Programs Used in Analyses

The staff reviewed FSAR Section 3.9.1.2, to ensure that the relevant requirements of GDC 1 and 10 CFR Part 50, Appendix B, were met in regard to computer programs used by NuScale in dynamic and static analyses to determine the structural and functional integrity of Seismic Category I mechanical components, including mechanical loads, transients, stress, and deformations.

FSAR Section 3.9.1.2 briefly describes each computer program used in the design and analysis of the Seismic Category I structures and components. NuScale used ANSYS, AutoPIPE, and NRELAP5, and computer programs are described in FSAR Section 3.7 for seismic design: RspMatch2009, SAP2000, SASSI, SHAKE2000. The staff already reviewed these computer programs during DCA and documented in Section 3.9.1 of the US600 DCA FSER. The staff accepted all above-mentioned Codes except NRELAP5.

FSAR Section 3.9.1.2 indicates that NRELAP5 was used to generate thermal hydraulic boundary condition inputs for the ANSYS shot-term transient structural loads in TR-121517, Revision 1, "NuScale Power Module Short-Term Transient Analysis". Staff audited NuScale calculations related to inadvertent primary valve opening boundary conditions for dynamic analysis and HELB blowdown methodology and noted NuScale departs from its approach the staff previously reviewed for the DCA in the short-term load determination. For the NPM-20 design which is the subject of this review, the 100 percent power at nominal reactor operating pressure with no bias (uncertainty) was used instead of the more bounding RPV design pressure. The sensitivity analyses performed by the staff indicate approximately 8 percent higher RVV/RRV thrust loads using the RPV design pressure, and about 3 percent higher loads with the biased reactor operating pressure. However, the staff also notes that in previous NRC reviews, the staff has accepted this approach of using unbiased nominal operating conditions. The staff considers that the nominal thermal hydraulic condition, when combined with other loads in downstream calculations, can provide adequate conservatism in the short-term load determination. Hence, the updated NuScale SDAA short-term load approach using unbiased nominal operating pressure is acceptable.

With respect to NRELAP5, FSAR Section 3.9.1.2 indicates that the program is used for safety-related design and analysis calculations of single phase and two-phase systems, including DWO in the SG tubes. It further indicates that the use of NRELAP5 is appropriate to model the pertinent physics of transient or oscillating pressures, temperatures, and flows in the NPM. NRELAP5 can be used to develop transient inputs that are sufficiently conservative for use in structural analyses.

The staff reviewed the impacts of secondary instabilities in the SG design and addressed mechanical components' integrity due to DWO in Section 3.9.2 of this report.

3.9.1.4.3 Experimental Stress Analysis

The applicant stated that the NuScale Power Plant design does not use experimental stress analysis. Therefore, the relevant requirements of GDC 1, 14, and 15, and Appendix II to ASME BPV Code, Section III, Division I, specific to experimental stress analyses methods do not apply.

3.9.1.4.4 Considerations for the Evaluation of Service Level D Condition

This section evaluated the analytical method used by the applicant for the Seismic Category I systems and components subjected to large strain in the Service Level D condition loading. The applicant stated that FSAR Section 3.9.3 describes the analytical methods used to evaluate stresses for Seismic Category I systems and components subjected to Service Level D condition loading. The staff finds the applicant's statement acceptable because SER Section 3.9.3 evaluates Service Level D loading loads, loading combination, and stress limits for evaluation of ASME BPV Code Class 1, 2, and 3 components and their supports.

3.9.1.5 Combined License Information Items

SER Table 3.9.1-1 lists COL information item numbers and descriptions from FSAR Table 1.8-1.

Table 3.9.1-1: NuScale COL Information Item for Section 3.9.1

Item No.	Description	FSAR Section
COL Item 3.9-1	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific seismic analysis in accordance with Section 3.7.2. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the certified seismic design response spectra, the standard design of NuScale Power Module components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.	3.9

3.9.1.6 Conclusion

On the basis of the above evaluation, the staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components are acceptable and meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 14, and 15; 10 CFR Part 50, Appendix B; and 10 CFR Part 50, Appendix S; and the guidelines in SRP Section 3.9.1. SG structural integrity is assured during DWO as stated in Section 3.9.2 of this report and further in SER Section 5.4.1.4.2.

3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components

3.9.2.1 Introduction

This section of the SER evaluates the analytical methodologies, testing procedures, and dynamic analyses used by the applicant to ensure the structural and functional integrity of the piping systems, mechanical equipment, RVIs including the internal SG, and their supports under vibratory loadings, including those caused by fluid flow, short term transients, and postulated seismic events.

This section addresses six main areas of review:

(1) Piping vibration, thermal expansion, and dynamic effects testing (3.9.2.4.1)

(2) Seismic analysis and qualification of Seismic Category I mechanical equipment (3.9.2.4.2)

(3) Dynamic response analysis for RVIs and SGs under operational flow transients and steady state conditions (3.9.2.4.3)

- o Design and Operation Summary
- o Analytic Flow-Induced Vibration Evaluation
- o Forcing Function Methodologies and Assumed Flow Velocities
- o Structural Mode Shapes and Resonance Frequencies
- o Structural Damping

- Turbulent Buffeting (TB) Analysis
- Flutter and Galloping Susceptibility
- Vortex Shedding (VS) and Fluid-Elastic Instability (FEI) Susceptibility
- Acoustic Resonance (AR) Susceptibility
- Leakage Flow Instability (LFI) Susceptibility
- Density Wave Oscillation
- Benchmarking Testing

(4) Preoperational flow-induced vibration testing of RVIs and SGs (3.9.2.4.4)

- Flow Induced Vibration Testing of SGs in Test Facility 3
- Initial Startup Testing of NPM
- Inspections

(5) Dynamic system analysis of the RVIs and SGs under faulted (service level D) conditions (3.9.2.4.5)

- Seismic Analysis
- Short-Term Transient Analysis
- Stress Evaluation of RVIs and SGs

(6) Correlations of RVIs and SG vibration tests with analytical results (3.9.2.4.6)

3.9.2.2 Summary of Application

FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment,” presents criteria, testing, and dynamic analyses employed to ensure structural and functional integrity of piping systems, mechanical equipment, and reactor internals and their supports under dynamic and vibratory loading, including those due to fluid flow during normal plant operation, transient conditions, and postulated seismic events. The NuScale NPM includes an internal SG system which is also evaluated for structural and functional integrity.

FSAR Section 3.9.2.1, “Piping Vibration, Thermal Expansion, and Dynamic Effects” addresses the initial startup testing that is performed to verify that the vibrations and thermal expansion and contraction of the as-built piping systems are bounded by the design requirements.

TR-121353, Revision 2, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report” (Comprehensive Vibration Assessment Program (CVAP) analysis technical report), issued January 2025 (ML25023A215 (proprietary) and ML25023A214 (non-proprietary)), is referenced in FSAR Section 3.9.2.3, “Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady State Conditions,” and Section 3.9.2.4, “Flow-Induced Vibration Testing of Reactor Internals Before NuScale Power Module Operation.” In addition, the applicant has submitted technical report TR-121354, Revision 1, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report,” (MIP technical report), issued August 2024 (ML24222A529 (proprietary) and ML24222A528 (non-proprietary)). FSAR Section 14.2 describes the SG prototype testing (Test #65) and the NPM initial startup vibration testing (Test #102).

The CVAP technical report describes the screening procedures and provides the results of the flow-induced vibration (FIV) analyses of (1) RVIs and structures, (2) SG components, and (3) primary and secondary RCS piping, up to the NPM disconnect flange. Components with small margins of safety against FIV effects were identified for validation testing. The MIP technical report describes a SG mockup, called the Società Informazioni Esperienze Termoidrauliche (SIET) test facility-3 (TF-3), which was built and tested in accordance with TR-121354, Revision 1. The TF-3 was tested over a wide range of flow conditions to confirm that significant SG FIV caused by VS and FEI will not occur in the NPM. The initial startup testing for FIV effects on the prototype NPM will include a set of external pressure sensors to detect any unexpectedly strong FIV of the RVIs and SGs. Prototype NPM initial startup testing will also confirm there are no strong ARs in the containment system (CNTS) steam piping.

FSAR Section 3.9.2.5, “Dynamic System Analysis of the Reactor Internals under Service Level D Conditions,” and Appendix 3A, “Dynamic Structural Analysis of the NuScale Power Module,” describe the structural and dynamic analyses of the NPM. Dynamic analyses for ASME Service Level D events include SSE and blowdowns induced by pipe ruptures and inadvertent valve actuations. Appendix 3A, references Technical Reports TR-121515, Revision 2, “US460 NuScale Power Module Seismic Analysis,” (seismic analyses technical report) issued March 2025 (ML25066A255 (proprietary) and ML25066A254 (non-proprietary)) and TR-121517, Revision 1, “NuScale Power Module Short-Term Transient Analysis,” (short term transient analyses technical report) issued August 2024 (ML24243A010 (proprietary) and ML24243A009 (non-proprietary)), for additional details. The seismic analyses were performed in three phases. First, a finite element model of the RXB, including linearized models of the NPMs, was analyzed with ANSYS for several postulated earthquake loading time histories and several soil types. A nonlinear model of a single NPM was then analyzed using bounding time histories of accelerations at the support locations. Short term transient loads induced by blowdown events were simulated with NRELAP5 and ANSYS. NRELAP5 was used to compute boundary conditions from thermal hydraulic analyses which were applied to the ANSYS model to compute accelerations and loads. The calculated in-structure time histories from the seismic and short term transient NPM analyses were saved along with in-structure response spectra from the seismic analyses for subsequent ASME Service Level D stress analyses of RVIs and SG tubes and supports.

FSAR Section 3.9.6, “Correlations of Reactor Internals Vibration Tests with the Analytical Results,” only states that future testing results will be compared to previous analysis results. Any significant deviations would require re-analyses and reconciliation with the test results.

Technical Specifications: There are no GTS for this area of review

Technical Reports:

- TR121515, Revision 2, “US460 NuScale Power Module Seismic Analysis”
- TR-121517, Revision 1, “NuScale Power Module Short-Term Transient Analysis”
- TR-121353, Revision 2, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report”
- TR-121354, Revision 1, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report”

3.9.2.3 Regulatory Basis

The following relevant NRC regulatory requirements apply to this review:

- GDC 1, as it relates to the design, fabrication, erection, and testing of SSCs in accordance with the quality standards that are commensurate with the importance of the safety function to be performed

- GDC 2, as it relates to the ability of SSCs, without loss of capability to perform their safety functions, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads, and to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics

- GDC 4, as it relates to the protection of SSCs against dynamic effects, including those of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit

- GDC 14, as it relates to designing SSCs of the RCPB to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

- GDC 15, as it relates to designing the RCS with sufficient margin to assure that the RCPB is not exceeded during normal operating conditions, including AOOs

- Appendix B to 10 CFR Part 50, as it relates to the QA criteria for the dynamic testing and analysis of SSCs

- Appendix S to 10 CFR Part 50, as it relates to certain SSCs that must be designed to remain functional for an SSE

- 10 CFR Part 50.55a, as it relates to the design, fabrication, erection, and testing of SSCs in accordance with the quality standards that are commensurate with the importance of the safety function to be performed

SRP Section 3.9.2 lists the acceptance criteria adequate to meet the above requirements and review interfaces with other SRP sections. In addition, the following guidance documents provide general acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.20, Revision 4, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” issued July 2015, as it relates to the vibration analysis and testing methodologies of the RVIs

- RG 1.61, Revision 1, “Damping Values for Seismic Design of Nuclear Power Plants,” issued March 2007, as it relates to the damping values used for a dynamic analysis

- ASME OM-S/G-2000, “Standards and Guides for Operation of Nuclear Power Plants” (ASME Operation and Maintenance of Nuclear Power Plants Code Standards and Guides (OM Code), 2000 Edition), Part 3, “Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems,” and Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems," as they relate to guidance for test specifications, as endorsed by SRP 3.9.2

3.9.2.4 Technical Evaluation

3.9.2.4.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

FSAR Section 3.9.2.1, addresses the initial startup testing that is performed to verify that the vibrations and thermal expansion and contraction of the as-built piping systems are bounded by the design requirements. The piping systems in the initial startup testing program include (1) ASME BPV Code, Section III, Class 1, 2, and 3 piping systems, (2) high-energy piping systems inside Seismic Category I structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and (3) Seismic Category I portions of moderate-energy piping systems located outside of the containment.

FSAR Section 3.9.2.1, states that the vibration, thermal expansion, and dynamic effect elements of this test program are performed during preoperational testing and initial startup testing. The preoperational tests are performed to demonstrate that the piping system components meet functional design requirements and that piping dynamic effects are acceptable. If test acceptance criteria are not met, corrective actions (e.g., reanalyzing with as-built values) are implemented, and the systems are retested. The initial startup testing is performed after the reactor core is loaded into a reactor module. These tests determine that the vibration level and piping reactions to transient conditions are acceptable and are bounded by the analyses. If the vibration levels are not bounded, the evaluations use the vibration level from the testing as input to verify that the design is acceptable. FSAR Section 3.9.2.1 lists the initial startup tests that included in FSAR Section 14.2 to verify the piping systems are within the thermal expansion and vibration limits.

FSAR Section 3.9.2.1.1, "Piping Vibration Details," states preoperational tests and initial startup tests demonstrate that piping systems withstand vibrations resulting from normal operation, including anticipated operational occurrences. If excessive vibration is observed that is outside the bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed. Vibration test specifications are developed in accordance with ASME Operation and Maintenance of Nuclear Power Plants, Division 2, 2017 Edition, Part 3. SRP Section 3.9.2, Revision 4, references the ASME OM Standards and Guides 2012 Edition. The NRC staff finds that the use of ASME OM Code, Division 2, 2017 Edition, is acceptable because the provisions for piping vibration and thermal expansion testing are equivalent.

FSAR Section 3.9.2.1, includes COL Item 3.9-2, for the COL applicant to complete an assessment of piping systems inside the RXB to determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. The COL applicant may select piping systems for the vibration testing using the piping vibration screening and analysis results of the CVAP. The staff finds that the COL item adequately addresses the assessment and selection of the piping system for vibration, thermal expansion, and dynamic effect testing during initial startup testing. Additionally, ASME OM Code, Division 2, 2017 Edition, Part 3, does not specify the criteria for selecting piping for vibration testing; therefore, considering the screening and analysis results of the CVAP for the selection of piping systems for vibration testing is an acceptable approach.

FSAR Section 3.9.2.1.1.1, "Main Steam Line Branch Piping Acoustic Resonance," addresses the concern of potential vibration or fatigue failure of main steamline branch piping due to flow-excited ARs. COL Item 3.9-3 addresses the detailed design of the main steam piping by the COL applicant, ensuring the detailed design of the MS line considers the phenomenon of AR and the piping vibration screening and analysis results of the CVAP. The staff finds that the COL item and the process used to complete the detailed design of the MS line to avoid AR is acceptable because COL Item 3.9-3 will ensure that the design of the piping systems will preclude significant ARs at pipe branches.

FSAR Section 3.9.2.1.2, “Piping Thermal Expansion Details,” states that the thermal expansion testing verifies that the design of the piping systems tested prevents constrained thermal contraction and expansion during normal operation. In addition, the tests verify that the component supports can accommodate the expansion of the piping during normal operation. FSAR Section 14.2, describes selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and startup testing will be made in accordance with ASME OM Code, Division 3, 2017 Edition, Part 7. The staff finds that performing the piping thermal expansion testing according to OM Code, Division 3, 2017 Edition, Part 7, is acceptable because this meets the SRP guidance. The initial startup testing provides adequate assurance that the piping and piping restraints of the tested systems can expand without obstruction and within design limits and therefore can withstand thermal effects during normal and transient operating conditions.

3.9.2.4.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

FSAR Section 3.9.2.2, “Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment,” references FSAR Section 3.7; Section 3.10, “Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment”; and Section 3.12. The corresponding sections of this SER include the review of these FSAR sections.

3.9.2.4.3 Dynamic Response Analysis of Reactor Vessel Internals and Steam Generators under Operational Flow Transients and Steady State Conditions

3.9.2.4.3.1 Design and Operation Summary

FSAR Chapter 1 describes the overall plant design, including the NPM, and Section 3.9.5, “Reactor Vessel Internals,” describes the RVI components. Aspects of the design that are relevant to FIV are summarized here. The NPM comprises a reactor core, pressurizer, and two integral once-through helical coil SGs within a cylindrical RPV, which is housed in a cylindrical steel CNV. The NPM operates with natural circulation primary coolant flow, which is much slower than in existing PWRs, reducing the strength of flow-induced forces compared to those in a typical PWR. The NPM rests in a reactor pool of water that acts as a heat sink and allows for passive operation (i.e., pumps are not used to circulate or inject coolant) and passive safety systems (i.e., DHRS and ECCS). Since the NPM has no pumps, there are no pump dynamic forces or inlet flow jets that impinge on reactor components. A power plant comprises up to a maximum of six NPMs.

The NuScale RVI is a first-of-its-kind design and is, therefore, classified as a prototype in accordance with RG 1.20. Following the NuScale RVI qualification as a valid prototype, future NPMs will be considered limited prototype or non-prototypes per RG 1.20. A single NPM is smaller than currently operating PWRs, and outputs power up to 250 megawatts thermal. Unlike traditional PWRs with forced primary coolant circulation, the core flow rate is proportional to the plant’s power. The SGs are integral to the NPM and therefore are evaluated for FIV along with RVIs. Also, all piping systems and valves, including those outside the CNV and up to the NPM disconnect flange, are evaluated for FIV effects. FIV effects on the RVIs, SG, and piping systems and valves are evaluated during normal operation, decay heat removal, and emergency core cooling conditions. Although the RVIs will experience worst-case FIV loads during normal operation, the main steam lines and isolation valves may experience stronger FIV loads during ECCS or DHRS operation.

The RPV is mounted within the steel CNV, which operates in a large pool of water. The CNV also contains auxiliary piping, including the chemical and volume control system (CVCS) and piping connection to the DHRS. The RPV is constructed of three sections: the head, upper, and lower sections, with the head welded to the upper section and a flanged bolted connection between the upper and lower sections. Several small RPV upper head penetrations accommodate the pressurizer spray, reactor vent valves (RVVs), reactor safety valves (RSVs), and in-core instrumentation. The CRDMs are mounted on top of the RPV with rods extending downward into the RPV.

The RVIs comprise a core support assembly (CSA) and hot-leg riser system. A lower riser assembly (LRA) rests on the CSA. The upper riser assembly (URA) is suspended from the upper riser hanger plate by the control rod drive (CRD) shaft sleeves and the bottom rests on the lower riser along a mating section and is secured against relative radial motion with pins. A bellows is included in the URA near the URA/LRA interface to allow for small relative movement between the upper and lower risers (primarily caused by thermal expansion) and to minimize the likelihood of significant leakage flow between the hot (inner) and cold (outer) legs.

The dome of the RPV houses the pressurizer system. The primary coolant turns downward below the pressurizer baffle plate and passes the SG tubes in the outer annulus of the RPV. Pressure is regulated by a pair of heater bundles, which may be activated to increase pressure, and two spray nozzles connected to the CVCS, which provide subcooled water at the top of the pressurizer to reduce pressure. The nozzle flow rates are very low and do not generate significant flow-induced forces.

The upper and lower risers are welded assemblies. Internal circular support frames (CRD shaft supports) are attached to the risers to accommodate CRD shafts and in-core instrumentation guide tubes (ICIGTs), which are inserted into the top of the RPV and extend downward through the CRD shaft supports and into the core to monitor and control the reactor. The CRD shafts can move upward and downward, whereas the ICIGTs are stationary. Nominal clearances are specified between the hole boundaries in the shaft supports and the CRD and ICIGT structures.

The once-through SGs consist of two independent bundles of tubes within the annulus between the riser and the wall of the RPV. The tubes for each bundle are welded to tubesheets at two integral feed (about halfway along the RPV) and steam (near the top) plenums, thus forming a pressure boundary between the primary and secondary coolant. The tubes are held in place by arrays of tube support assemblies mounted on upper SG supports attached to the pressurizer baffle plate and interfaced with lower SG supports that are attached to the RPV. A series of set screws preload the inner most tube support hanging back strip against the upper riser shell. Nominal small clearances are specified between the tube supports and tubes, but during operation the tubes are expected to have tight contact with the supports due to thermal and hydraulic forces along with the set screw preloading.

A CVCS purifies the primary coolant as needed. CVCS injection piping protrudes through the RPV wall, passes through the downcomer and terminates in the URA above the core exit.

The reactor operates passively, with primary coolant flowing upward through the core and the lower and upper riser assemblies, then moving radially outward below the pressurizer baffle plate and then downward though the annulus between the riser and RPV wall over the SG tube array. After passing over the SG tubes and through the downcomer, the flow moves radially inward before proceeding upward through the core and riser assemblies again. The secondary coolant enters the bottom of the SG tubes, as preheated subcooled liquid, and travels upward opposite the primary coolant flow direction. As heat is transferred from the primary to the secondary coolant, the secondary coolant within the SG tubes boils and transitions to superheated steam, which then exits into a plenum and steam supply nozzles near the top of the RPV where it travels to the steam turbines. Because the primary flow is passive, the velocity is about 5 to 20 times slower than the flow in a traditional PWR. However, due to the prototype design and relatively smaller size of NuScale internal components compared to traditional PWRs, FIV still needs to be assessed.

The SG tubes also function in conjunction with the DHRS. The DHRS provides secondary side reactor cooling for non LOCAs when normal FW is not available. For DHRS operation, the FW and MS isolation and bypass valves are closed, and the DHRS valves are opened. Water/steam in the secondary loop circulates naturally through the DHRS in the reactor pool and SG loops inside the RPV. The DHRS condensers are connected to the two SG loops, rejecting heat to the water in the reactor pool. During DHRS operation the flow rates through the SG are lower than those during normal operation; therefore, DHRS operating conditions do not need SG FIV evaluation. However, flow over cavities and standpipes in the DHRS piping is evaluated for AR.

The ECCS provides primary side cooling and coolant inventory control for LOCAs. For ECCS operation, two sets of emergency core cooling valves are opened. The RVVs release the primary coolant in the RPV to the CNV, where it condenses on the inner walls. The reactor recirculation valves (RRVs) located above the core also open to allow natural circulation between the condensed water in the annulus, between the CNV and the RPV, and the water within the RPV. Because flow rates throughout the steady-state ECCS conditions are low, FIV loads are small. Also, because the duration of any initial transients is short, any induced alternating stresses do not occur for a significant number of cycles.

Inlet flow restrictors (IFRs) are tightly fitted into all SG tube inlets to add stability against possible density wave oscillation (DWO) behavior in the secondary coolant system. It is well known that DWO can occur in parallel SG channels/tubes filled with fluids at different states (see for example Oh, S., Kim, D., and Lee, J., “Prediction of Density Wave Oscillation in Helical Steam Generators Using the MARS-KS Code,” International Journal of Heat Mass Transfer, Volume 235, 2024, 126226 along with its many references to other papers). NPM-20 SG tubes are filled with subcooled liquid at their inlets, followed by a boiling boundary and a section of two-phase flow (steam and liquid), followed by the dryout location and a final column of superheated vapor at the outlet. There are no restrictors at the tube outlets. Small inlet mass flow oscillations can induce density waves in the two-phase region (which has a varying density throughout) which can sometimes generate a pressure drop in the vapor region that is out of phase with the inlet flow oscillations, reinforcing them. The mechanism is described by Oh, as well as Reyes, “A Semi-Empirical Correlation for the Onset of Density Wave Oscillations in a Helical Coil Steam Generator,” Nuclear Technology, Volume 210, Issue 5, 906-918.

DWO has a characteristic time cycle, usually on the order of tens of seconds. During DWO, flow in some tubes reverses while flow in neighboring tubes flows forward – this is called “incoherent DWO.” In severe cases with high amplitude oscillations the boiling water boundary can approach the inlet, potentially leading to cavitation as well as condensation-induced water hammer (CIWH) in the tube inlet region and in the FW plenum. In very rare cases the oscillations in all tubes are in-phase (all in reverse or all forward at any given instant in time) – this is "coherent DWO." Incoherent DWO is usually not evident in any of the usually monitored parameters (temperature, pressure), but coherent DWO will induce observable changes, allowing control systems to take action to mitigate it.

The NPM-20 IFRs are long rods with narrow circular center orifices which induce a large inlet pressure drop. Any pressure oscillation in a SG tube which pushes the subcooled liquid back toward the inlet will be resisted by the IFR pressure drop, stabilizing the system against DWO initiation. The IFRs also mitigate the strength/amplitude of DWO oscillations should they occur. NuScale acknowledges in FSAR Section 3.9.1.1.1 and in Section 4.2-21 (Transient A21 – Density Wave Oscillations) of ER-101144, "Pressure and Thermal Transient Definitions for Analysis of NSSS Components," Revision 3 (referenced in response to Audit Question A-5.4.1.3-3, item 16 (ML25013A243 (proprietary) and ML25013A242 (nonproprietary)) that DWO instabilities may occur in the SG at limited transient conditions and when the DHRS system is activated and the FW temperature can no longer be controlled. Following the initial SDAA submission NuScale submitted updated secondary coolant system operating conditions in Section 2.1 (Inputs) of EC-110662, "Primary and Secondary Steady State Parameters," Revision 2, along with a temperature-based approach for monitoring NPM operating time in DWO conditions in FSAR Section 5.4.1.3 with details in EC-174500, "DWO Approach Temperature Limit," Revision 1 in response to SDAA Audit Question DWO-SC-25 (ML25013A222 (proprietary) and ML25013A221 (nonproprietary)). NuScale also submitted a series of technical evaluations of the potential FIV-induced effects of operating at severe DWO conditions in response to Audit Questions A-3.9.2-26 (ML24346A148 (proprietary) and ML24346A147 (non-proprietary)) and A-3.9.2-34 (ML24346A158 (proprietary) and ML24346A157 (non-proprietary)). Finally, NuScale updated the SG Technical Specification to ensure tube integrity.

3.9.2.4.3.2 Analytic Flow-Induced Vibration Evaluation

The NRC staff based this evaluation of the applicant's FIV, RVI, and SG analyses on (1) the CVAP analysis and MIP technical reports and (2) an audit of the applicant's internal documents, drawings, and test data conducted from March 27, 2023, through August 31, 2024 (ML24211A089). The staff used the audits to assess the details of the analyses. The SER presents only significant aspects of the CVAP analysis and MIP technical reports and audits.

The applicant screened the following components for FIV:

- RVIs
- SG components
- primary and secondary coolant piping up to the NPM disconnect flanges

Based on the screenings, the applicant identified selected components for more detailed FIV evaluations. The staff finds the screening procedures to be acceptable because they are consistent with the guidance in ASME BPV Code, Appendix N, "Dynamic Analysis Methods," and the open literature.

The applicant evaluated the following components in more detail for FIV effects resulting from primary coolant flow:

- SGs

- tube support bars
- hanging backing strip and set screws
- SG tube support spacer
- lower SG support

- URA
  - upper riser shells and transition shell
  - upper riser bellows and bellows threaded limit rods
  - set screw assemblies
  - ICIGTs and riser level sensor GTs
  - CRD shaft
  - CRD shaft support
  - CRD shaft sleeve

- LRA
  - lower riser section
  - control rod assembly guide tubes (CRAGTs)
  - CRAGT support plate
  - ICIGT funnels and lower riser ICIGTs
  - upper core plate

- CSA
  - core barrel
  - upper support block assembly
  - CSA mounting brackets
  - reflector block
  - lower core plate
  - fuel pin interface

- Other RVIs
  - RCS injection RVI
  - pressurizer spray RVI
  - thermowells
  - component and instrument ports
  - ECCS valves

- Primary coolant piping
  - RCS injection line tee location
  - CNTS drain valve tee locations

16 small slots in the upper riser permit flow between the upper riser and SG primary coolant regions during DHRS conditions when the top of the riser is uncovered. As discussed later in this SE, the applicant evaluated the potential impacts of the riser holes on FIV.

The applicant evaluated the following components for FIV effects resulting from secondary coolant forward flow:

- Steam generator system (SGS) and CNTS steam piping, MS isolation valves (MSIVs)
- SG steam plenum
- DHRS steam and condensate piping
- SG tubes
- SG tube IFRs
- SGS pressure relief valve branch, CNTS FW drain valve branch

The NRC staff does not usually review SGs as part of an RVI CVAP. However, because the SG tubes are integral to the NuScale reactor module, the staff reviewed it for FIV. The staff finds that the components evaluated for FIV are reasonable and that it is unlikely that any other RVI are susceptible to FIV based on low-flow conditions or robust structural designs, or both.

The applicant addressed the following FIV mechanisms:

- TB – random flow turbulence driving structures into broad-band, usually small amplitude, vibration

- F/G – bluff bodies, like circular or rectangular cross sections, locking into cross-flow into large amplitude vibrations

- VS – vortices shed from the aft ends of objects, usually pipes, which lock-in to pipe structural modes of vibration, leading to large amplitude vibrations

- FEI – multiple adjacent tubes vibrating in various patterns in response to cross flow, leading to very large amplitude vibrations

- AR – flow over side branch openings locking in to acoustic modes in the side branch, leading to high amplitude acoustic pulsations which can in turn lead to strong vibrations

- LFI – flow through narrow gaps which generates oscillating forces which lock on to structural modes of vibration

These are the usual FIV mechanisms evaluated in a CVAP, and the NRC staff finds them to be acceptable. For TB, the applicant also assessed fatigue life and wear associated with intermittent contact and relative motion between adjacent components. All other FIV mechanisms are evaluated only for their potential to occur since they are associated with the “lock in” of structural or acoustic motion with a flow-induced excitation mechanism or instability. Below a so-called “critical flow velocity,” determined for each component, this lock-in cannot occur and the flow-induced forces are small. If, however, lock-in were to occur, rapid failure (days or weeks) of the associated SSC would be expected.

For much of its screening and analyses, the applicant relied heavily on the following references:

- ASME BPV Code, Section III, Nonmandatory Appendix N1300, “Flow-Induced Vibration of Tubes and Tube Banks”;

- a workbook by M.K. Au-Yang, “Flow-Induced Vibration of Power and Process on Plant Components: A Practical Workbook,” issued 2001;

- a book by R.D. Blevins, “Flow Induced Vibration,” 2nd Edition, issued 1990;

- NUREG/CR-6031, “Cavitation Guide for Control Valves,” by J.P. Tullis, issued 1993;

- a paper by S.S. Chen on FEI and VS in helical coil SG tubing (see S.S. Chen, “Tube Vibration in a Half-Scale Sector Model of a Helical Steam Generator,” *Journal of Sound and Vibration* 91(4), pages 539–569, issued 1983); and

- four papers on LFI by F. Inada referenced in TR-121353

  (1) Inada, F., “A Study on Leakage Flow Induced Vibration From Engineering Viewpoint,” PVP2015-45944, ASME 2015 Pressure Vessels and Piping Conference, Volume 4: Fluid-Structure Interaction, July 19–23, 2015, American Society of Mechanical Engineers, New York, NY, 2015,

  (2) Inada, F. and S. Hayama, “A Study on Leakage-Flow-Induced Vibrations. Part 1: Fluid-Dynamic Forces and Moments Acting on the Walls of a Narrow Tapered Passage,” *Journal of Fluids and Structures*, issued in 1990: pages 4:395-412,

  (3) Inada, F. and S. Hayama, “A Study on Leakage-Flow-Induced Vibrations. Part 2: Stability Analysis and Experiments for Two-Degree-Of-Freedom Systems Combining Translational and Rotational Motions,” *Journal of Fluids and Structures*, issued in 1990: pages 4:413-428, and

  (4) Inada, F., “A Parameter Study of Leakage-Flow-Induced Vibrations,” Proceedings of the ASME 2009 Pressure Vessels and Piping Division Conference, July 26–30, 2009, American Society of Mechanical Engineers, New York, NY, issued in 2009.

Unlike previous applicants that have submitted a comprehensive scale model or full-scale plant test data, operating history of a similar design, or all of the above, NuScale has performed less extensive benchmarking to date to substantiate its analysis procedures. The applicant evaluated each FIV mechanism for a given component using a combination of the following:

- forcing function methodologies (from the ASME BPV Code or Au-Yang’s workbook)

- assumed flow velocities (from computational fluid dynamics (CFD) or bulk flow estimates)

- structural cross sections and lengths, mode shapes, and lowest resonance frequencies (from ANSYS FEAs), used to estimate critical velocities for each FIV mechanism

- assumed structural damping

The flow velocities, along with the forcing function methodologies, are required to estimate the flow-induced forces. The faster the flow, the higher the forces. The structural dimensions, boundary conditions, and material properties dictate the shapes and resonance frequencies of modes of vibration, and therefore the structural response functions. FIV from TB is computed by multiplying the estimated flow-induced forces by the structural response functions.

FIV mechanisms that involve flow instabilities and possible lock-in with acoustic or structural resonances are evaluated using criteria in the ASME BPV Code and the Inada references. These criteria generally combine the coincidence of flow-induced forcing and structural or acoustic response frequencies and the structural or acoustic damping. If the force and response frequencies coincide and damping is small, then lock-in and strong vibration or sound can occur. The NRC staff's evaluations of the analysis methodologies for each FIV mechanism are described below.

3.9.2.4.3.3 Forcing Function Methodologies and Assumed Flow Velocities

The applicant selected TB empirical forcing functions that are most appropriate for the flow and geometries of a given component, such as annular flow for the risers and axial and cross-flow over long beamlike structures (like the CRD shafts and ICIGTs). These forcing function definitions are scaled with geometric and flow variables, such as peak velocity at the center of an annulus flow, and the height of the flow profile. Therefore, flow velocity estimates were needed to compute actual forces and were calculated using the results of the CFD thermal-hydraulic analysis. The CFD calculations were over the full primary coolant region and are based on assumed reactor core and SG power density and loss coefficients. Therefore, spatial variations of the flow through the core and SG were not computed (only bulk velocities are available for those regions). CFD grid refinement studies verified flow velocity convergence throughout the primary coolant flow path. The CFD solution for the highest reactor power and flow conditions was processed to compute average and maximum velocities over several critical cross sections near the components evaluated for FIV. The applicant used the average velocities from its CFD analyses over the cross sections for the TB analyses and VS evaluations of the RVIs. The applicant estimated gap flow velocities for the SG TB, SG FEI and VS analyses based on the geometric blockage of the tubes and the bulk velocity.

All velocities used by the applicant for TB analyses may not be conservative, given the lack of detailed resolution in the CFD models. The applicant's assumptions regarding the location of peak velocity for some components may also be nonconservative. However, other aspects of the applicant's TB forcing function modeling approach, particularly with the parameters chosen in the empirical models (e.g., convective velocities and correlation lengths), are conservative. Given the large margin against TB-induced vibration (due to very low primary coolant flows), it is unlikely that any nonconservative biases in the applicant's assumed peak velocities will lead to significant vibration-induced damage for TB.

Although the primary coolant flow is the main source of FIV in the NPM, turbulent secondary coolant flow will also drive the inner walls of the SG tubes. The secondary coolant enters the SG tubes as preheated water, transitions to boiling on its way to the steam headers, and exits as superheated steam. The applicant used simple turbulent pipe flow empirical models for these forces, but also conducted "separate effects" testing of the wall pressures in the SIET TF-1 test facility (FSAR Section 1.5.1.3, "Steam Generator Thermal-Hydraulic Performance Testing—Electrically Heated Facility"). Strong spectral peaks were observed in the wall pressure data measured in TF-1.

The applicant applied the forces measured in TF-1 to its models of the SG piping in the TF-2 test (FSAR Section 1.5.1.4, "Steam Generator Thermal-Hydraulic Performance Testing—Fluid-Heated Facility"), where tube vibration was measured in the presence of both primary and secondary flow. The calculated TF-2 strains using the TF-1 forces are lower than the TF-2 vibration measurements, showing that the internal forces observed in TF-1 testing do not induce significant vibration (details are provided in NuScale TR-121354). There are also no peaks visible in the TF-2 measurements that are indicative of strong internal flow excitation. Based on this combination of TF-1 and TF-2 measurements and FIV analyses the NRC staff finds that there is reasonable assurance that the secondary coolant flow will not cause adverse FIV effects on the SG tubes.

3.9.2.4.3.4 Structural Mode Shapes and Resonance Frequencies

The ANSYS software suite, which includes structural FE and CFD modeling tools, was used to estimate structural mode shapes and resonance frequencies of the NPM RVIs and SG. FSAR Section 3.9.1.2, states that ANSYS is a pre-verified and configuration-managed FEA program used in the design and analysis of safety related components. The NRC staff finds that NuScale's ANSYS models are acceptable because the applicant has demonstrated that its meshing procedures and spatial resolution, boundary condition assumptions, and fluid loading effects are appropriate and conservative.

External (primary coolant) fluid mass loading was not modeled explicitly; instead, it was assumed to be that of the volume displaced by a given structure, which is a reasonable bounding approximation per FE modeling practices. The secondary coolant mass densities were also added to those of the SG tubes to compute the in-service resonance frequencies, which is also bounding. The applicant modeled all RVIs as an assembly and confirmed the appropriateness of the meshing density with convergence studies. Some structures, like the ICIGTs and control rod drive system (CRDS), were modeled individually using assumed boundary conditions at adjacent structural locations. The NRC staff finds the structural FE modeling reasonable because in general, conservative boundary conditions were assumed for these individual models (leading to lower resonance frequencies, which is conservative when assessing lock-in FIV mechanisms).

3.9.2.4.3.5 Structural Damping

Damping of all RVIs is assumed to be less than or equal to 1 percent, which the staff finds acceptable as it is in accordance with RG 1.20. However, the applicant assumed 1.5 percent damping for the SG tube FEI analysis but did not provide validated test data to substantiate this increased damping. The higher damping is assumed to be caused by friction between the tubing and tube supports, which depends on the tightness of fit, which in turn depends on thermal expansion and operational loads on the tubes and tube supports at normal plant operating conditions. The higher assumed damping led to higher estimated margins against EI occurring in the SGs. However, NuScale performed testing in the SIET TF-3 facility in Summer and Fall 2024 and showed that VS and FEI do not occur in the TF-3 under tight SG tube to tube support conditions. Representative data from this testing were provided for staff review in October-December 2024 and a final test report summary docketed in January 2025 (ML25027A395 (proprietary) and ML25027A394 (non-proprietary)). Since no FEI was observed in the testing, the higher assumed damping for the initial screening calculations is irrelevant.

3.9.2.4.3.6 Turbulent Buffeting Analyses

The applicant evaluated TB-induced vibration of components with fundamental resonance frequencies below 200 Hz, which the NRC staff finds reasonable since the TB loading above 200 Hz is negligible. Since TB loads are spatially and temporally random, random forced response analysis methods are used with conservative estimates for convection velocity and turbulence integral length scales. The calculated vibration and alternating material stresses are very small due to the low primary coolant flow speeds.

Fatigue and wear due to impacts were estimated for components with nonnegligible TB-induced vibration amplitudes and, in particular, for cases with high relative motion between components and neighboring supports. These include the CRAGT on the CRAGT support, the CRD shaft impact on the alignment cone, the upper ICIGT impact on the second highest CRD shaft supports, and the lower ICIGT impact on the upper core plate. The SG tubes are also assessed for impact wear, but the steady pressure forces on the tubes are expected to maintain nearly constant contact between the SG tubes and tube supports, minimizing the number of impacts that occur over service life.

Peak relative vibration amplitudes were assumed to be 5 times the predicted root mean square amplitudes, which capture a statistically appropriate number of peak occurrences that the NRC staff finds reasonable, based on guidance in Au-Yang's workbook. Previous applicants have extensively referenced Au-Yang's workbook for the FIV analyses, and therefore, the NRC staff finds that the guidance in this reference is acceptable. The number of impacts is based on the average crossing frequency, estimated using well established methods. Worst case contact and estimates for all evaluated components are negligible.

In Section 2.3.3.1 of TR-121353 the applicant discusses the potential FIV effects of the 16 small upper riser slots which provide a flow path for boron redistribution. The slots are small enough to not affect the structural modes of the upper riser significantly. Also, although the slots introduce stress concentrations in the upper riser, the alternating stresses in the upper riser walls induced by TB are so small that the safety margin against the material fatigue endurance limit is not challenged. Finally, the turbulent jet flow through the riser hole may impinge on some SG tubes. However, the slot heights and inclination angles direct any through flow downward to minimize flow-induced loads on the SG tubes. The staff finds that the structural integrity of the riser will not be significantly affected because the holes are small and the safety margin against TB remains very high. Additionally, the NRC staff finds that the jet flow loads will not significantly affect the SG tubes because jet flow loads are low.

The NRC staff finds that the applicant's TB assessments of the NPM RVIs and SG are based on appropriate modeling procedures, assumptions, and inputs, and are reasonable and conservative. No significant TB-induced degradation of RVI or the SGs is expected.

3.9.2.4.3.7 Flutter and Galloping Susceptibility

The applicant examined the shapes and cross sections of any structure subjected to cross-flow and compared them to guidelines for avoiding F/G. These guidelines are well established in the open literature and are acceptable. All NuScale components have significant margin against F/G, and any bias errors in velocity estimation will not challenge the margins; therefore, the NRC staff finds the F/G analyses to be acceptable.

3.9.2.4.3.8 Vortex Shedding and Fluid Elastic Instability Susceptibility

Although there may be some risk of structural wear caused by TB, FIV risks are much higher for stronger mechanisms like VS and FEI. If the frequency of VS aligns with those of structural resonances and if the impedances of those resonances are small, lock-in can occur and cause significant vibration and damage. Structural impedance at resonance is related to the mass-damping parameter in ASME Code, Section III, Appendix N. In addition, if velocities are high enough to induce FEI in arrays of tubes (like the SG tubes), even higher vibrations and damage could occur. All components subjected to cross-flow were screened for susceptibility to VS. The only components that warrant additional evaluation against VS/lock-in are the lower regions of

SG tubes. All other components were designed to ensure that the VS frequencies are well below any structural resonance frequencies.

Only the lower SG tubes are subject to VS/lock-in since the primary coolant flows downward, and the lower tubes have no downstream structures to break up the shed vortices. However, all SG tubes may experience FEI at and above critical flow velocities. FEI is therefore evaluated throughout the SGs. The applicant acknowledged that these components need validation testing to ensure margin against these mechanisms, and reported the following margins in TR-121353, in accordance with ASME BPV Code, Appendix N:

- lower SG tube VS/lock-in: **{{** **}}** percent
- SG tube FEI: **{{** **}}** percent (steam region) and **{{** **}}** percent (feedwater region)

The margins are for primary coolant flow rates at 100 percent power where **{{** **}}** margin implies plant power would need to increase **{{** **}}** above 100 percent to induce FEI in the SG. The VS margins are negative, implying VS could occur near 100 percent power. However, NuScale and the NRC staff notes that the VS margins are based on conservative assumptions in ASME BPV Code, Appendix N, Criterion A, which assumes a Strouhal Number (fD/U, where f is frequency, D is tube diameter, and U is flow velocity) of 1.0. Experimental evidence supports a much lower Strouhal Number (0.3 is typical). The actual best estimate margin of safety for VS using a Strouhal Number of 0.3 is **{{** **}}** percent for the lower SG tubes.

The method used by the applicant for FEI assessment is consistent with those in the ASME BPV Code Appendix N. However, the analysis inputs to the method were not conservative. The assumed damping for FEI analysis is 1.5 percent instead of the traditionally accepted 1.0 percent, thereby increasing the mass-damping parameter (increasing margin against FEI). Also, the so-called Connors constants used by NuScale to assess susceptibility to FEI (C=1.9, a=0.05) are not typical and deviate significantly from commonly accepted values (C=2.4, a=0.50 recommended in Section N-1331.3, “Suggested Inputs” of Appendix N of ASME BPV Code). The staff estimates that if 1 percent damping and the ASME recommended Connors constants were used the NPM SG would have no margin against FEI at 100 percent power.

Therefore, some form of testing was needed to confirm that FEI will not occur in the NPM. Testing was performed in the SIET TF-3 facility in Summer and Fall 2024 at flow conditions spanning very low to very high power (well above 100 percent equivalent NPM power). The staff examined preliminary test data both on-site at SIET in October 2024 and at an in-person audit at NuScale October 22-24, 2024. The staff concluded that the test facility reasonably represented the flow-induced vibration behavior of an NPM SG. In particular the tube-to-tube support connections were tight during flow testing and the corresponding damping of tube modes was small (on the order of 1 percent which is consistent with RG 1.20 Revision 4 guidance). Therefore, any differences between the TF-3 tube support design and that of the NPM-20 have no impact on the TF-3 test results (both are expected to be tight fitting). A preliminary data analysis report was provided in December 2024 and confirms that neither VS nor FEI occurred in the testing. A summary of the final report was submitted in January 2025 and further confirms that VS and FEI are not expected to occur in the NPM.

3.9.2.4.3.9 Acoustic Resonance Susceptibility

AR issues in nuclear power plants are usually associated with flow instabilities that form over side openings in the pipe flow. The fundamental acoustic modes in valve standpipes are the most commonly excited resonances. ARs have occurred in existing nuclear power plants and have led to extensive damage to valves and RVIs, particularly in boiling-water reactors. The flow instabilities occur when a half or full wavelength of the vortices shed from the leading edge of a side branch and coincides with the diameter of the opening. The first order (half-wavelength) instability is strongest and is most likely to lock in to any acoustic modes within the side branch. However, strong second order (full-wavelength) instabilities can induce damage to the valve components and other RVIs.

The applicant has evaluated the following piping and valve components for susceptibility to the first and second order AR, including components in the CNTS. Analysis margins against the first order AR are shown in parentheses for each component.

- RCS injection line to ECCS reset lines (**{{ }}** percent)
- CNTS RPV high point degasification drain valve branch (**{{ }}** percent)
- CNTS FW drain valve branches (**{{ }}** percent)
- SG system pressure-relief valve branches (**{{ }}** percent)
- DHRS condensate line to SGS feedwater line (**{{ }}** percent)

Due to adherence to the best design practices for AR avoidance, including rounding of the cavity upstream edges where possible, no NuScale piping or valve components are expected to experience first-order instability AR at full plant power conditions (all have more than 100 percent margin). However, these components might experience second-order instability AR at less than the full plant power level:

- DHRS condensate line to SGS feedwater line with **{{ }}** percent margin
- CNTS RPV high point degasification drain valve branch with **{{ }}** percent margin

Two locations evaluated in the DCA with initially low AR margins were the CNTS MS branch connection to the DHRS steam piping and the DHRS MS drain. Flow disruptors will be installed at the leading edges of the side branch openings at these locations to mitigate possible AR. As noted in FSAR Table 14.2-102, “NuScale Power Module Vibration Test # 102,” these locations will be monitored during the initial startup testing (to ensure that the vibrations at these plant power levels are not excessive). The NRC staff concluded that the applicant’s AR analysis methods and calculations are based on validated methods, and there is margin against both first- and second-order AR. Components treated with leading edge spoilers will be tested during the initial startup. Therefore, the NRC staff finds that there is reasonable assurance that significant AR-induced vibration will not occur, and that if AR occurs it will be detected so that changes could be implemented to preclude damage.

In FSAR Section 3.9.5.1, the applicant evaluated the through holes in the upper riser for susceptibility to AR effects. The upward flow in the upper riser and the downward flow in the SG annulus pass over the holes, which could induce shear flow instabilities and could potentially generate appreciable pressure pulsations within the primary coolant. However, the pressure difference between the upper riser and SG annulus drives a modest amount of flow through the hole, which eliminates the possibility of flow instabilities. Nevertheless, the applicant also compared the possible range of flow instability frequencies to those of acoustic modes within the upper riser coolant. The frequencies are far apart, eliminating the possibility of a flow instability driving an AR. The NRC staff conducted an audit during the NuScale DCA review of the applicant's evaluations (see audit report ML20160A247) and found that the riser holes would not cause AR because there is flow through the holes and the flow instability frequencies and the upper riser acoustic frequencies are well separated. This conclusion is still valid for the SDAA.

3.9.2.4.3.10 Leakage Flow Instability Susceptibility

Fluid-dynamic forces induced by leakage flow in the gaps between a structure and an external passage can couple with translational and rotational modes of the structure, sometimes to the point where self-excitation or lock-in occurs. Self-excited vibration amplitudes can be very high and cause contact between the structure and passage. Over time, repeated contact can cause wear and/or material fatigue damage. Damaging LFI has been observed previously in commercial nuclear reactors (M.P. Paidoussis, "Real-life Experiences with Flow-Induced Vibration," *Journal of Fluids and Structures,* Volume 22, pages 741–755, 2006), and design guidance has been developed for its avoidance (T.M. Mulcahy, "A Review of Leakage Flow Induced Vibrations of Reactor Components," Argonne National Laboratory Report ANL-83-43, May 1983).

The applicant has designed its components using best practices for LFI avoidance. In particular, there are no diverging passages between components, nor are sudden structural expansions located at the entry to a passage. Also, most components with leakage flow paths have very low-pressure differentials to ensure that leakage flow rates are small. As with the other instability mechanisms investigated by the applicant, a critical flow velocity is estimated and compared to the localized velocity at full plant power conditions. Margin is based on the ratio of the critical to localized velocity.

The applicant evaluated the following RVI components for LFI using the methodology defined in the TR-121353 references (Inada, 1990, 1990, 2015):

- CRD shafts adjacent to all through holes in surrounding support structures
- CRD shaft sleeve
- ICIGT adjacent to all through holes in surrounding support structures

The NRC staff evaluated the LFI evaluation procedures in the applicant's cited references and performed sample confirmatory calculations. The NRC staff finds that the procedures are reasonable and validated against test data, and the NRC staff's confirmatory calculations are consistent with the applicant's calculations (provided during a 2019 audit for the NuScale DCA review; ML19340A015) showing that NuScale's methods are valid and reasonable. Therefore, SDAA LFI calculations using these same methods are also reasonable.

The LFI evaluation methodology requires knowledge of the pressure difference across a passage and the loss coefficients for flow into and out of a passage. The pressure differences were estimated from the applicant's CFD analyses. The loss coefficients were estimated using standard thermal-hydraulic methods. Structural damping was assumed to be 1 percent, which is consistent with RG 1.20. The applicant also conservatively assumed a slightly diverging **{{ }}** annular gap area increase in all passageways to account for manufacturing tolerance uncertainties (even though this is unlikely to occur). Critical velocity was estimated as the point where total effective damping becomes negative (where LFI effects cancel the 1 percent structural damping). All components have more than 100 percent margin against LFI. Since there is significant margin, there is no need for testing prior to the initial startup.

Despite the high estimated margins against LFI for forward flow, pressure sensors will monitor the acoustic field of the RPV during the initial startup testing of the first NPM reactor to ensure that no unexpectedly high vibrations occur due to LFI or any other FIV mechanism (per FSAR Table 14.2-102, "Test # 102 NuScale Power Module Vibration" and Section 6.0, "Initial Startup Measurement Testing" of TR-121354). The NRC staff concluded that the applicant's LFI analysis methods and calculations are based on validated methods, there is significant estimated margin against LFI, and those components are part of the post-initial startup inspection plan. Therefore, the NRC staff finds that there is reasonable assurance of no significant LFI-induced vibration and structural damage for the life of an NPM.

3.9.2.4.3.11 Density Wave Oscillation Instability

A DWO instability in the secondary coolant within the SG tubes would lead to slow oscillations of the boundaries between the inlet subcooled liquid and the two-phase region and outlet steam. For mild small amplitude DWO instability, some of the subcooled liquid in the lower SG tubes will flow backwards through the IFRs with no significant thermal or structural loading. In severe high amplitude DWO conditions however, the boiling boundary, normally well above the SG tube inlets, and even steam could flow backwards towards the IFRs, leading to potentially strong and sudden transient loads. NuScale has specified the primary and secondary steady state parameters to minimize the likelihood of DWO onset, as well as the possibility of severe DWO. During normal operation at all power levels Figure 4-9, **{{ }}** of EC-110662, Revision 2 (see response to Audit Question A-5.4.1.3-3, Figure 18) shows that the boiling water boundary should remain far from the SG tube inlets (at least 30 percent of tube length) and **{{ }}**, minimizing the chance of any two-phase flow inducing cavitation or CIWH loads on the tube inlets or in the FW plenum. The SG tube inlets are also fitted with IFRs which add significant resistance to any oscillatory subcooled liquid flow, further stabilizing the SG system.

Although the operating conditions and IFRs will limit the conditions under which DWO instabilities can occur (those conditions are during specific transient events as outlined in ER-101144, Revision 3 included in the response to SDAA Audit Question A-3.9.2-28 (ML24346A152 (proprietary), ML24346A151 (non-proprietary)), primarily when the DHRS is engaged), NuScale has stated that the RVI and SGs can withstand up to 2840 days of operation at DWO over the life of a plant. The plant operational time in conditions where DWO might occur will be tracked by monitoring the "approach temperature" (defined in FSAR Section 5.4.1.3) – the difference between the RCS hot temperature and steam temperature Thermal-hydraulic simulations of the NPM-20 SG at a wide range of operating conditions show that **{{ }}** when the approach temperature is small or nearly 0 (as shown in the example in Figure 3-4, "SG Tube Fluid Normal and DWO Onset Temperature Profiles" of EC-174500, Revision 1 (see response to Audit Question A-5.4.1.3-3, Figure 22). A small approach temperatures imply **{{**

}}. SIET TF-1 data, however, showed that **{{**

}} and are included in NuScale's DWO monitoring approach described in FSAR 5.4.1.3. These conditions, where the boiling boundary **{{** **}}**, should not lead to any mechanical loads on the tube inlets, IFRs, or tube sheet.

Any plant operations where the approach temperature is lower than acceptable limits (shown as a function of plant power in FSAR Figure 5.4-16, "Approach Temperature") is counted as potential time at DWO regardless of whether the actual DWO onset has been reached. **{{**

}}. The ability of the approach temperature limit to provide reasonable assurance of protection against onset of DWO, along with the appropriateness of NuScale thermal-hydraulic modeling of the locations of **{{** **}}**, is evaluated in Section 5.4.1.3 of this SER. As an added fail-safe, NuScale's calculations show that the NPM-20 has the potential to trip before DWO onset during most of the plant life (this implies **{{**

}}). The trip points are well before expected DWO onset for several initiation conditions at and above 25 percent NPM power, implying the reactor will usually trip before DWO onset can occur. The staff finds that the IFRs and the conservative approach temperature limit, the high likelihood that the **{{** **}}** and limiting the amount of time operating below the approach temperature limit will ensure minimal DWO-induced damage of the SG tubes and IFRs.

Given the unlikelihood of incoherent DWO, the possibility of coherent DWO developing is extremely low. NuScale addressed this possibility in the response to SDAA Audit Question DWO-SC-22 (ML25013A207 (proprietary) and ML25013A206 (nonproprietary)). Should coherent DWO occur it would be observable in system monitoring and considered a system-level instability and very likely mitigated by existing control systems. Operating procedures which describe how to monitor and mitigate system level instabilities like coherent DWO will be developed per FSAR Section 5.4.1.3 and COL item 13.5-3. Therefore, the staff finds that NPM-20 operating with coherent DWO conditions is highly unlikely.

{{ **}}** several relevant cases in the open literature discussed above demonstrate clearly that the period of a secondary coolant DWO instability cycle is much longer (usually more than 10 seconds) than the periods of structural resonance frequencies in the SG system and tubes. Therefore, the staff finds that DWO instabilities, should they occur, will not couple strongly with any system resonances and any mechanical loading will be benign.

Although operating at severe DWO is unlikely and operating at low approach temperature conditions will be limited, NuScale considered and assessed the following loading mechanisms associated with severe DWO in their responses to various audit questions:

- Sudden (CIWH) like events impinging on the SG tubes just downstream of their inlets and on the IFRs in the FW plenum

- cavitation loads on the IFRs and IFR mounting systems and SG tube interior walls

- tube wear from slow sliding between the tubes and tube supports induced by thermal gradients

- Fatigue wear on tubes, tubesheet (at the tube inlets), and tube-to-tubesheet welds

In the response to SDAA Audit Question A-3.9.2-34 (ML24346A158 – Proprietary, ML24346A157 – Non-proprietary) NuScale compared thermal-hydraulic conditions in the NPM-20 SG at severe DWO conditions to those which induce water-hammer like loads observed in previous BWRs and PWRs and cited in NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," NUREG/CR-5220, "Diagnosis of Condensation-Induced Waterhammer," and NUREG/CR-6519, "Screening Reactor Steam/Water Piping Systems for Water Hammer." The geometry of the SG tubes (inclined at helical angles) precludes typical CIWH loads. In the unlikely event that large "slugs" of liquid form in the two-phase flow region and accelerate backwards toward the tube inlets, NuScale evaluated bounding impulsive loads based on a peer-reviewed article (Riverin, "Fluctuating forces caused by internal two-phase flow on bends and tees," Journal of Sound and Vibration, 298 (2006) 1088-1098). NuScale determined that the bounding loads are negligible and if they occurred would not lead to tube or other component damage. CIWH in the feedwater plenum was not evaluated given the conservative approach temperature operating limit (as discussed above) Therefore, the staff concurs with NuScale that there is very low probability of two-phase flow through the IFRs and into the FW plenum.

NuScale evaluated the potential for cavitation-induced surface wear in the SG tube inlet region in Section 3.2.9, "Steam Generator Tube Inlet Flow Restrictor Density Wave Oscillation Cavitation Flow Assessment" of the CVAP analysis report (TR-121353-P) and in the response to SDAA Audit Question A-3.9.2-26F (ML24346A150 – Proprietary, ML24346A149 – Non-proprietary) and found it minimal. NuScale also evaluated the possibility of cavitation in the FW plenum at and around the IFR mounting hardware in the response to SDAA Audit Question A-3.9.2-28 (ML24346A152 - Proprietary, ML24346A151 – Non-proprietary). Significant damage to the IFR mounting hardware is highly unlikely. Should any mounting hardware fail, other preloading mechanisms would prevent the IFR from dislodging.

Thermally-induced damage including tube wear caused by sliding against supports and stresses in the tubes, tubesheet, and tube-to-tubesheet welds were evaluated in the response to SDAA Audit Question A-3.9.2-26 (ML24346A148 – Proprietary, ML24346A147 – Non-proprietary). A bounding thermal transient was assumed based on the lower and upper temperature limits of the secondary coolant (to maximize thermally induced deformations) and the longest possible DWO time (which maximizes sliding distances). In a sensitivity study, the maximum number of cycles was assumed based on the 2840 days of allowable operation at low approach temperature conditions and the conservative shortest possible DWO cycle time. Even with these conservatisms wear was less than the allowable limit. In the unlikely event significant wear occurs, it would be observable within NuScale's inspection interval (see below). NuScale shows that thermally-induced stresses are very small with negligible fatigue life impact **{{ }}**.

NRC Office of Research performed confirmatory studies of the NPM SG under various conditions conducive to possible DWO (ML25007A231). The studies used a thermal-hydraulic software – TRACE (https://www.nrc.gov/about-nrc/regulatory/research/safetycodes.html#th) –

and showed that severe DWO is highly unlikely. As was shown in NuScale's NRELAP5 simulations the boiling boundaries will remain far from the tube inlets during any DWO event. This implies that CIWH and cavitation near the tube inlet is highly unlikely.

Although severe DWO is highly unlikely to occur, and all of the loading mechanisms above appear to be benign, to ensure any unexpected wear or erosion is detected, FSAR, Section 5.4.1.6.1 and the SG inspection program (Section 5.5.4, "Steam Generator (SG) Program" of the US460 GTS, Volume 1) ensures, in part, that all SG tubes will be visually inspected after the first refueling outage and on a staggered basis every six years (72 effective full power months) afterwards. This inspection will involve removing the IFRs and the IFR mounting hardware, which will therefore also be inspected. Any tubes violating the steam generator tube plugging criteria in the Technical Specifications will be plugged. Section 5.5.4 of this SER evaluates NuScale's planned inspection program along with plugging criteria.

The staff has evaluated:

- The approach temperature monitoring methodology (in this section and in Section 5.4.1.3 of the SER)

- The current primary and secondary steady state conditions in and around the SG, including the estimated heights of the boiling water boundary (with appropriateness of calculation methodology confirmed by the staff in SER, Section 5.4.1.3 under item G2.1, "The evaluation model contains the appropriate modeling capabilities") and steam transition boundary

- The estimated amount of time approach temperature conditions which preclude DWO could be violated

- The possibility of coherent DWO occurring in the SG as well as being induced in the secondary coolant system

- The bounding loads that could be induced during the unlikely event of severe DWO operation including thermal transients, cavitation, and CIWH

- The SG Technical Specification which requires full inspection of both SGs after the first refueling outage and every 72 effective full power months of plant operation afterwards; along with plugging non-compliant tubes (see Section 5.4.1.3 of this SER)

The staff finds the likelihood of strong incoherent (or coherent) DWO occurring to be small throughout the life of the plant. In the highly unlikely event the NPM-20 operates in a sustained incoherent DWO state, the bounding loads are not expected to induce thermal or structural fatigue or tube failures within a 72 effective full power month period. Any unexpectedly high and excessive damage should be discovered during inspections and mitigated by plugging tubes. Should coherent DWO occur, NuScale would consider it a system level instability that is mitigated by existing control signal responses.

3.9.2.4.3.12 Benchmarking Testing

The applicant performed limited testing to benchmark its FIV analysis methodologies and relied more heavily on screening and analysis results to identify RVI, piping, and SG components that are at risk of damage resulting from FIV and to identify the analysis areas that require subsequent validation testing. Benchmark testing was performed for the SG using:

- SIET TF-1 secondary flow testing
- SIET TF-2 modal testing
- SIET TF-2 primary and secondary flow testing
- SIET TF-3 ”build-out” modal testing
- SIET TF-3 flow testing

TR-121354 describes the results of the benchmark tests with the exception of the flow testing which occurred in Summer and Fall 2024.

Some resonant peaks exist in TF-2 SG tube vibration spectra during flow testing, along with unexpected strong forces induced by two-phase secondary flow within the tubes (TF1 testing). However, the mild variation of TF-2 vibration peaks with increasing flow is not indicative of VS or FEI behavior (where vibration can increase substantially due to minor flow changes). Also, the unexpected TF-1 forces, when applied to models of the SG tubes, do not induce significant vibration. Finally, simulations of the TF-2 vibrations using the TB tools applied to the full-scale plant FIV analyses were shown to be conservative when compared to measured data.

NuScale’s NPM SG FEI assessments assumed tube damping ratios of 1.5 percent which is higher than the 1 percent allowable in RG 1.20 without confirmatory testing. Modal testing of the “build-out” in-air configuration of TF-3 shown in Section 3, “Benchmark Testing” of TR-121354 shows damping levels lower than the assumed 1.5 percent (in fact nearly identical to the usually accepted 1 percent). However, successfully demonstrating that neither VS or FEI can occur in TF-3 with on the order of 1 percent damping resolved this issue as shown in the final TF-3 test report summary issued in January 2025.

Finally, no benchmark testing was performed to assess the possibility of AR in the steam system since margins are expected to be much higher than 100 percent. Side branches with flow disruptors installed will be instrumented during the initial startup testing of the prototype NPM. These planned tests are evaluated in Section 3.9.2.4.4 below.

3.9.2.4.4 Flow-Induced Vibration Validation Testing and Inspection of Reactor Vessel Internals and Steam Generators

The planned measurement program details for validation testing of the NPM RVI and SG system are provided in TR-121354, Revision 1. A second report (to be submitted after the SDAA) will include the post-measurement evaluations and will be submitted after the completion of the validation testing and after the initial startup testing.

Validation testing was performed on the following:

- Prototypic SG tubes without secondary coolant with near-prototypic supports (SIET TF-3 for modes, damping, VS, and FEI) as described in FSAR Section 14.2, Test #65

According to Section 4.0, “Vibration Measurement Program,” of TR-121353, the initial startup testing (FSAR Section 14.2, Test #102) will be performed on the following:

- any RVI with less than 100 percent margin against a significant FIV mechanism

- selected sections of the CNTS piping for AR, including novel design changes such as flow disrupters

General acceptance criteria are defined for the prototypic SG validation testing in TR-121354 and in Table 14.2-65, "Test # 65 Steam Generator Flow-Induced Vibration" of FSAR Chapter 14.2 and are:

- The SG tube testing shows that FEI and VS do not occur under primary side flow rates consistent with any operating condition, considering all applicable uncertainties and biases of this separate effects test.

- The SG tube testing shows that for primary side flow rates consistent with 100 percent power operation, the SG tube vibration responses are less than those predicted with the TB analysis methodology.

General acceptance criteria are defined for the NPM prototype vibration testing in TR-121354 and in Table 14.2-102, "Test # 102 NuScale Power Module Vibration" of FSAR Chapter 14.2 and are:

- Measured vibration amplitudes in the CNTS steam piping branches confirm there is no AR concern.

- Measured vibration responses in the NPM confirm there are no resonant peaks that could indicate a strongly-coupled flow induced vibration mechanism.

Development of the detailed acceptance criteria for NPM initial startup testing is deferred to the COL applicant. COL Item 3.9-4, states that a COL applicant will provide the applicable test procedures before the start of testing and will submit the test and inspection results from the CVAP for the NPM, in accordance with RG 1.20. To ensure that the acceptance criteria are appropriate and corrective actions will be taken if the criteria are violated, the applicant committed (in FSAR Section 14.1 and 14.2), to meet Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The NRC staff finds that COL Item 3.9-4 provides reasonable assurance that the COL applicant will establish appropriate test acceptance criteria and the 10 CFR Part 50, Appendix B requirements ensure that the test acceptance criteria are met.

The NRC staff's detailed evaluations of each test are provided in the sections below.

3.9.2.4.4.1 Final Design Testing of the Steam Generator Tube Inlet Flow Restrictors

There are no plans to perform final testing of the steam generator tube inlet flow restrictors since the simple design does not require additional verification by testing due to very high margins against damaging FIV.

3.9.2.4.4.2 Flow-Induced Vibration Testing of Steam Generators in Test Facility 3

The susceptibility of the SG to VS and FEI was evaluated prior to the initial plant startup in the SIET TF-3 test facility. Evaluating a reasonably prototypic SG in a separate facility allowed more rigorous measurements with extended instrumentation and over much higher flow rate conditions than those possible in the actual plant. The expanded instrumentation and higher flow rates (over two times that planned for an operating NPM) allowed NuScale to more conclusively establish the ranges of operating conditions under which VS and FEI will and will not occur.

The applicant specified in Section 5.1, "TF-3 Validation Test" of TR-121354 requirements for the test facility, instrumentation, data acquisition, and operation, along with a list of planned tests. The NRC staff also audited the TF-3 preliminary test plans (NuScale internal document TSD-T050-54312, Revision 7, in response to Audit Question A-3.9.2-24 (ML24346A146)).

TF-3 flow testing was conducted in Summer and Fall 2024 and observed by the staff at SIET in October 2024. Preliminary staff assessments of measured tube vibrations showed no evidence of loose connections. Damping loss factors estimated from tube vibrations under flow testing were in the range of 1 percent. The TF-3 SG assembly therefore appears to be reasonably representative of that of an NPM with tight tube to tube support interfaces.

To establish allowable ranges of critical flow velocities at which VS or FEI may occur in TF-3, the applicant performed an uncertainty analysis. The uncertainty analysis is based on procedures described in the ASME Standard for Verification and Validation in Computational Fluid Mechanics and Heat Transfer and implemented as discussed in TR-121354, Section 4. The procedures are reasonable and include uncertainty in modeling and analysis (used to estimate the actual in-plant margins against VS and FEI), measurement methods, and differences between the TF-3 and the NPM with one exception discussed below. The applicant computed these total uncertainties for both VS and FEI as shown in Section 5 of TR-121354. Allowable ranges of flow velocities are specified for both mechanisms in TF-3 in Tables 5-18, "Vortex Shedding Test Range" and 5-19, "Fluid Elastic Instability Flow Tests." The staff estimates that the lower allowable limits for critical FEI and VS flow velocities in TF-3 correspond to **{{ }}** percent and **{{ }}** percent margins compared to equivalent 100 percent flow conditions in the NPM. The applicant acknowledged that testing uncertainties may decrease in the future. Such decreases will effectively increase the margins against FEI and/or VS and, therefore, the NRC staff finds this to be acceptable.

TB-induced vibration of SG tubes was also measured in TF-3. The predicted vibration levels and corresponding alternating stresses are very low. Nevertheless, the applicant provided pretest predictions of maximum allowable TB-induced vibrations. Based on the benchmarking studies in TF-1 and TF-2, the NRC staff finds that the predictions are expected to be conservative and are reasonable. The NRC staff also finds that the applicant's uncertainty analysis captures the important analysis and measurement variables and is appropriate.

NuScale submitted their final TF-3 test report summary in January 2025. The report shows:

- The tube damping at flow rates up to about 200 percent full power is small and indicates tight tube-to-tube support connections, showing the test facility is appropriate for screening for possible FEI and VS (loose connections would have induced non-prototypic high damping which would have prevented VS or FEI from occurring).

- There is no evidence of VS or FEI in any of the flow tests. Measured tube vibrations show resonance peaks at frequencies consistent with those in FE simulations. The peak vibrations increase with increasing flow at a rate typical of TB for speeds up to 250 percent full power.

- At very high flow speeds the tube vibration is strong enough to induce acoustic pressures in the water that propagate to the external pressure sensors mounted on the outer walls. The external pressure sensors clearly show the tube vibrations, indicating they are suitable for monitoring high internal structural vibrations.

Based on the TF-3 test data, the staff concludes that VS and FEI is not likely to occur in the NPM.

3.9.2.4.4.3 Initial Startup Testing of NPM

Since there are no NPM components that are expected to experience excessive FIV, planned initial startup testing of the prototype NPM for FIV mechanisms of RVIs is limited to:

- performing a flow test of the CNTS MS piping to confirm the lack of significant AR, and
- identifying and localizing any unexpectedly strong FIV effects in RVIs and the SG.

A short summary of the test method is provided in FSAR Table 14.2-102:

- Perform load ramp up to 100 percent power, then operate the NPM for a sufficient duration at 100 percent power to ensure one million vibration cycles for the component with the lowest structural natural frequency.

- Monitor the vibration of the CNTS steam piping branches, including the DHRS steam lines and MS drain valve branches. Also monitor the signals of the dynamic pressure sensors. If an unacceptable vibration response develops at any time during initial startup testing, the test conditions must be adjusted to stop the vibration and the reason for the vibration anomaly are investigated before continuing with the testing.

More details are available in TR-121354 and additional details will be provided by the COL applicant consistent with COL item 3.9-4.

Three instrumentation suite options have been proposed in Section 6.4 of TR-121354 to monitor the vibrations of RVI and the SG. All options use 14 dynamic pressure sensors with three candidates listed in Table 6-9, “Candidate Dynamic Pressure Sensors” which can withstand the temperatures and pressures within the NPM. No accelerometers or strain gages are recommended. The three options have different numbers of internally and externally mounted pressure transducers:

- Option 1: 2 internally mounted and 12 in RPV shell
- Option 2: 10 internally mounted and 4 in RPV shell
- Option 3: 14 internally mounted

Each option specifies sensor locations intended to monitor significant FIV at and near the:

- Upper riser and top of SG (4 internally or externally mounted)
- Middle of SG tube bundle (4 internally or externally mounted)
- Bottom of the SG and downcomer (4 internally or externally mounted)
- Upper riser (2 internally mounted on upper riser hanger plate)

All sensors are intended to be mounted such that the sensing element is nearly flush with the inner diameter of the mounting surfaces

NuScale has estimated the acoustic pressures which would be measured by the different sensors due to SG tube motion and CRDS motion, including the potential for metal-to-metal contact. NuScale believes significant FIV should be measurable by all pressure sensors.

The adequacy of using only pressure transducers for monitoring FIV (without any internally mounted accelerometers or strain gages) was benchmarked using the SIET TF-3 test facility. TF-3 is instrumented with both pressure transducers as well as vibration sensors on the tubes. At high flow speeds TB of the tubes was audible in the external pressure sensors at tube resonance frequencies as shown in the test report summary. Therefore, the staff concurs that NuScale's proposed external pressure sensor suites should be adequate to detect any unexpectedly high RVI or SG vibrations. TR-121354 also provides specifications for testing CNTS MS line branch connections (DHRS steam piping tees, MS drain valve branches) to assess any significant AR effects. Flow disrupters will be installed at both locations to minimize the chances of AR. Because any strong AR will lead to high vibrations and internal pressures, both are monitored. Several accelerometers will be mounted to the piping and branch connections to monitor vibration. Pressure taps (which will penetrate the piping to directly measure pressures) or circumferential arrays of externally mounted strain gauges (which indirectly measure pressure through hoop strain) will be installed. The NRC staff finds that the measurement procedure is acceptable because the pressure taps measure the pressures directly, and strain gauge arrays have been used successfully in many boiling water reactor MS measurements during extended power uprates. Both MS lines will be instrumented. This combination of instrumentation is sufficient to determine whether significant ARs are present.

Provisions for NPM initial startup vibration testing procedures and data acquisition have not yet been provided and will be submitted prior to the initial startup testing based on actual instrumentation and acquisition systems to be used. The NRC staff finds that the frequency ranges specified are reasonable, consistent with measurement programs used in previous plants, and should bound any significant resonant and/or FIV peak frequency responses, including metal to metal impacts. The testing duration was determined based on the lowest structural natural frequency from the analyses, and a goal of 1 million cycles of vibration is to be achieved, which is acceptable and consistent with common practice. AR testing will include varying flow rates to ensure significant AR does not occur across the full range of operating conditions. AR testing will also comply with the ASME Standard for Operation and Maintenance (OM) of Nuclear Power Plants Part 3: "Vibration testing of piping systems."

3.9.2.4.4.4 Inspections

In accordance with TR-121354, Section 7, "Inspection Program," NPM components that were evaluated for FIV will be inspected after the initial startup testing, following the guidelines and requirements provided in ASME BPV Code, Section III, paragraph NG5111, "General Requirements," and paragraph NB-5111, "Methods," and using the methods defined in ASME BPV Code, Section V, "Nondestructive Examination." VT-1 and VT-3 will be used to perform the visual inspections, as defined by ASME BPV Code, Section XI, Subarticle IWB2500, "Examination and Pressure Test Requirements," Table IWB-2500-1 (B-N1, B-N2, BN3), "Examination Categories BN1, Interior of Reactor Vessel; B-N2, Welded Core Support Structures and Interior Attachments to Reactor Vessels; B-N3, Removable Core Support Structures." The inspected areas include major load-bearing elements of the RVIs, restraints inside the RPV, locking and bolting components whose failure could affect the RVI integrity, contact surfaces, critical locations identified by the analysis program, and the RPV interior for loose parts. Visual examinations will be performed to assess the evidence of (1) cracks, defects, or abnormal distortion on critical surfaces, (2) cracks on welds, (3) wear, distress, or abnormal corrosion on interface surfaces, and (4) looseness of fittings. The applicant also plans periodic ISIs of the RVIs per FSAR Section 5.2.4.1. The SG program of the US460 GTS, Volume 1, specifies approximately six-year (72 effective full power months) inspection intervals after the first refueling outage. The NRC staff finds that the inspection methods and areas are consistent with those in previous applications and with the guidance in RG 1.20.

3.9.2.4.5 Dynamic System Analysis of the Reactor Vessel Internals and Steam Generators under Service Level D Conditions

This section contains three subsections. The first subsection contains the NRC staff's evaluation of the calculated loads on the NPM from postulated seismic events. The second subsection contains the NRC staff's evaluation of the calculated loads on the NPM induced by postulated short-term transient events. The third subsection is the RVI components stress analysis under the Service Level D faulted conditions.

3.9.2.4.5.1 Seismic Analysis of NuScale Power Module

TR-121515 documents the NPM seismic analysis. The technical report contains analysis methodology, input motion, structural modeling of the major NPM components (i.e., the containment, reactor vessel, upper RVI, lower RVI, CRDM, and TSS) and analysis results, including displacements, ISRS, forces, and moments at component interfaces. The major NPM components were modeled by ANSYS FE meshes. The calculated component interface displacements, forces, and moments were used as inputs for the component-level stress analyses of RVI and the SG tubes. This subsection evaluates the analysis methodology, structural modeling of the NPM (including the RVI and SG), and analysis results as they pertain to ASME Service Level D assessments of RVI and SG stresses. Assessments of the seismic modeling of structures other than the NPM are in Section 3.7 of this report.

3.9.2.4.5.1.1 Analysis Methodology

The NPM seismic analysis methodology consists of the following steps:

1. Analyze a model of the entire DB which includes the RXB, the RWB, and the engineered backfill using postulated seismic ground motion spectra and several soil profiles (these analyses are reviewed in Section 3.7 of this SER). The analysis uses procedures in TR-0118-58005-A, Revision 2, "Improvements in frequency domain soil-structure-fluid interaction analysis" (ML20353A439) and assumes six NPMs are installed. Linearized simplified NPM (NPM-SE) models are used in the RXB model and coupled with both the structural motion of the building at various support locations and with the water in the pool. NuScale compared analyses of the DB using the simplified and detailed NPM (NPM-DM) models in Appendix A of TR-121515-P, Revision 2. The bounding ISRS at several locations in the RXB and RWB are nearly identical for both NPM models. Therefore, the staff concurs that the simplified NPM model is suitable for DB seismic analyses.

2. Analyze the detailed NPM (NPM-DM) model using bounding time histories of the accelerations computed from the DB model as boundary conditions. **{{ }}** of time history are used which includes at least **{{ }}** percent of the earthquake energy. The individual NPM analyses are nonlinear, allowing for contact and nonlinear material response. In-structure displacement time histories, bounding response spectra, and bounding load amplitudes are computed for subsequent ASME stress analyses of RVI and the SG tubes. The same general NPM model is used for both analyses. All analyses were performed with ANSYS. The staff finds

the analysis methodology is consistent with acceptable practice and is acceptable. NPM component seismic analysis uses **{{ }}**. RG 1.61, Table 6, "Damping Values for Mechanical and Electrical Components," recommends 3 percent SSE damping for pressure vessels and major pressure boundary components. However, the NRC staff finds that using **{{ }}** damping in NPM subsystems and system analysis is reasonable and acceptable. The integrated NPM with many connections and internal structures is unlike traditional shell type pressure vessels, and use of damping higher than the 3 percent damping value listed in RG 1.61, Table 6, is reasonable based on the additional energy dissipation provided by the connections and internal structures. Also, the Rayleigh damping method actually induces less than the target **{{ }}** damping at frequencies between the "grounding frequencies." In the NuScale analyses damping is less than **{{ }}** the most important frequency ranges for different soil types. The staff finds the NuScale assumed damping reasonable.

3.9.2.4.5.1.2 NuScale Power Module Modeling

The NPM model, used in both the DB model in linearized form, and for detailed single NPM nonlinear analyses, includes the CNV, piping inside containment, TSS, RPV, LRVI and URVI, CRDM, and CRDM support structures, the fuel assembly, and pool bay and walls. Pool water is modeled with ANSYS fluid elements and coupled to neighboring structural surfaces. All important load transmission paths are included in the model, including the main load transmission paths to the RVI through the upper hanger plate and the upper and lower core plates (UCP and LCP). Transmission paths into the SG are also included through the FW inlet and steam outlet and the radially oriented set screws which push through the upper riser walls into the backing strips of the SG supports. Coupling through the fluid annulus between the RVI and RPV walls is modeled.

The RVI is subdivided into upper (URVI) and lower (LRVI) sections. The URVI includes the upper riser, CRAGT and CRAGT support plates, CRDS and CRDS support plates, and the upper hanger plate. The LRVI includes the lower riser, core barrel, reflector blocks, fuel assemblies, LCP, and UCP. The water inside the RVI is modeled as simple added mass. The URVI rests on the LRVI along an annular interface. Pins are used to ensure the interface is secure. A bellows section in the URVI just above the interface allows for some relative movement between the two sections. The bellows design and stiffnesses are not yet finalized so NuScale performed sensitivity analyses of the dynamic behavior of the RVI assembly with ranges of expected bellows stiffnesses in Appendix B of EC-170084, "ASME Service Level D Finite Element Evaluation of the RVI" (referenced in response to RAI 10111, Question 3.9.2-1, Revision 1 (ML24353A030 (proprietary) and ML24353A029 (nonproprietary))). The results show little difference across the expected stiffness ranges and the staff concurs with NuScale's assessment that the RVI model is insensitive to the expected range of bellows design stiffnesses.

The staff finds that the modeling approaches, models, and interconnections between models are consistent with common practices and ANSYS is an accepted FE analysis code for nuclear power plant linear and nonlinear dynamic response analysis.

3.9.2.4.5.1.3 NPM Reactor Vessel Internals and Steam Generator Seismic Load Analysis

TR-121515 provides examples of ISRS at the following locations to support subsequent structural integrity analyses of RVI and SG components:

- Core plate
- LRVI
- URVI
- CNV
- RPV and CRDM supports

Bounding forces and moments over all soil types and NPM locations were extracted from the time histories to perform engineering calculations of stresses in fasteners and interfaces between components as well as simple structures. The staff finds these forces and moments to be conservative since they span all loading cases and the maximum of each force and moment is extracted separately. The ISRS for all four soil cases and six NPM locations were enveloped and broadened by +/-15 percent to account for uncertainty in the seismic inputs and the DB and NPM models. The broadened enveloped ISRS were later applied to the ASME Service Level D RVI FE stress analyses. The staff finds that this approach is consistent with standard practices and is acceptable. The peak spectral accelerations are clustered about lower (soft soil) and higher (hard rock) frequencies which should bound most construction locations. In-plane accelerations are amplified at low frequencies by a nonlinear sliding mechanism between the interface of the CNV skirt flange and the RXB basemat and its interaction with soil column resonance frequencies. In Revision 1 of TR-121515 NuScale bounded the loads from both linear (no sliding) and nonlinear (with sliding and ground resonance) analyses over all soil types to ensure conservatism. NuScale's ASME Service Level D stress estimates were based on this bounding set of loads. NuScale updated their nonlinear analyses in Revision 2 and confirmed they are bounded by the original loading. The stress calculations were therefore not updated since they are also bounding. The staff finds the final loads on the RVI interfaces reasonable. Finally, NuScale performed transient analyses of SG and SG support stresses using two load cases which mostly bound the upper envelope of all load cases. In particular, a soft soil and hard rock case are chosen which includes spectral peaks at low and higher frequencies. The staff finds the two load cases to be a reasonable approximation of the loading upper envelope.

3.9.2.4.5.2 Short-Term Transient Analysis of the NuScale Power Module

TR-121517, Revision 1, documents the NPM short-term transient analysis. High energy pressure boundary breaches cause short term transient events that result in an asymmetric cavity pressurization load between the CNV and RPV and blowdown loads within the RPV. The technical report contains the analytical methods, benchmarking for validating the analysis methods, and the resulting asymmetric cavity pressurization and blowdown loads. The thermal---hydraulic code NRELAP5 and the ANSYS model were used to calculate the short-term transient loads. NRELAP5 generates thermal---hydraulic boundary condition inputs for the ANSYS model, which is used to calculate the short-term- transient structural loads within the NPM, including forces, moments, and differential pressure loads. This section of the SER evaluates the ANSYS analyses of TR-121517 while SER Section 3.9.1 addresses the RELAP5 analyses.

NuScale evaluated the following valve opening and break events in TR-121517, all listed under Service Level C:

- Inadvertent opening of an RSV (saturated breach)
- Spurious RVV actuation (saturated breach)
- Spurious RRV actuation (subcooled breach)
- RCS injection line break (subcooled breach)

Eight total cases were analyzed. Different modeling parameters were used for saturated and subcooled breaches. NuScale confirmed the suitability of the modeling parameters using sensitivity and convergence studies. Hand calculations of mass flow rates at the breach locations confirmed that the chosen parameters yield reasonable results. The staff concurs that the modeling parameters are appropriate for the postulated breaks.

The ANSYS approach was benchmarked against several test cases commonly used in the nuclear reactor community; therefore, the staff finds that the benchmarking performance is reasonable. NuScale stated that no uncertainty factor is needed in the short-term transient loads analysis, citing the conservatism of the benchmarking. Preliminary ASME Service Level D stress analyses provided during the audit and in response to RAI 10111, Question 3.9.2-1, Revision 1 (see Section 3.9.2.4.5.3 of this report) show that stresses induced by seismic loads are much higher than those caused by the transient ones. Also, NuScale ignores the inadvertent actuation block feature of the RRVs which would mitigate loads from that break. Given these conservatisms and the low transient loads, the staff finds that omitting uncertainty factors is reasonable.

The CNV, RPV, RVI, and other structures are modeled with solid FEs. The fluid is modeled with solid acoustic elements. The fluid-structure interface is conformal, with coincident nodes on the structural and acoustic surfaces coupled in the normal direction. The CNV has fixed boundary conditions at the CNV lugs (circumferential direction) and CNV skirt (vertical direction at all points and radial direction at four mounting points). Flow acceleration initial conditions from the NRELAP5 analyses are applied to the acoustic elements and thrust forces applied to the solid elements. The analyses are short – 0.2 seconds long - and capture the initial transient appropriately by ensuring suitably small time-step sizes are used early in the analyses. The staff finds the ANSYS modeling and analysis approach appropriate and consistent with best practices.

To calculate blowdown loads for valve inadvertent opening in TR-121517, NuScale used the reactor coolant pressure at normal operation for the RRVs and RVVs, and the design pressure (1.1 times normal operation pressure) for the RSVs at which the RSVs are set to lift. ASME BPV Code Section III, Subsection NB requires the use of coincident pressure associated with the operating loading for the Service Level D analysis; therefore, the staff finds that the pressures selected for the blowdown analysis are appropriate.

The applicant provided bounding values of the calculated forces and moments at 123 component interface locations for all cases in Table 6-4, “Maximum Forces and Moments at Component Interfaces.” In Table 6-6, “Maximum forces and moments within component sections,” bounding values of maximum forces and moments for 30 internal sections of the NPM components such as the CNV, RPV, riser assemblies, and core barrel assembly for all break and valve opening conditions are provided. Differential pressures across the pressurizer baffle plate were also provided. The applicant stated that the highest forces and moments and differential pressures result from both RVVs opening case due to the high mass flow rate and high fluid accelerations generated in this valve opening event. The maximum forces and moments and differential pressures on RPV, CNV, and RVI due to the CVCS injection line break are bounded by the case of both RVVs opening. The NRC staff finds that the applicant considered an appropriate range of transient events and identified the most limiting transient loading conditions for the NPM.

3.9.2.4.5.3 Stress and Deflection Evaluation of Reactor Vessel Internals and Steam Generators at Service Level D Faulted Conditions

NuScale submitted a summary of the peak stress intensities of the RVI and SG at ASME service level D conditions in their response to RAI 10111, Question 3.9.2-1, Revision 1, along with detailed calculations in:

- EC-170084, Revision 0 (FE calculations for RVI using ISRS inputs),
- EC-157683, "ASME Code Qualification for Service Level D Condition of RVI Components – Classical Engineering Calculations," Revision 1 (classical engineering calculations using peak forces and moments for RVI fasteners, welds, pins, and other small components not included in the RVI FE model), and
- EC-157339, "ASME Code Level D Evaluation of Steam Generator Tubes," Revision 0 (Finite element calculations for SG using transient loading inputs).

NuScale included the Service Level C short term transient loads in the Service Level D evaluations. The staff examined NuScale's detailed analyses during an in-person audit October 22-24, 2024, and in the internal calculation documents provided in response to RAI 10111, Question 3.9.2-1, Revision 1.

NuScale shows low-order modes of the NPM in TR-121515-P, Revision 2. The CNV fundamental lateral modes are those of a free beam, with the RPV moving as a rigid body cantilevered from the bottom of the CNV. The fundamental vertical modes show strong motion of the URVI with the LRVI moving in phase with the RPV. The strong URVI motion is due to the low stiffness at the URVI/LRVI interface and the bellows.

The RVI FE stress intensities were calculated using a submodel of the URVI, LRVI, and core region. The SGs were not included in the stress analysis (but the effects are included in the input loads generated from the NPM models). NuScale shows modes of the RVI assembly and of the ICIGTs in EC-170084, Revision 0 (referenced with the response to RAI 10111, Question 3.9.2-1, Revision 1). The staff examined the shapes and frequencies of the modes with the highest mass fractions and finds them reasonable.

Seismic and blowdown boundary displacements were applied to all major interfaces to the RPV including at the core barrel to lower RPV interface, hanger plate to upper RPV interface, the CRDS connections, and in the lateral directions at the set screw locations. ANSYS multi-point response spectrum analysis was used to compute displacements and stress intensities. This analysis approach assumes statistically independent inputs. The staff believes statistically independent loads are unlikely for a seismic or blowdown loading which induces nearly rigid body motion of the CNV and RPV where all inputs would be correlated with each other with various relative phasing. However, it is possible that the conservatism in the bounding ISRS inputs accounts for any errors associated with ignoring relative phasing. To confirm the conservatism of the ISRS calculations, NuScale performed a full transient analysis of the RVI using a single strong transient loading case (based on the SG transient calculation experiences) to confirm the uncorrelated ISRS approach is bounding. The peak stresses from the transient calculation are within acceptable limits, but at different locations than in the ISRS calculations implying inconsistent deformations for the ISRS and transient analyses. However, because the stresses are well within acceptable limits, the staff finds the ISRS approach as implemented by

NuScale (in particular the use of bounding spectra spanning all load cases and NPM locations) to be acceptable for NPM analysis.

Total deformations are dominated by the seismic response. **{{**

}}. The staff finds the deformations are small and reasonable for seismic loading events.

ASME stress intensities (the maximum difference between principal stresses) were calculated for the entire RVI submodel. The stresses induced by the seismic and blowdown loads were combined by square of the sum of squares (SRSS). Seismic induced stresses are dominant at nearly all locations. Blowdown stresses are important at the lower set screw locations but are highly localized and only occur very early in the blowdown process, with little contribution to SRSS of stresses at other locations. This means any bias errors in the blowdown loads, should they exist, are likely inconsequential for the stress analyses.

The SRSS of the stress intensities in non-ICIGT components is **{{**
}} at the CRAGT support plate. The ICIGTs have a higher maximum stress of **{{**
}} due to their long unsupported spans. A mesh density sensitivity study confirms the stresses are reasonably converged. For conservatism NuScale compared the SRSS of the maximum stress **{{**

}}. The stress intensity generally includes both membrane and bending stresses so comparing to the low membrane limit is conservative. Final ASME stress analyses (to be delivered under ITAAC 02.01.01) will include full tables of computed and allowable stresses.

Engineering calculations of the peak stresses in bolts and other small interface components (gussets, tabs, mounting studs, threaded inserts, and welds) and simple larger structures (lower riser and core barrel cylindrical sections) were performed using the maximum forces and moments extracted from time histories of the seismic and blowdown loads. The geometries of these components are well suited to closed-form equations. Peak stresses were compared to the appropriate ASME allowables for general membrane, membrane+bending, and shear for the different component materials at 540 $^{\circ}$F. This temperature is slightly lower than the 550 $^{\circ}$F used for the RVI FE analyses but the staff finds it to be reasonably similar.

Stresses induced by the seismic and blowdown loads were summed by SRSS and combined with the other ASME Service Level D loads (DW – deadweight, EXT – external mechanical loads, SCR – SCRAM loads). The peak loads were provided in Table 3-4, “RVI Internals – Forces and Moments” of EC-157683 (see response to Audit Question A-5.4.1.3-3, Table 12) and are **{{**

}}. The staff finds that the calculated stress intensities met ASME stress allowables and small positive margins are reasonable, particularly given the conservatism in the loads used. ITAAC 02.01.01 will ensure that the ASME components will meet the ASME stress limits and are designed to withstand design-basis events. This ITAAC is evaluated in SER Section 14.3.

NuScale performed transient analyses of a FE model of the SGs including all tubes, tube supports, and backing strips. Tubes were modeled with pipe elements and the supports and backing strips with shell elements. The shell element approach was validated by comparing structural responses (static and modal analyses) of shell and solid models. The tube mesh density is higher near the bottom to capture the peak stresses more accurately. Single nodes were used to connect the tubes with the adjacent supports in all DOF, which the staff finds conservative since all transferred loading is concentrated at a single point. Tube support to tube support connections were limited to the horizonal directions. Tube material properties vary with height to account for the different temperatures of the primary and secondary coolants and the varying densities of the secondary coolant (subcooled at the bottom transitioning to steam at the top). **{{ }}** was applied similar to that used in the loading calculations. The staff finds the SG modeling approach reasonable and consistent with best practices.

The model was analyzed for static deflections and stresses using deadweight loads to confirm that all constraints and connections were appropriate. The model was also evaluated for structural modes to ensure they are appropriate. The mode shapes and frequencies are generally consistent with those from the model used in the CVAP. The lowest modal frequencies are in the range of the peak seismic loading frequencies for hard rock conditions which the staff finds is conservative. Finally, a short time history analysis was run for this baseline model along with a model with a finer mesh. The calculated baseline model stresses are nearly identical to those in the refined model confirming the adequacy of the mesh density.

Displacement time histories from the seismic **{{ }}** and blowdown **{{ }}** were applied at the inlet FW plenum, set screw locations at the SG tube supports, and at the baffle plate main steam plenum. Pressure and deadweight loads were included per ASME Code requirements. The most limiting load cases were two seismic events (out of the 24 analyzed), one for hard rock and one for soft rock, both for NPM module 1; and the blowdown case with both RVVs opening (which are assumed to be the same for all modules). The staff compared the chosen seismic load cases with the upper envelope of all load cases **{{ }}** and concurs with NuScale's choice of limiting load cases.

The seismic and blowdown stresses were combined by SRSS and added to the deadweight and pressure stresses. The final stresses are dominated by seismic loads. The peak stress intensity of **{{ }}**, is less than the ASME Service level B allowable limit of 29.4 ksi. NuScale design specification requires that the Service Level D analyses of the ASME Code Class 1 components meet Service Level B limits, which are lower than the Service Level D limits. Shear stresses are very small – about 20 times less than the allowable limit of 16 ksi.

The staff has evaluated the seismic and blowdown loading, structural analysis methodologies, and the estimated peak stresses and displacements in the RVI and SG and finds them reasonable and within allowable ASME Code limits.

3.9.2.4.6 Correlations of Reactor Vessel Internal and Steam Generator Vibration Tests with Analytical Results

Some benchmarking test data have been compared to analytic results. In particular, the forced response of the SG TF-2 was predicted using the same bounding models used by the applicant for its NPM SG design analyses. The resulting predicted response generally exceeds, sometimes significantly, TF-2 measurements, providing confidence in the conservatism of the applicant’s design analysis methods. Based on this, the NRC staff finds the TF-2 benchmarking to be acceptable, but it is not sufficient to fully validate the conservatism of NuScale’s CVAP methods. TF-3 testing, which models the NPM internal SGs, validated the adequacy of NuScale’s methods for evaluating the possibility of VS and FEI in the SGs and confirmed that no VS or FEI is expected in the NPM SG.

3.9.2.5 Combined License Information Items

SDAA Part 2, Table 1.8-1, lists COL information item numbers and descriptions related to dynamic testing and analysis of SSCs from FSAR Section 3.9.2.

Table 3.9.2.-1: NuScale COL Items for FSAR Section 3.9.2

COL Item No.	Description	FSAR Section
COL 3.9-2	An applicant that references the NuScale Power Plant US460 standard design will complete an assessment of piping systems inside the Reactor Building to determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. Piping systems within the scope of this testing include American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Class 1, 2, and 3 piping systems, other high-energy piping systems inside seismic Category I structures or those whose failure would reduce the functioning of any seismic Category I plant feature to an unacceptable level, and seismic Category I portions of moderate-energy piping systems located outside of containment. The applicant may select the portions of piping in the design for which vibration testing is performed while considering the piping system design and analysis, including the vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program.	3.9.2.1
COL 3.9-3	An applicant that references the NuScale Power Plant US460 standard design will verify that evaluations are performed during detailed design of the main steam lines, using acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology in “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” TR-121353 is acceptable for this purpose. The applicant will update Section 3.9.2.1.1.1 to describe the results of this evaluation.	3.9.2.1.1.1
COL 3.9-4	An applicant that references the NuScale Power Plant US460 standard design will provide applicable test procedures before the start of testing and will submit test and inspection results from the Comprehensive Vibration Assessment Program for the	3.9.2.4

	NuScale Power Module in accordance with Regulatory Guide 1.20.	

3.9.2.6 Conclusion

Pending verification of the references in FSAR Section 3.9.2.1 regarding the FSAR Section 14.2 initial startup piping vibration and thermal expansion testing, the NRC staff concludes that by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during the initial startup on specified high- and moderate-energy piping, and all associated systems, restraints, and supports, the design will meet the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, 14, and 15. These tests provide confirmation that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service and that adequate clearances exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations.

The NRC staff finds that the applicant's FIV analysis and testing procedures are reasonable and conservative. Important FIV mechanisms have sufficient margin to provide reasonable assurance against their occurrence during normal and faulted NPM operation. TF-3 test data confirmed the safety of the SG for FIV and VS and FEI in particular. NPM initial startup testing will capture any unexpectedly high FIV using arrays of externally mounted pressure sensors. Initial startup testing will also confirm the lack of significant ARs in the CNTS steam piping. The applicant has committed to performing all testing in compliance with Criterion XI of 10 CFR Part 50, Appendix B.

Full inspection following the initial startup testing, followed by periodic inspections throughout the life of the plant, including the SGs, provide further confidence in the safety of the RVIs and SG. The NRC staff concludes there is reasonable assurance that there will be no significant RVI degradation due to FIV during the life of an NPM. The NRC staff also concludes that the NuScale design meets the requirements of 10 CFR Part 50, Appendix A, GDC 1 and 4, for design and testing of reactor internals to quality standards commensurate with the importance of the safety functions performed with appropriate protection against dynamic effects, including FIV and AR. The NRC staff also concludes that the CVAP for the first reactor module, in accordance with the regulatory positions of RG 1.20, provides an acceptable basis for design adequacy of the reactor internals under test loading conditions comparable to those experienced during operation without significant secondary coolant DWO instabilities. Finally, the NRC staff concludes that the design will meet the relevant requirements of Appendix B to 10 CFR Part 50 and GDC 1 and 4, with regard to the internals of a prototype reactor being tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects.

The staff has evaluated the possibility of severe DWO occurring in the NPM and finds it highly unlikely that DWO-induced loads at and around the SG inlets, such as those due to CIWH and cavitation, will occur. NuScale's approach temperature limits will ensure that the **{{ }}**, precluding these loading types. In the highly unlikely event repeated severe DWO occurs, NuScale has confirmed that CIWH and cavitation events will not lead to significant SG damage. Finally, the inspection program ensures all tubes are inspected after the first refueling outage and after 72 effective full power months of plant operation afterwards. Any tubes violating wear limits will be plugged.

The NRC staff concludes that appropriate dynamic system analyses have been performed to confirm that the structural design of the reactor internals is able to withstand the dynamic loadings of the most severe short-term transient events in combination with SSE, and other loads with no loss of function. The NRC staff also concludes that the methods and procedures for dynamic systems analyses, the considerations in defining the mathematical models, the descriptions of the acceptance criteria, and the interpretation of the analytical results comply with the relevant requirements of Appendix S to 10 CFR Part 50 and GDC 2 and 4.

3.9.3 ASME BPV Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Introduction

The structural integrity and functional capability of pressure-retaining components, their supports, and core support structures are ensured by designing them in accordance with ASME BPV Code, Section III, or other acceptable industry standards. This section addresses the loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME BPV Code Class 1, 2, and 3 components and component supports.

The criteria for the SSC design include the following considerations:

- loading combinations, transients, and stress limits
- pump and valve operability assurance
- design and installation criteria of ASME BPV Code Class 1, 2, and 3 pressure-relief devices
- component supports

3.9.3.2 Summary of Application

SDAA: SDAA Part 8, Chapter 2, “Module-Specific Structures, Systems, and Components Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC,” addresses component design and provides the NPM SSC design descriptions and design commitments.

FSAR: FSAR Section 3.9.3 addresses several areas of review, including loading combinations, system operating transients, and stress limits for component design; the design and installation of pressure-relief devices; pump and valve functional capability; and the design of component supports.

ITAAC: SDAA Part 8, Tables 2.1-1 and 2.1-2 describe Design Commitments and ITAAC discussion for the NPM ASME BPV Code Class 1, 2, and 3 piping systems and mechanical components. SER Section 14.3 discusses the NuScale ITAAC.

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

Technical Reports: There are no Technical Reports for this area to review.

3.9.3.3 Regulatory Basis

SRP Section 3.9.3, Revision 3, provides the relevant Commission regulations for this area of review, summarized below, the associated acceptance criteria, and the review interfaces with other SRP sections:

- 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1, as they relate to the design, fabrication, erection, construction, testing, and inspection of structures and components to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2 and 10 CFR Part 50, Appendix S, as they relate to the design of structures and components important to safety to withstand the effects of earthquakes without loss of capability to perform their safety functions

- GDC 4, as it relates to the design of structures and components important to safety to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs

- GDC 14, as it relates to the design, fabrication, erection, and testing of the RCPB to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture

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

The guidance in SRP Section 3.9.3 lists the acceptance criteria adequate to meet the above requirements, as well as 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 1.124, Revision 3, “Service Limits and Loading Combinations for Class 1 Linear-Type Supports,” issued July 2013

- RG 1.130, Revision 3, “Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports,” issued July 2013

3.9.3.4 Technical Evaluation

GDC 1, 2, and 4 and 10 CFR 50.55a require, in part, that SSCs important to safety be constructed and tested to quality standards commensurate with the importance of the safety functions to be performed and designed with appropriate margins to withstand the effects of anticipated normal plant occurrences, natural phenomena such as earthquakes, and postulated accidents, including LOCAs. GDC 14 and 15 require that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture and be designed with sufficient margin to ensure the design conditions are not exceeded.

The staff reviewed the structural integrity and functional capability of pressure retaining components and their supports, as well as core support structures that are designed in accordance with ASME BPV Code, Section III, Division 1. The staff reviewed loading combinations and their respective stress limits, the design and installation of pressure relief devices, and the design and structural integrity of ASME BPV Code Class 1, 2 and 3 components and component supports, as well as ASME BPV Code Class CS core support structures.

3.9.3.4.1 Loading Combinations, System Operating Transients, and Stress Limits

In accordance with SRP Section 3.9.3, the staff reviewed in the FSAR loading combinations, design transients, and stress limits that are used for the design of the safety-related ASME BPV Code, Section III, Class 1, 2, and 3, components, component supports, and core support structures. The following tables in the FSAR provide load combinations and supporting information associated with the ASME BPV Code, Section III, Class 1, 2, and 3 components:

- Table 3.9-2, “Load Definitions,” defines the pressure, mechanical, and thermal loads.
- Table 3.9-3, “Load Combinations for Reactor Pressure Vessel Class 1 Components,” gives the required load combinations and stress limits for the RPV.
- Table 3.8.2-2, “Load Combinations for ASME Stress Analysis of Containment Vessel Pressure Retaining Items,” and Table 3.8.2-3, “Load Combinations for ASME Stress Analysis of Class 1 Supports,” relate to the CNV.
- Table 3.9-6, “Load Combinations for Reactor Pressure Vessels Class 1 Supports.”
- Table 3.9-7a, “Load Combinations for Reactor Vessel Internals.”
- Table 3.9-7b, “Load Combinations for Reactor Vessel Internals Injection Line.”
- Table 3.9-7c, “Load Combinations for Reactor Vessel Internals Upper Riser Bellows.”
- Table 3.9-8, “Load Combinations for Control Rod Drive Mechanism Support Structure.”
- Table 3.9-9, “Load Combinations for Control Rod Drive Mechanism Pressure Housing American Society of Mechanical Engineers Stress Analysis.”
- Table 3.9-10, “Load Combinations for Decay Heat Removal System Condenser.”
- Tables 3.9-12, “Loading Combinations for Class 2 Containment Isolation Valves and Decay Heat Removal System Actuation Valves,” through 3.9-16, “Load Combinations for Thermal Relief Valves,” relate to RVV and RRV loads, the secondary system containment isolation valves (SSCIVs), primary system containment isolation valves (PSCIVs), and thermal relief valves.

FSAR Section 3.9.3.1.1 describes the design and service level loadings used for the design of ASME BPV Code, Section III, Class 1, 2, and 3, components, component supports, and core support structures and states that the design transients and the number of events used in the fatigue analysis are in FSAR Section 3.9.1. FSAR Section 3.9.3.1.2, “Load Combinations and Stress Limits,” defines the loading combinations for the ASME BPV Code Class 1, 2, and 3 components, component supports, and core support structures. These sections also define the stress limits applicable to the various load combinations. The loading combinations and corresponding stress limits for ASME BPV Code design are defined for the design condition; Service Levels A, B, C, and D; and test conditions. FSAR Section 3.9.1 provides the design transients and number of events and occurrences for fatigue analyses. ITAAC No. 02.02.01 in SDAA Part 8, Table 2.1-2, will verify that the as-built final design analyses for ASME BPV Code

Class 1, 2, and 3 components meet the ASME BPV Code requirements through inspection of design reports. FSAR Section 3.12 describes and discusses the loads used for piping analysis for thermal stratification, cycling, and striping. The staff reviewed piping and pipe support loading combinations and corresponding stress design criteria and documented the acceptance in Section 3.12 of this report. In SER Section 3.6, the staff evaluates pipe whip and pipe impingement loads from pipe breaks.

The staff reviewed FSAR Section 3.9.3.1.1 and Table 3.9-2, for applicable loads for components, component supports, and component support structures, considering loads such as pressure, deadweight, thermal expansion, seismic, system operating transients, wind, pipe break, thermal stratification, cycling and striping, friction, scram, load test, lifting, handling, and transportation.

The staff reviewed the lower RPV stress analysis during an audit. The staff identified that the heatup to hot shutdown transient input for the stress analysis is not consistent with thermal hydraulic heatup to hot shutdown transient. NuScale responded that the staff-reviewed calculation was completed with a preliminary transient that was current at the time. NuScale stated that the final version of the stress calculation will use the finalized transients, as governed by the certified ASME Design Specification and final ASME calculations are documented in the certified ASME Design Report that confirms the analysis meets ASME Design Specification and further enforced by the ITAAC acceptance criteria defined in Section 2.1.2 Inspections, Tests, Analyses, and Acceptance Criteria of FSAR Part 8. On the basis that ITAAC will address this issue, the staff finds this acceptable.

The staff reviewed the flange connection between the lower RPV section and upper RPV section. The staff noted that the upper RPV section is made of low alloy steel and the lower RPV section is made of austenitic stainless steel with different material properties (e.g., thermal expansion coefficients, modulus of elasticities, etc.). Because of the differing materials, shearing stress would also need to be accounted for in estimating the total load to determine whether the bolting stress limit would be exceeded. The ASME Code closure Bolting stress limit in XIII-4000 does not address combined tension and shear in Service Levels A, B and C. The staff requested NuScale to provide the flange bolt design calculation for ASME Code stress qualification to understand the impact of the additional bolt shear loading due to the difference in material thermal expansion properties at the connection.

The staff issued RAI-10150-R1 (Q3.9.3-11) (ML24069A014) to request NuScale to provide a summary of the inputs, assumptions, allowable stress limits, and results demonstrating that the upper-to-lower RPV flange bolted joint meets the ASME Section III stress limits, including consideration of the shear loading results from differential thermal expansion, and to make the analysis available for staff review and update the FSAR to clarify how the combined tension and shear for RPV closure bolts are accounted for and meet stress limits. NuScale performed preliminary finite element analysis with significant preload for closure bolt to alleviate the bending and shear stresses for differential thermal expansion of the upper/lower flanges. The COL applicant’s final design report for the flanges and closure bolt confirming that the analysis meets ASME Design specification will be enforced by the ITAAC. On the basis that ITAAC will address this issue, the staff finds this acceptable.

FSAR Section 3.9.3.1.1 addresses feedwater pipe break (FWPB) and main steam pipe break (MSPB), which are high energy pipe breaks to be considered for analyses outside the CNV but not inside the CNV, where BTP 3-4 is applied.

For seismic loads, FSAR Section 3A describes analysis of seismic loads on ASME Class 1, 2, and 3 components and support. The staff reviewed Appendix 3A and documented the acceptance in Section 3.9.2 of this report.

3.9.3.4.2 Design and Installation of Pressure-Relief Devices

The RCS reactor safety valves (RSVs) located on the RPV are designed as ASME BPV Code, Section III, Class 1, pressure-relief, pilot-operated devices. There are two RSVs, which are not connected to any piping on their discharge sides and vent directly into the CNV. The RSV function is to prevent RCS pressure from exceeding 110 percent of design pressure under normal and abnormal conditions and to prevent the exceedance of service limits. The two valves, each with sufficient capacity to limit over-pressurization of the RPV, are normally closed, are low leakage, and are used infrequently. The RCS and PZR steam space are sized to avoid an RSV lift for anticipated transients.

The ECCS valves are also located on the RPV and are part of the RCPB. These ECCS valves are Seismic Category I components and designed as ASME BPV Code, Section III, Class 1, components. SER Section 6.3.2 discusses the ECCS valves in detail. These valves are normally closed during startup, shutdown, and power operation; however, they are normally open during refueling. They are remotely actuated by an MPS signal, loss of power, or operator action, to allow flow between the RPV and CNV.

The applicant stated that RSVs and ECCS valves are designed to withstand vertical and lateral loading from seismic ground accelerations considering the appropriate damping values for pressure boundary valve bodies. The staff reviewed the RSV and ECCS valve design, including the following: (1) how these ASME BPV Code Class 1 components are qualified by analysis or test, or both, using static analysis or dynamic analysis, (2) the loads considered for calculating fatigue CUF and effects of the environment-assisted fatigue for these valves, and (3) damping values were used in the analysis.

FSAR Section 3.10.2 states that the guidance and requirements of RG 1.100, Revision 4, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” issued September 2009, and IEEE 344-2013 are the source of the methods and procedures used for seismic and dynamic qualification of mechanical and electrical equipment and ASME Standard QME-1-2017, “Qualification of Active Mechanical Equipment Use in Nuclear Power Plants,” is used with the exceptions noted in RG 1.100, Revision 3, issued September 2009 (ML091320468), for the qualification of active mechanical equipment. In SER Section 3.9.6, the staff evaluates the acceptability of the qualification of ASME BPV Code Class 1 valves.

FSAR Section 3.9.1.1, discusses the individual transients and the number of cycles included in the design basis. As described in FSAR Section 3.9.3.1.1, a fatigue analysis is performed in accordance with ASME BPV Code, Section III, Subsections NB-3200 or NG-3200, considering the effects of the LWR environment, in accordance with RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effect of the Light Water Reactor Environment for New Reactors,” and NUREG/CR-6909, “Effect of LWR Water Environments on the Fatigue Life of Reactor Materials.” This analysis considers the effects of EAF. The staff finds that this complies with SRP Section 3.9.3, Acceptance Criterion II.1, and is therefore acceptable.

FSAR Section 3.7.1,2,1 states that the percentage of critical damping for mechanical components, including pressure boundary valve bodies, use the damping value of RG 1.61, Revision 1. The staff finds this consistent with NRC recommended position and therefore, acceptable.

The SG thermal relief valves are classified as Seismic Category I, ASME Class 2 relief valves and are designed to maintain structural integrity and to function under ASME Level A, B, C, and D loading combinations. The staff finds this acceptable to meet 10CFR 50.55a requirement.

3.9.3.4.3 Component Supports

The staff reviewed the design and analysis of component supports in accordance with SRP Section 3.9.3. The staff reviewed all information in FSAR Section 3.9.3.4, “Component Supports,” to ensure that ASME BPV Code Class 1, 2, and 3 component supports are designed to meet the pertinent requirements of the regulations discussed in SER Section 3.9.3. The review included an assessment of the design criteria, analysis methods, and loading combinations used in establishing a basis for structural integrity of the supports. FSAR Section 3.9.3.1.1 and Table 3.9-2 define applicable loads. Dynamic loads are combined using square-root-sum-of-square (SRSS), considering the statistical independency of time phasing of events in accordance with RG 1.92 and NUREG-0484, Revision 1, “Methodology for Combining Dynamic Responses,” issued May 1980. In SER Section 3.9.4, the staff evaluates the control rod design adequacy and the rod ejection event.

FSAR Section 3.9.3.1.2 states that the ASME BPV Code Class 1, 2, and 3 component and piping supports are designed in accordance with ASME BPV Code, Section III, Subsection NF. The core support structures are designed to ASME BPV Code, Section III, Subsection NG. The applicant also stated that the SG tube supports are internal supports and, therefore, are also designed using ASME BPV Code, Section III, Subsection NG. FSAR Table 3.9-7a, Table 3.9-7b, and Table 3.9-7c specify the required load combinations and allowable stress limits for the design of the RVI and supports. RG 1.124 and RG 1.130 supplement the allowable stress criteria in ASME BPV Code Class 1 lineartype- and plate-and-shelltype supports, respectively. The ASME BPV Code Class 1 supports consider highcycle fatigue design in accordance with ASME BPV Code, Section III, Subarticle NF-3320, and the effects of the plant operating environment in accordance with RG 1.207 and NUREG/CR-6909. The applicant stated- that operating-basis earthquake (OBE) loading is applicable only to the fatigue analysis. The TSS mounted to the CNV provides support for piping systems and valves attached to penetrations in the CNV top head and for electrical cables and conduit for various equipment in the NPM. The TSS is a Seismic Category I component and classified as an ASME BPV Code, Section III, Class 2, support to be designed in accordance with ASME BPV Code, Section III, Subarticle NF-3250, using FSAR Table 3.8.2-5, load combination and allowable stress limits. SER Section 3.12 evaluates ASME BPV Code Class 1, 2, and 3 piping supports.

FSAR Section 3.9.3.1.2 states that piping supports are designed in accordance with ASME BPV Code, Section III, Subsection NF. The core support structures are designed to ASME BPV Code, Section III, Subsection NG. The applicant also stated that the SG tube supports are internal structures and therefore are also constructed to not adversely affect the integrity of the core support structures. On the basis that SG tube supports are internal structures, the staff finds that design to NG is acceptable.

The applicant also stated that design and construction requirements for SG supports and tube supports are discussed in FSAR Section 5.4. FSAR Section 5.4.1.6 states that the SG program monitors the performance and condition of the SGs to ensure they are capable of performing their intended functions and the program provides monitoring and management of tube degradation and degradation precursors that permit preventative and corrective actions to be taken in a timely manner, if needed. The acceptance of the SG program to address tube degradation mechanism is documented in SER Section 5.4.

FSAR Section 3.9.3.4 states that FSAR Section 3.9.3.1 provides the load combinations, system operating transients, and stress limits for component supports. The applicant also stated that, as described in FSAR Section 3.9.3.3, “Pump and Valve Operability Assurance,” the functionality assurance, environmental, and seismic qualification programs that are applied to components are also applied to the associated supports. FSAR Section 3.12.6.6 states that snubbers are not used in the NuScale Power Plant for ASME BPV Code Class 1, 2, or 3 piping. The staff evaluated FSAR Section 3.9.3.3 and documented its acceptance in Section 3.9.6 of this report.

3.9.3.5 Combined License Information Items

SER Table 3.9.3-1 lists COL information item numbers and descriptions related to FSAR Section 3.9.3, from FSAR Table 1.8-1.

Table 3.9.3-1: NuScale COL Information Item for Section 3.9.3

Item No.	Description	FSAR Section
COL Item 3.9-1	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific seismic analysis in accordance with Section 3.7.2. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the certified seismic design response spectra, the standard design of NuScale Power Module components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.	3.9

3.9.3.6 Conclusion

Based on the above discussion, the staff concludes that the NuScale SDAA provides reasonable assurance that the pressure-retaining components, component supports, and core support structures are designed in accordance with ASME BPV Code, Section III, or other industry standards. The staff finds that the design of the ASME BPV Code Class 1, 2, and 3 components, component supports, and core support structures meets the relevant requirements of GDC 1, GDC 2, GDC 4, GDC 14, and GDC 15 and 10 CFR Part 50, Appendix S.

3.9.4 Control Rod Drive Systems

3.9.4.1 Introduction

The CRDS consists of the control rod drive mechanisms (CRDMs) and the related mechanical components that provide the means for mechanical movement. GDC 26, “Reactivity Control System Redundancy and Capability,” and GDC 27, “Combined Reactivity Control Systems Capability,” require that the CRDS provide one of the independent reactivity control systems. The control rods and the CRDMs shall be capable of reliably controlling reactivity changes under conditions of normal operation, including AOOs, and under postulated accident conditions. A positive means for inserting the control rods is maintained to ensure appropriate margin for malfunction, such as stuck rods. This SER section reviews the applicant's information on design criteria, testing programs, summary of method of operation of the CRDS, applicable design codes and standards, design loads and combinations, and operability assurance program. This information pertains to the CRDS, which is considered to extend to the coupling interface with the control rod assembly in the RPV. The review in this section is limited to the CRDM portion of the CRDS.

3.9.4.2 Summary of Application

FSAR: FSAR Section 3.9.4, "Control Rod Drive System," discusses the CRDS, including the CRDMs. A CRDM is an electromagnetic device that moves the CRA in and out of the nuclear reactor core to control reactivity under conditions of normal operation and under postulated accident conditions. The CRDM assembly is composed of a control rod drive (CRD) shaft, drive coil assembly, pressure housing, latch mechanism, and sensor coil assembly. Portions of the CRDS are part of the RCPB (specifically, the pressure housings of the CRDMs), and are safety related. The CRDM internals that ensure positive control rod assembly insertion consist of the latch mechanism and CRD shaft are classified as safety-related and non-risk-significant. Controlled movement of the control rod assemblies is performed by energizing the drive coils in a particular sequence, which generates magnetic fields that actuate latch arms and engage the drive shaft. If the reactor trip breakers open, power to the CRDM control cabinet is interrupted, which causes the control rods to be inserted by gravity. Rod position indication is provided by coils located in the sensor coil assembly, which slides over the pressure housing and sits on the rod disconnect mechanism coil housing.

ITAAC: SDAA Part 8, "License Conditions: Inspections, Tests, Analyses, and Acceptance Criteria," Section 2.1, discusses the NPM, which contains the CRDS. SDAA Part 8, Section 2.1.1, contains the design description for the CRDS, including the CRDMs. SER Section 14.3 discusses the NuScale ITAAC.

Technical Specifications: GTS Section 3.1.4, "Rod Group Alignment Limits," contains surveillance requirements pertinent to the review scope of SRP Section 3.9.4, Revision 4, "Control Rod Drive Systems," namely, a partial movement check and CRA drop test.

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

3.9.4.3 Regulatory Basis

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

- 10 CFR 50.55a and GDC 1, as they relate to the CRDS, require that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 2, as it relates to the important-to-safety functions performed by the CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

- GDC 14, as it relates to the CRDS, requires that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

- GDC 26, as it relates to the CRDS, requires that the CRDS be one of the independent reactivity control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation, including AOOs.

- GDC 27, as it relates to the CRDS, requires that the CRDS be designed with appropriate margin, and, in conjunction with the ECCS, be capable of controlling reactivity and cooling the core under postulated accident conditions.

- GDC 29, "Protection against Anticipated Operational Occurrences," as it relates to the CRDS, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of AOOs.

SRP Section 3.9.4 lists the acceptance criteria adequate to meet the above requirements, as well as 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:

- RG 1.26, which provides guidance to licensees for assigning components to QGs and specifying quality standards applicable to each QG

3.9.4.4 Technical Evaluation

3.9.4.4.1 Descriptive Information

The staff reviewed FSAR Section 3.9.4, in accordance with SRP Section 3.9.4. FSAR Section 3.9.4, provides information on the CRDS design, describing the CRDM components and their operation; CRDM design specifications; design loads, stress limits, and allowable deformations; and operability assurance program.

FSAR Section 4.6, "Functional Design of Control Rod Drive System," provides the majority of the figures depicting the CRDS. FSAR Figure 4.6-1, "Representative Overview of Control Rod Drive Mechanism Locations in Relation to the Reactor Pressure Vessel and Containment Vessel," shows CRDM support structures, and FSAR Section 3.9.3.1.2, briefly mentions the CRDM seismic supports located on both the RPV and CNV head as ASME BPV Code Class 1, Seismic Category I component supports. FSAR Section 3.9.4.1, also discusses the CRDM support structures. FSAR Section 4.5.1 discusses the structural materials of construction for non-pressure boundary portions of the CRDM. FSAR Section 5.2.3 discusses materials for pressure boundary portions of the CRDM. FSAR Section 4.6 provides additional information on the CRDMs. Section 4.3.1.4 provides the CRA withdrawal speed.

The staff reviewed the description of the method of operations provided in the FSAR and found it to be acceptable, as it provides an adequate level of detail to summarize the method of operation and make a determination that the operation sequence does not place the system in a non-fail-safe configuration.

SRP Section 3.9.4 states that "of particular interest are any new and unique features that have not been used in the past." In FSAR Section 3.9.4.4, the applicant states that there are two unique features: a remote disconnect mechanism and a longer CRD shaft. The remote disconnect coil is used to remotely connect and disconnect the drive shaft from the CRA, as described in FSAR Section 3.9.4.1.1. During an audit, the applicant asserted that the remote disconnect mechanism has no impact on the safety-related function of inserting control rods. This feature, as well as the rod holdout mechanism, are not used during normal plant operations and are not safety-related components. The staff reviewed the applicant's docketed information and determined that these new and unique features will not adversely impact the safety-related function of inserting control rods.

3.9.4.4.2 Codes and Standards

FSAR Section 3.9.4.2, "Applicable Control Rod Drive System Design Specifications," describes the classification of the CRDM components, stating that the components forming the pressure boundary are in accordance with the requirements of ASME BPV Code, Section III, Subsection NB. The staff finds this classification consistent with GDC 1, 10 CFR 50.55a, and RG 1.26 and therefore acceptable, as components of the RCPB are classified as ASME BPV Code, Section III, Subsection NB (Quality Group A) components.

Section 3.13 of the FSER provides an evaluation of the NuScale US460 design's novel approach to bolted connections, which utilize threaded inserts. The CRDMs use these threaded inserts, so the evaluation provided in Section 3.13 related to the RAI response provided in ML57646A067 is also applicable to the design of the CRDMs.

There is additional discussion of the design, fabrication, inspection, and testing of non-pressure-retaining components; specifically, they do not typically come under the jurisdiction of the ASME BPV Code, with the exception of the CRDM seismic supports that fall under ASME BPV Code, Section III, Division I, Subsection NF. Material specification mechanical property requirements are the basis for materials without established stress limits.

The CRD shaft was further specified as being analyzed to the guidance of ASME, Section III, Nonmandatory Appendix F for linear type supports. The CRD shaft is also evaluated to not adversely affect the integrity of the core support structures in accordance with ASME BPV Code, Section III, NG-1122(c). The specific requirements for the CRD shafts are provided in FSAR Section 3.9.4.1.1. The staff finds these requirements acceptable for the CRD shafts and other non-pressure-retaining components because they satisfy the requirements of GDC 1.

3.9.4.4.3 Load Combinations and Stress Limits

FSAR Section 3.9.4, describes the function of the CRDM and specifies the necessary requirements pertaining to its materials, design, inspection, and testing before and during service. FSAR Table 3.9-8, presents the loading combinations and corresponding stress limits for the ASME BPV Code design defined for the design condition; Service Levels A, B, C, and D (i.e., normal, upset, emergency, and faulted conditions); and test Conditions of the CRDM pressure housing. This information supports the review of applicable design loads and their appropriate combinations, the corresponding design stress limits, and the corresponding allowable deformations. SER Section 3.9.3 documents the staff's evaluation of this topic.

3.9.4.4.4 Operability Assurance

SRP Section 3.9.4, Acceptance Criterion 4, states that "[t]he operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meet system design requirements." FSAR Section 3.9.4.4, "Control Rod Drive System Operability Assurance Program," briefly discusses a series of tests for the CRDS, including life cycle testing, production testing, pre-operational testing, and TS testing. FSAR Section 4.6.3 also discusses CRDS testing. FSAR Section 4.2.4.2.3, discusses control rod testing. FSAR Section 1.5.1.6 contains further discussion of testing related to control rod assembly drop and control rod drive shaft alignment. The staff asked a series of questions related to this testing, as the SDAA appeared to be crediting testing conducted during the DCA review. The staff sought to verify that this testing remained applicable in light of design changes between the DCA and SDAA. NuScale provided a series of responses (ML23304A424, ML23304A426, ML23304A428, and ML24215A068) which demonstrated that the seismic displacements used in the SDAA design are bounded by those used for the DCA testing program. The staff therefore finds that the control rod drive system of the US460 design will remain capable of performing its safety-related function to insert control rods under the worst-case deflection postulated by the testing program.

COL Item 3.9-5 requires a COL applicant to create an operability assurance program for the CRDS. FSAR Section 3.9.4.4 provides a description of the operability assurance program requirements and directs a COL applicant to implement a program. The staff finds the description of the operability assurance program requirements and direction that a COL applicant implement a program in accordance with the requirements in Section 3.9.4.4 of the FSAR and provide a summary of the testing program and results acceptable because it outlines the needed requirements for the program (as provided in SRP Section 3.9.4, Acceptance Criterion 4) and directs a COL applicant to implement such a program and submit its program summary and results, which will be evaluated by the NRC staff as part of the COL application review.

3.9.4.4.5 Boric Acid Accumulation

During the Advisory Committee on Reactor Safeguards subcommittee meeting for the NuScale DCA on April 17, 2019 (ML19114A107), several members inquired about unique environmental conditions for the control rod drive mechanisms (CRDMs), which are very similar in configuration to those in existing pressurized-water reactors (PWRs), but operate in different environmental conditions. While there is operating experience with existing PWR CRDMs, the CRDMs typically operate in a water solid environment, so NuScale's unique design, where the mechanisms operate in a borated steam environment and are cooled by cooling coils, introduces additional uncertainties. Specifically, a member of the Advisory Committee inquired about the potential for chemical buildup to form from substances evaporating off the top of the PZR water level and whether this buildup would prevent the rod from inserting into the core. Significant accumulation of particulates such as boric acid crystals around the movable elements of the CRDM latch mechanism could inhibit the ability of the latches to release the CRD shafts and scram the reactor. This accumulation could affect multiple CRDMs and could lead to a common-cause failure of some or all CRDMs. The NRC staff had asked for assurance from the applicant that accumulation of boric acid crystals would not adversely impact the ability of the CRDMs to perform their safety-related function of inserting the control rods. The applicant had provided a response that it had considered the phenomenon in the design process but did not consider it to adversely impact the ability of the CRDMs to drop the control rods. For the US460 design, the general operating environment (e.g., a borated steam environment and external cooling system for the CRDMs) remains similar to the design reviewed under the Design Certification, but the new design may have introduced additional susceptibilities to this previously identified phenomenon. Therefore, the staff requested additional information to confirm that this phenomenon would not adversely impact the ability of the CRDMs to drop the control rods. The applicant provided a response on the docket (ML23205A218) which asserted that the original response and the subsequent evaluation remain applicable in principle. The staff finds that the applicant has considered this phenomenon in the design process for the

CRDMs, as evidenced by its response, and the applicant does not consider this phenomenon to adversely impact the ability of the CRDMs to drop the control rods.

3.9.4.4.6 Steam Generator Tube Support Assembly Impact on CRDS

During an in-person audit conducted on September 28, 2023, the staff determined that a new configuration of the steam generator tube supports could potentially impact the safety-related function of the CRDS and other nearby safety-related SSCs. Specifically, a series of threaded components are located inside the upper riser assemblies directly above the reactor core. The staff requested additional information on these novel features, with an emphasis on any potential impact on the safety-related function of inserting control rods. The staff also requested information on how the applicant will ensure that the integrity of these welds and threaded elements of the steam generator tube support assembly will remain intact. In its RAI response (ML24205A156; ML24205A157-proprietary), the applicant provided an explanation of the configuration of these assemblies and discussed their review of operating experience. The RAI response includes that integrity of the assemblies, and consequently retention of parts within the assemblies, relies substantially on several tack welds within the installed assembly. The RAI response also discussed the addition of augmented inservice inspections to monitor that the set screw assemblies are not substantively degrading over time. The applicant proposed a visual (VT-1) examination for a 10 percent sample of the set screw assemblies once per inservice inspection interval. The scope of this visual examination is limited to the welded nut and welded set screw and is performed in accordance with ASME Section XI IWA-2211. The staff agrees that a VT-1 inspection in the subject areas would be capable of detecting degradation that has the potential of generating loose parts within the operation path of the control rod drive systems, such as cracking, loose hex nuts, or other defects. Further, the staff conducted independent statistical calculations to evaluate the adequacy of the proposed sampling strategy to provide timely identification of potential patterns of degradation before failure trends emerge. The staff determined that the applicant’s proposed 10% sample strategy is sufficiently sensitive to identify potential patterns of degradation within the site (e.g. sampling across the proposed six modules will constitute an adequate program to provide the necessary sampling). Based on the above, the staff finds that the proposed inspections provide reasonable assurance that the design assumptions made by the applicant will be appropriately validated with direct evidence, and if a pattern of degradation were to occur that it would be identified in a timely manner.

3.9.4.5 Combined License Information Items

Table 3.9.4-1 lists COL information item numbers and descriptions related to the CRDS from FSAR Table 1.8-1.

Table 3.9.4-1: NuScale COL Information Item for Section 3.9.4

Item No.	Description	FSAR Section
COL Item 3.9-5	An applicant that references the NuScale Power Plant US460 standard design will implement a control rod drive system Operability Assurance Program that meets the requirements described in Section 3.9.4.4 and provide a summary of the testing program and results.	3.9.4.4

3.9.4.6 Conclusion

The staff finds that the applicant has met the requirements of 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR 50.55a, with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed because the design and construction of the pressure boundary components of the CRDS conform to the requirements of ASME BPV Code, Section III, Subsection NB. Furthermore, non-pressure boundary components of the CRDS, such as the CRD shaft, are designed to standards commensurate with the importance of their safety functions, as discussed in the evaluation above.

In addition, the staff finds that the applicant has met the requirements of 10 CFR Part 50, Appendix A, GDC 2, 14, and 26, with respect to designing the CRDS to withstand effects of earthquakes and conditions of normal operation, including AOOs, with adequate margins to assure the system’s reactivity control function and with extremely low probability of leakage or gross rupture of the RCPB. SER Section 3.9.3 documents the staff’s evaluation of the specified design transients, design and service loadings, combination of loads, and resulting stresses and deformations under such loading combinations.

The staff finds that the applicant has designed the CRDS to reliably control reactivity changes under postulated accident conditions, as discussed in GDC 27, since it has been designed to quality standards commensurate with its safety functions. The staff also finds that the applicant has met the requirements of GDC 29, “Protection against Anticipated Operational Occurrences,” with respect to designing the CRDS to assure an extremely high probability of accomplishing its safety functions in the event of AOOs, as it has been designed to accommodate the effects of earthquakes and conditions of normal operation, as mentioned earlier in this section. As discussed above, the staff further concludes that the operability assurance program that will be implemented by a COL applicant is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Introduction

This section verifies that the FSAR describes the arrangement of the RVIs and their specific functions, the flowpath through the RPV, and the applicant’s design criteria. The RVIs serve several functions. They provide support and alignment for the reactor core, a flowpath that directs and distributes the flow of reactor coolant through the nuclear fuel under all design conditions, and support for the control rod assemblies (CRAs).

The objectives of the staff’s review are to confirm the following:

- The RVIs have been designed and tested to appropriate quality standards.
- The portions of the RVI that provide structural support for the core meet the applicable requirements of ASME BPV Code, Section III.
- The appropriate design transients and loading combinations have been specified.

- The RVI mechanical stresses, and deformations will not result in a loss of structural integrity or impairment of function.

The designation “reactor vessel internals” in the context of this review section includes the core support structures, internal structures, and all structural and mechanical elements inside the RPV with the following exceptions:

- reactor fuel elements and the reactivity control elements
- control rod assemblies
- in-core instrumentation (ICI)
- upper SG supports and pressurizer spray nozzles

3.9.5.2 Summary of Application

FSAR: FSAR Section 3.9.5 describes the arrangement of the RVI assembly and the flowpath of reactor coolant through the RPV. The RVI assembly comprises several subassemblies that are located inside the RPV. The RVIs support and align the reactor core system, which includes the CRAs; support and align the CRD rods; and include the guide tubes that support and house the ICI. In addition, the RVIs channel the reactor coolant from the reactor core to the SG and back to the reactor core.

FSAR Section 3.9.5 states the following as the RVI primary functions:

- Provide structures to support, properly orient, position, and seat the fuel assemblies to maintain the fuel in an analyzed geometry to ensure that core cooling capability and physics parameters are met under all modes of operational and accident conditions.

- Provide support and properly align the control rod drive system (CRDS) without precluding the full insertion of control rods under all modes of operational and accident conditions.

- Provide the flow envelope to promote natural circulation of the RCS fluid with consideration given to minimizing pressure losses and bypass leakage associated with the RVIs and to the flow of coolant to the core during refueling operations.

FSAR Section 3.9.5 states that the design and construction of the core support structures comply with the ASME BPV Code, Section III, Division 1, Subsection NG. The internal structures are constructed to not adversely affect the integrity of the core support structures.

ITAAC: The ITAAC associated with FSAR Section 3.9.5 are given in SDAA Part 8, Section 2.1.2, “Inspection, Tests, Analyses, and Acceptance Criteria.” These ITAAC are evaluated in SER Section 14.3.

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

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

3.9.5.3 Regulatory Basis

As noted in SRP 3.9.5, “Reactor Pressure Vessel Internals,” the following NRC regulations contain the relevant requirements for this review:

- GDC 1 and 10 CFR 50.55a require, in part, that reactor internals be designed to quality standards commensurate with the importance of the safety functions performed.

- GDC 2 requires, in part, that SSCs important to safety, such as the reactor internals, be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform safety functions.

- GDC 4 requires, in part, that SSCs important to safety, such as the reactor internals, be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including LOCAs. Dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for piping.

- GDC 10, "Reactor Design," requires, in part, that reactor core and associated coolant, control, and protection systems (including reactor internals) be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any conditions of normal operation, including the effects of anticipated operational occurrences (AOOs).

The staff notes that, for the purposes of this review area, if the RVIs meet the requirements of GDC 1, GDC 2, and GDC 4, there is reasonable assurance that they will meet the pertinent requirements of GDC 10. Furthermore, although SRP Section 3.9.5 specifically references 10 CFR 50.55a as a relevant regulatory requirement, the primary requirement for quality standards in this review area is GDC 1.

3.9.5.4 Technical Evaluation

3.9.5.4.1 Loads and Load Combinations

FSAR Section 3.9.5.3 states that the RVI core support structures and internal structures are designed for the service loadings and load combinations shown in FSAR Table 3.9-7a, Table 3.97b, and Table 3.97c. Section 3.9.3 of this report addresses the method of combining loads for ASME Service Levels A, B, C, and D and test conditions.

The staff reviewed FSAR Table 3.9-7a, Table 3.97b, and Table 3.97c, and found that it adequately lists load combinations under all four service level conditions. The plant event rod ejection accident is categorized as a Service Level D condition and uses a Level C allowable limit. Acceptance Criterion 2 under SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," requires that the postulated reactivity accident would result in neither damage to the RCPB greater than limited local yielding nor sufficient damage to significantly impair core cooling capacity.

Staff noted that the rod ejection accident is classified as a Level D condition because of the low number of anticipated occurrences and low consequences of the event over the 60-year design life, and the classification of this event as a Level D condition is consistent with other recent design control documents. The stress limits are set to Level C limits according to the guidance in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Position C.2, which states the following:

Maximum reactor pressure during any portion of the transient will be less than the value that will cause stresses to exceed the Emergency (Level C) condition stress limit as defined in Section III of the ASME Boiler and Pressure Vessel Code.

The ASME BPV Code permits a more restrictive stress limit to be specified than the service limit at which the event is classified. The more restrictive service limit further reduces potential damage from using the higher service limit. The staff finds considering the rod ejection accident event as a Level D event while using Level C service limit acceptance criteria acceptable because using a Level C service limit for a Level D event is more conservative and thus provides a higher safety margin.

FSAR Section 3.9.3.1.2, in the subsection “Core Support Structures,” states that the SG tube supports are Seismic Category I components, are designated as internal structures, and are constructed to not adversely affect the integrity of the core support structures

The staff finds this acceptable because the SG tube supports do not provide a core support function. Therefore, the classification as internal structures is acceptable, and using ASME BPV Code, Section III, Subsection NG, as a guide meets the acceptance criteria in SRP Section 3.9.5 and is therefore acceptable.

3.9.5.4.1.1 Design—Core Support Structure

FSAR Section 3.9.5 states that the RVIs comprise several subassemblies located inside the RPV. The RVIs support and align the reactor core system (which includes the CRAs), support and align the CRD rods, and include the guide tubes that support and house the ICI.

FSAR Section 3.9.5 states that the RVI assembly comprises these subassemblies:

- core support assembly (CSA)
- lower riser assembly
- upper riser assembly (URA)
- CSA mounting brackets
- SG tube supports
- lower SG supports
- ICI and riser level sensor guide tubes

FSAR Section 3.9.5, Figure 3.9-2, “Upper Riser Assembly,” Figure 3.9-3, “Lower Riser Assembly,” and Figure 3.9-4, “Core Support Assembly,” provide basic sketches of the URA, lower riser assembly, and the CSA, respectively. These figures reference multiple RVI components.

In SRP Section 3.9.5, the area of review specifies the physical or design arrangements of all reactor internals structures, components, assemblies, and systems, including the positioning and securing of such items within the RPV; the provision for axial and lateral retention and support of the internals assemblies and components; and the accommodation of dimensional changes resulting from thermal and other effects. The SRP Section 3.9.5 review procedure states that the configuration and general arrangement of all mechanical and structural internal elements covered by the SRP section are to be reviewed and compared to those of previously licensed similar plants.

The staff reviewed the configuration and general arrangement of the reactor vessel internals. The lower riser assembly includes the lower riser, the upper core plate, CRA guide tube assemblies, CRA guide tube support plate, and ICI guide tubes. The lower riser assembly is located immediately above the CSA and is aligned with and supported on the CSA by the four upper support blocks. The lower riser channels the reactor coolant flow exiting the reactor core upward toward the upper riser, and separates this flow from the flow outside the lower riser that is returning from the SGs, with the exception of flow paths in the riser shell that permit a small amount of reactor coolant to bypass the top of the riser and flow into the downcomer region. The CSA is mounted to the bottom head of the RPV, which provides the primary support to the reactor core.

TR-121353-P, Revision 2, provides further detail of the RVI assembly and description of the SG, which wraps around the URS, and PZR, which is located above the URS. Specifically, details are given for the steam plenum and the PZR baffle plate, in which a portion of the PZR baffle plate forms the steam plenum tubesheet, which allows the steam to travel through on the secondary side. This report also gives details of the SG tube Inlet Flow Restrictors (IFRs) and mounting plate, the helical SG tube bundle, and the SG support bars that provide support to maintain the tube bundle structural integrity.

There are no separate boundaries for components in the PZR or SG region. In the context of the ASME BPV Code, the SG (including the tube support structures) and PZR are fully integral to the RPV; these three items are designed as a single ASME component. All other ASME components that are contained within the volume of the RPV RCPB are part of the RVI components, except for the CRD shafts, fuel, control rod assemblies, and various instruments.

The integral steam plenum, including the sections that make up the SG tubesheets, and PZR baffle plate are designed in accordance with ASME BPV Code, Section III, Subsection NB. The FW plenums, including the tubesheets, are designed in accordance with ASME BPV Code, Section III, Subsection NB. Both the integral steam plenum and the FW plenums form part of the RCPB.

The core support blocks welded to the RPV are structural attachments to the RPV providing core support. The core support mounting brackets are part of the RVI component and are designed in accordance with ASME BPV Code, Section III, Subsection NG. As a structural attachment, the core support blocks do not directly form part of the RCPB.

The SG tubes are part of the RCPB and are designed in accordance with ASME BPV Code, Section III, Subsection NB. The SG flow restrictors and associated hardware (including mounting plates, bolts, nuts, spacers, and studs) are nonpressure boundary items and are not inside or integral to the RCPB and therefore are not RVI components.

The SG tube supports, including the upper tube support bars, lower tube support cantilevers, and tube support bar assemblies, are a structural attachment to the RPV. The SG tube supports do not form part of the RCPB. The code classification boundary (NB to NF) between the upper SG tube supports and the RCPB portion of the RPV is at the weld between the upper tube support bars and the PZR baffle plate. The code classification boundary (NB to NG) between the lower SG tube support and the RCPB portion of the RPV is at the weld between the lower tube support cantilevers and the RPV shell. These welds are designed in accordance with ASME BPV Code, Section III, Subsection NB. The SG tube supports are designed as internal structures, in accordance with ASME BPV Code, Section III, Subsection NG. SER Section 5.4.1 provides more detail for the SG tube supports.

FSAR Section 5.4.1.5, “Steam Generator Materials,” states that the design code for the SG flow restrictors is ASME BPV Code, Section III, Subsection NC. The qualification of the SG flow restrictors is detailed in SER Section 5.4.2.

The staff finds the classification and design code and standard for the integral steam plenum and the SG tubes appropriate because they form part of the RCPB. The staff also finds the classification and design code and standard for the core support mounting brackets appropriate because they are structural attachments to the RPV.

3.9.5.4.1.2 Design—Core Support Structure (Upper Riser Assembly)

FSAR Section 3.9.5.1 states that the URA is located immediately above the lower riser assembly and extends upward to the PZR baffle plate. The upper riser channels the reactor coolant leaving the core upward and permits the reactor coolant to turn in the space above the top of the riser and below the PZR baffle plate. The reactor coolant then flows downward through the annular space outside of the riser and inside of the RPV, where the SG helical tube bundles are located.

The URA hangs from the PZR baffle plate and is supported by the RPV integral steam plenum.

The URA includes a hanger plate with CRD shaft sleeves connecting to the upper riser section. The CRD shaft sleeves attached to the upper CRDS support plate are welded to the upper riser shell. The attachment of the hanger plate to the bottom of the PZR baffle plate is by threaded fasteners. The components in the URA are classified as internal structures constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c).

The interface between the upper and lower riser assemblies. This conical shaped interface is kept closed by the force exerted by a bellows assembly in the upper riser. The upper riser and the lower riser transition that form this interface are both classified as internal structures constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c).

Five CRD shaft supports are bolted to the upper CRDS support plate. The supports are generically referred to as CRD shaft supports; however, each provides support to both the CRD shafts and in-core instrumentation guide tubes (ICIGTs). These supports are part of the URA and are classified as internal structures constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c).

The staff finds the design code for these components as ASME BPV Code, Section III, Subsection NG acceptable and that the code meets GDC 1 because the code provides assurance that these components meet quality standards commensurate with the importance of the safety functions to be performed and are acceptable.

FSAR Section 3.9.5.1 states that there is a bellows assembly in the lower portion of the upper riser to accommodate differential vertical thermal expansion between URA and RPV and additionally provides a downward compressive load on the lower riser assembly to minimize leakage flow between the assemblies.

The bellows is part of the URA. The bellows is located a few inches above the cone structure at the base of the upper riser. The URA, including the bellows, is classified as an internal structure and constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c), and to meet GDC 1 because the code provides assurance that the upper riser and bellows assembly meets quality standards commensurate with the importance of the safety functions to be performed and is acceptable.

The annulus between the upper riser and the vessel wall contains the SG tubes and the tube supports. The upper riser is supported radially by the SG tube supports. The tube supports are stacked to provide radial support to the upper riser. At the base of the SG, there are SG lower tube support cantilever beams that are part of the SG tube support structure.

The staff finds the information provided about the SG tube support structure and upper riser acceptable because the SG tube support structure limits the movement of the upper riser and thus limits the movement of the CRD shaft in the radial direction. Further details about the SG tube support structure are documented in SER Section 5.4.2. In addition, Section 3.9.2 of this report provides more details for the design of the bellows. Specifically, CRD shafts are supported laterally above and below the Upper Riser Bellows, but pass through the Upper Riser Bellows without interference, which satisfies GDC 2.

The evaluation of the upper riser holes is addressed in Section 3.9.2 of this report.

3.9.5.4.1.3 Design—Core Support Structure (Lower Riser Assembly)

FSAR Section 3.9.5.1 states that the lower riser assembly channels the reactor coolant flow leaving the reactor core upward toward the upper riser and separates the flow from the flow outside the lower riser. The lower riser assembly includes the lower riser, upper core plate, CRA guide tube assemblies, and CRA guide tube support plate. The lower riser assembly is located immediately above the CSA and is aligned with and supported on the CSA by four upper support blocks.

The CRA guide tube support plate is classified as an internal structure and is constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c).

The upper core plate is attached to the upper support blocks, supports and aligns the top end of the fuel assemblies. The CRA guide tubes are fastened to the upper core plate, extend upward, where they are fastened to the CRA guide tube support plate. These guide tubes house the portion of the CRAs that extend above the top of the reactor core. The upper core plate is classified as a core support structure and is designed to ASME BPV Code, Section III, Subsection NG.

Based on the information provided by the applicant, the staff finds that the design code for the lower riser assembly meets GDC 1 because the code provides assurance that the lower riser assembly meets quality standards commensurate with the importance of the safety functions to be performed and is acceptable.

3.9.5.4.1.4 Design—Core Support Structure (Core Support Assembly)

FSAR Section 3.9.5.1 states that the CSA includes the core barrel, upper support blocks, lower core plate, shared lower fuel pins and nuts, and reflector blocks. The core barrel is a continuous cylinder with no welds. The upper support blocks are fastened to the core barrel. The lower core plate, which is welded to the bottom of the core barrel, supports and aligns the bottom end of the fuel assemblies.

The core barrel is classified as a core support structure and is designed to ASME BPV Code, Section III, Subsection NG.

The upper support blocks are fastened to the core barrel. The upper support blocks are classified as core support structures and are designed to ASME BPV Code, Section III, Subsection NG.

The lower core plate is welded to the bottom of the core barrel, supports and aligns the bottom end of the fuel assemblies. All the deadweight and other mechanical loads from the fuel are transferred to the upper and lower core plates. The lower core plate is classified as a core support structure and is designed to ASME BPV Code, Section III, Subsection NG. Each of the fuel pins in the lower core plate includes a shaft with threads at the end to be secured with a nut located on the lower side of the lower core plate. The fuel pins, including the nuts, are classified as core support structures and are designed to ASME BPV Code, Section III, Subsection NG.

Section 3.9.2 of this report provides more details about the core support blocks and their attachment mechanism to the lower core plate.

The reflector blocks contain no welds. The reflector blocks are aligned by reflector block alignment pins and stacked on the lower core plate inside the core barrel. The shape of the reflector block assembly closely conforms to the shape of the peripheral fuel assemblies and constrains lateral movement of the fuel assemblies and minimizes the reactor coolant flow that bypasses the fuel assemblies. The fuel is surrounded by a heavy neutron reflector. The heavy reflector, termed reflector blocks, reflects neutrons back into the core to improve fuel performance. The heavy reflector provides the core envelope and directs the flow through the core. The heavy reflector blocks do not provide support to the core and are classified as an internal structure. During seismic and other accident events the heavy reflector limits lateral movement of the fuel assemblies and transfers potential loads to the core barrel assembly. Both alignment pins and reflector blocks are constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c).

During an audit, the staff held multiple discussions with the applicant to gain a better understanding of the mechanism to secure the core support blocks to the lower core plate, as well as the relative movement of the reflector blocks. Both of these issues are discussed in Section 3.9.2 of this report.

Based on the information provided above and information provided in Section 3.9.2 of this report, the staff finds that the design code of standard of the CSA and its interface with other RVIs is acceptable because it meets GDC 1.

3.9.5.4.2 Design—Reactor Vessel Internals Other than Core Support Structures

SRP Section 3.9.5 states that the design of the reactor internals other than core support structures should meet the guidelines of ASME BPV Code, Section III, Subsection NG-3000, and be constructed so as not to adversely affect the integrity of the core support structures.

3.9.5.4.2.1 Design—Reactor Vessel Internals Other than Core Support Structures (Control Rod Assembly Guide Tube)

FSAR Section 3.9.5.1 states that there are CRA guide tubes extend upward to the CRA guide tube support plate. These guide tubes house the portion of the CRAs that extend above the top of the reactor core.

TR-121354-P explains the CRA guide tubes in more detail. Specifically, each CRA guide tube consists of four CRA cards, a CRA lower flange, and an alignment cone. All of these components are welded to the CRA guide tubes.

The CRA guide tube is classified as an internal structure and is constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c). The CRA has fully withdrawn and fully inserted positions.

The staff finds that the design code of the CRA guide tubes meets GDC 1 because the design meets quality standards commensurate with the importance of the safety functions to be performed and is, therefore, acceptable.

3.9.5.4.2.2 Design—Reactor Vessel Internals Other than Core Support Structures (In-Core Instrumentation Guide Tube)

FSAR Section 3.9.5.1 states that an The ICI guide tubes are supported at the top by the CRA guide tube support plate, and at the bottom by the upper core plate, both of which maintain alignment of the ICI guide tubes with their respective fuel assemblies. FSAR Figure 3.93 shows a typical ICIGT.

Each ICIGT is divided into four separate segments to facilitate assembly and disassembly of the NPM. The first segment extends from the instrument seal assemblies on the RPV head, through the PZR region, to the baffle plate at the base of the PZR, terminating in a slip fit. The next segment is connected to the underside of the hanger plate with a socket weld. This segment extends through the length of the upper riser. Within the upper riser, each ICIGT is supported by the five CRD shaft supports. The interface between the ICIGT and the CRD shaft support grid structure is a clearance/slip fit; there is no welding or expansion of the guide tubes at this interface.

The third segment of each ICIGT spans the height of the lower riser. The top end of this segment is welded to the CRA guide tube support plate at the top of the lower riser assembly. The bottom end of these guide tube segments extends below the upper core plate, and have a clearance fit to allow for thermal expansion. The fourth segment is part of the fuel assemblies. The upper three ICIGT segments are classified as internal structures and are constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c).

The staff finds that the design code of the ICIGT meets GDC 1 because the design meets quality standards commensurate with the importance of the safety functions to be performed and is, therefore, acceptable.

3.9.5.4.3 Deflection Limit

Deflection limits are imposed on the RVI components to assure that the CRD shaft is sufficiently aligned so that the capacity to insert the CRAs is not compromised. CRA drop and CRD shaft alignment testing provide data to support determination of specific values for the maximum deflections that allow CRA insertion requirements to be met. The deflection limits include considerations for both fabrication tolerances and static, thermal, and dynamic motion from applicable service loads. The testing includes imposed deflections along the length of the CRD shaft at each of the support locations in the upper riser. Deflections in the lower riser, which contains the CRA guide tubes, are also part of the testing.

FSAR Section 3.9.4 states that in accordance with the technical specifications, the CRDMs are subjected periodically to partial-movement checks to demonstrate the operation of the CRDM and acceptable core power distribution. In addition, drop tests of the CRA are performed as specified in Technical Specification Surveillance Requirement 3.1.4.3 to verify the ability to meet trip time requirements. COL Item 3.9-5 directs a COL applicant to create an operability assurance program for the CRDS. FSAR Section 3.9.4.4 provides a description of the operability assurance program requirements and directs a COL applicant to implement a program. The staff finds the description of the operability assurance program requirements and direction that a COL applicant implement a program in accordance with the requirements in Section 3.9.4.4 of the FSAR and provide a summary of the testing program and results acceptable because it outlines the needed requirements for the program (as provided in SRP Section 3.9.4, Acceptance Criterion 4) and directs a COL applicant to implement such a program and submit its program summary and results, which will be evaluated by the NRC staff as part of the COL application review. Based on the information provided, the staff finds that the information provided by the applicant meets GDC 2 because it ensures the CRD shafts can withstand the effects of natural phenomena, such as earthquakes, without the loss of capability to perform safety functions.

3.9.5.4.4 Asymmetric Blowdown Loads

Because of the integrated nature of the NuScale design, there is no hot leg or cold leg piping attached to the RPV. However, because of the integrated nature of the NuScale design, a pipe break that occurs at the main steamline or FW line may adversely affect the integrity of the RVI components. FSAR Table 3.9-7a, Table 3.9-7b, and 3.9-7c include this potential plant event as a Level D condition. The staff finds this acceptable and that it meets GDC 4 because the RVI components are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including LOCAs. The Level D evaluation for the RVIs under asymmetric loading is documented in Section 3.9.2.4.5 of this report.

3.9.5.4.5 Flow-Induced Vibration

Section 3.9.2 of this report addresses the results of the comprehensive vibration assessment program, including the vibration test program plan for the RVI.

3.9.5.4.6 Other Design Parameters

3.9.5.4.6.1 Other Design Parameters—Core Bypass Flow

The core bypass flow is through two paths: the cooling channels (holes) in the reflector blocks and the fuel assembly guide tubes and instrument tubes as discussed in FSAR Section 4.4.3.1.1. The adequacy of the thermal-hydraulic aspects of core bypass flow is evaluated in Section 4.4 of this report.

3.9.5.4.6.2 Other Design Parameters—Reactor Vessel Internals Gap Fit

The RVIs will be analyzed to different load combinations, including both Service Levels A and B. Service Level A and B events include plant heatup and cooldown. RVIs will be evaluated for Service Level A and B hot-to-cold (and vice versa) transient loading conditions to applicable ASME BPV Code stress limits. This includes consideration of secondary stresses that evolve because of thermal transient events. In addition, deformation limits including consideration of thermal effects will be imposed on the RVIs. Satisfying deformation limits ensures necessary gap fit-up in cases where working clearances are necessary, as merely satisfying the ASME BPV Code stress limits does not ensure required functionality. Based on these design and evaluation requirements, the effects of any possible interference resulting from hot gap and cold gap fit-up are addressed. In addition, the applicant stated that SDAA Part 8, Table 2.1-2 (NPM ITAAC), Items 1 and 3, would provide verification that the ASME BPV Code requirements are satisfied.

The staff understands that satisfying the ASME BPV Code stress limits does not ensure required functionality. By satisfying deformation limits for RVIs during transients, especially those that have a large thermal effect, the applicant has ensured the necessary gap fit-up in cases where working clearances are necessary, or in cases where interference is undesirable. Therefore, because the applicant's analysis accounts for the deformation limits of the RVI in the design process, which in turn ensures the necessary gap fit among the different RVI components where working clearances are necessary, the RVI components' gap fit is acceptable

3.9.5.4.6.3 Other Design Parameters—Core Load Transfer

FSAR Section 3.9.5.1 states that under normal operation, the reactor core is supported by the core support structures of the CSA that surround the fuel assemblies. The deadweight and other mechanical and hydraulic loads from the fuel are transferred to the upper and lower core support plates. The motion of the upper and lower core support plates is coupled through the core barrel. Under seismic and accident conditions, the core barrel transfers lateral loads to the RPV shell through core support mounting brackets at the bottom of the RPV. The vertical loads are transferred from the core barrel to the RPV lower head through the core support mounting brackets.

During normal operation conditions (Level A condition), there are no lateral loads transmitted between the core barrel and the RPV. The core support mounting bracket assemblies located at the bottom of the RPV are welded to the RPV. The core support mounting bracket top plate provides support for the shoulder stud and alignment dowels that hold down the core support. Section 3.9.2 of this report provides more detailed information and the staff's evaluation of the core support assemblies.

3.9.5.4.6.4 Other Design Parameters—Refueling Operation

FSAR Section 3.9.5.1 states that during refueling and maintenance outages, the URA stays attached to the upper section of the NPM (upper CNV, upper RPV, and SG), while providing physical access for potential inspection of the FW plenums, SG, RPV, and CRD shaft supports. The lower riser assembly and CSA remain with the lower NPM (lower CNV, lower RPV, core barrel, and core plates) when the module is parted for refueling and maintenance.

During refueling, the lower CNV section and the lower RPV section (including the fuel, the core support structure, and the lower riser assembly) are separated from the rest of the NPM. The portion of the NPM (after removal from the lower RPV and CNV sections) is referred to as the upper NPM.

The upper NPM is stored in the module inspection rack (MIR). The MIR is located in the RXB pool, within a dry dock area that may be maintained partially or fully flooded as needed to support specific inspection and maintenance activities. While in the MIR, the upper NPM is laterally and vertically supported by the seismic support lugs, spaced 90 degrees apart on the upper CNV, which are normally used to laterally support the NPM in the operating bay. The upper riser is supported in its normal configuration suspended from the PZR baffle plate.

During refueling, the lower riser assembly is located in one of two configurations. The lower riser assembly may be located on top of the CSA (in the same configuration and with the same support as when the NPM is fully assembled), with the CSA placed in the RPV section. The second configuration is used when access to the fuel is needed. In that case, the lower riser assembly is detached from the CSA and lifted and is stored in a designated stand in the refueling pool. While the lower riser assembly is in the stand, it is supported by loadbearing features that prevent the loading of the fuel pins or ICIGTs that protrude below the upper core plate. The lower riser assembly stand is not a safety related component. The staff finds this acceptable because it prevents damage of the fuel pins during refueling.

3.9.5.4.6.5 Other Design Parameters—Stress and Fatigue Analysis

Based on the existence of ITAAC that will ensure compliance with ASME BPV Code requirements (e.g., SDAA Part 8, Table 2.1-2, ITAAC No. 02.01.01), the staff finds that the stress and fatigue analysis of RVIs meets GDC 1 because the design meets quality standards commensurate with the importance of the safety functions to be performed and is, therefore, acceptable.

3.9.5.5 Combined License Information Items

Table 3.9.5-1 lists the COL information item number and description related to the RVIs from FSAR Table 1.8-1.

Table 3.9.5-1: NuScale COL Information Items for Section 3.9.5

Item No.	Description	FSAR Section
COL Item 3.9-7	An applicant that references the NuScale Power Plant US460 standard design will provide a summary of reactor core support structure American Society of Mechanical Engineers (ASME) service level stresses, deformation, and cumulative usage factor values for each component and each operating condition in conformance with ASME Boiler and Pressure Vessel Code Section III Subsection NG.	3.9.5.2
COL Item 3.9-6	An applicant that references the NuScale Power Plant US460 standard design will develop a Reactor Vessel Internals Reliability Program to address industry identified aging degradation mechanism issues.	3.9.5.2

3.9.5.6 Conclusion

Based on the above, including evaluation of the COL Items, the staff finds that the applicant has met the requirements of 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR 50.55a with respect to designing the RVIs to quality standards commensurate with the importance of the safety functions to be performed because the design and construction of the RVIs conform to the requirements of ASME BPV Code, Section III, Subsection NG.

In addition, the staff finds that the applicant has met the requirements of 10 CFR Part 50, Appendix A, GDC 2, 4, and 10, with respect to designing the RVIs to withstand effects of earthquakes and dynamic effects such as postulated pipe rupture, as well as conditions of normal operation, including AOOs.

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

3.9.6.1 Introduction

This section evaluates the descriptions of the functional design, qualification, and inservice testing (IST) programs for pumps, valves, and dynamic restraints (snubbers) used in the NuScale Power Plant, as described in the NuScale SDAA.

3.9.6.2 Summary of Application

The following summarizes its provisions with respect to functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used in the NuScale Power Plant.

SDAA Part 8, “License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC),” specifies ITAAC for as-built components to confirm that their design requirements have been satisfied in the NuScale Power Plant.

FSAR: FSAR Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” describes the functional design and qualification provisions and preservice testing (PST) and IST programs for safety-related valves that are designated as ASME BPV Code Class 1, 2, or 3 and meet the requirements of the ASME OM Code, Subsection ISTA, “General Requirements,” paragraph ISTA-1100, “Scope.” This section also includes valves not categorized as ASME BPV Code Class 1, 2, or 3, but that meet the criteria of the ASME OM Code, Subsection ISTA, paragraph ISTA-1100. Inservice testing of valves is performed in accordance with the ASME OM Code, or as specified by the NRC where relief is granted, or alternatives are authorized by the NRC in accordance with 10 CFR 50.55a.

FSAR Section 3.9.6 references the 2017 Edition of the ASME OM Code in the description of the NuScale IST program. In addition, Section 3.9.6 specifies that the IST plan includes augmented testing for valves that do not meet ASME OM Code, paragraph ISTA-1100, but are relied on in the NuScale safety analyses. FSAR Section 3.9.6 notes that the NuScale Power Plant does not include any pumps or dynamic restraints that perform a specific function identified in the ASME OM Code, paragraph ISTA-1100.

FSAR Section 3.9.6.1, “Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints,” specifies that the functional design and qualification of safety-related valves is performed in accordance with ASME Standard QME-1-2017.

FSAR Section 3.9.6.1 indicates that safety-related valves are designed and provided with access to enable the performance of inservice testing to assess operational readiness in accordance with the ASME OM Code and as defined in the IST program. Section 3.9.6.1 also specifies that the QA requirements for the design, fabrication, construction, and testing of safety-related valves are controlled by the NuScale QA program in accordance with 10 CFR Part 50, Appendix B.

FSAR Section 3.9.6.2, “Inservice Testing of Pumps,” indicates that the NuScale Power Plant design does not contain pumps that meet the criteria of OM Code, paragraph ISTA-1100. Therefore, the NuScale IST program does not include pumps.

FSAR Section 3.9.6.3, “Inservice Testing of Valves,” specifies that valves that meet the criteria of ASME OM Code, paragraph ISTA-1100, are subject to the IST requirements of the ASME OM Code, Subsection ISTC, “Inservice Testing of Valves in Water-Cooled Reactor Nuclear Power Plants.” The FSAR indicates that valves subject to inservice testing include those valves that perform a specific function in shutting down the reactor to a safe-shutdown condition, in maintaining a safe-shutdown condition, or in mitigating the consequences of an accident. FSAR Table 3.9-18, “Valve Inservice Test Requirements per ASME OM Code,” identifies the valves in the NuScale IST program including their description, valve and actuator type, safety position, functions, ASME Class and IST Category, IST type and frequency, and valve grouping. FSAR Table 3.9-19, “Valve Augmented Requirements,” summarizes the augmented testing provisions for specific additional valves. FSAR Section 3.9.6.3 specifies that the NuScale design does not use safety-related, motor-operated valves (MOVs), manual valves, or valves that are actuated by an energy source capable of only one operation (such as a pyrotechnic-actuated (squib) valve).

FSAR Section 3.9.6.3.1, “Valve Functions Tested,” specifies that the NuScale IST plan identifies the intended safety-related functions for valves in NuScale systems. Section 3.9.6.3.1 indicates that an active valve is defined as a valve that is required to change the obturator position to reach its inservice test function position. Section 3.9.6.3.1 also notes that there are no passive valves in the NuScale design that meet the requirements of ASME OM Code, paragraph ISTA-1100.

FSAR Section 3.9.6.3.2, “Valve Testing,” specifies that the testing of valves used in the NuScale Power Plant is described in ASME OM Code, Subsection ISTC, and Appendix I, “Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants,” Appendix II, “Check Valve Condition Monitoring Program,” and Appendix IV (2017 Edition), “Preservice and Inservice Testing of Active Pneumatically Operated Valve Assemblies in Nuclear Power Plants.” Section 3.9.6.3.2 indicates that five types of inservice tests have been identified for the NuScale Power Plant. These types of inservice tests include (1) valve position verification tests, (2) valve leak tests including containment isolation, DHRS boundary, and pressure isolation valves (PIVs), (3) power-operated valve (POV) tests including preservice performance assessment testing and QME-1, exercise tests, skid-mounted components, and performance assessment testing (4) check valve tests, and (5) pressure relief device tests.

FSAR Section 3.9.6.3.3, “Valve Disassembly and Inspection,” specifies that the program for periodic check valve disassembly and inspection includes an evaluation to determine which of the valves identified in the IST plan require disassembly and inspection and the frequency of the inspection.

FSAR Section 3.9.6.3.4, “Valve Accessibility,” specifies that the design of the NuScale Power Plant allows for the ability to access valves for the performance of PST and IST as required by 10 CFR 50.55a and the ASME OM Code. Section 3.9.6.3.4 indicates that valves in the IST plan are located in the following areas: (1) inside containment, (2) CNV head, and (3) RXB.

FSAR Section 3.9.6.4, “Relief Requests and Alternative Authorizations to the Code,” indicates that in the event that compliance with the ASME OM Code is impractical, the licensee will submit a relief request from the OM Code in accordance with 10 CFR 50.55a. If any ASME OM

Code Cases will be implemented as part of the IST plan, Section 3.9.6.4 notes that the Code Cases will have been previously accepted in RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” as incorporated by reference in 10 CFR 50.55a, or will be submitted as an alternative authorization request pursuant to 10 CFR 50.55a(z).

FSAR Section 3.9.6.4.1, “Cold Shutdown Definition Relief Request,” requests relief from paragraph ISTC-3520, “Exercising Requirements,” in the ASME OM Code. This paragraph refers to full-stroke exercise testing at cold shutdown if testing during operation at power is not practical. Section 3.9.6.4.1 proposes that NuScale Mode 3 “safe shutdown with all reactor coolant temperatures < 200 °F” meets the definition of “cold shutdown outage” in paragraph ISTA-2000, “Definitions,” in the 2017 Edition of the ASME OM Code.

FSAR Section 3.9.6.4.2, “Inadvertent Actuation Block Test Frequency Alternate Authorization,” proposes an alternative testing approach for the inadvertent actuation block (IAB) valves in the NuScale ECCS valve system in lieu of the specified testing provisions in ASME OM Code, 2017 Edition. Section 3.9.6.4.3 describes the alternative testing approach to provide reasonable assurance of the satisfactory performance of the IAB valves and an equivalent level of safety in comparison to the ASME OM Code provisions.

FSAR Section 3.9.6.4.3, “Active Hydraulically Operated Valve Alternate Authorization,” proposes an alternate testing approach for the hydraulically operated valve assemblies to apply the ASME OM Code, 2017 Edition, Appendix IV.

FSAR Section 3.9.6.4.4, “Emergency Core Cooling System Valve Alternate Authorization,” proposes an alternate testing approach for the ECCS RRVs and RVVs to apply the ASME OM Code, 2017 Edition, Appendix IV.

FSAR Section 3.9.6.5, “Augmented Valve Testing Program,” specifies that components not required by ASME OM Code, paragraph ISTA-1100, but do serve an augmented function are included in an augmented IST program. Section 3.9.6.5 also specifies that these components will be tested to the intent of the ASME OM Code and applicable addenda, as incorporated by reference in 10 CFR 50.55a, or where the NRC has authorized alternatives in accordance with 10 CFR 50.55a(f) commensurate with its augmented requirements. Section 3.9.6.5 notes that the augmented test requirements for valves are presented in Table 3.9-19.

Other sections of FSAR also specify provisions for various safety-related valves in the NuScale Power Plant design. For example, FSAR Section 3.9.3.2, “Design and Installation of Pressure Relief Devices,” describes the ASME Class 1 pressure-relief devices and ASME Class 2 pressure-relief devices. FSAR Section 3.9.3.3 references ASME Standard QME-1-2017 and the ASME OM Code for valves in the NuScale Power Plant. Section 5.2.2, “Overpressure Protection,” describes the overpressure protection features of each NPM, including the design and operation of the reactor safety valves (RSVs) and RVVs. FSAR Section 6.2.4, “Containment Isolation System,” describes the CNTS, including the design and operation of the CIVs. FSAR Section 6.3, “Emergency Core Cooling System,” describes the ECCS, which provides core cooling during and after AOOs and postulated accidents, including design and operation of the RVVs and RRVs.

ITAAC: SDAA Part 8 includes ITAAC to verify that specific components in the NuScale Power Plant are designed, qualified, and constructed in accordance with the NuScale certified design. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: Part 4 of the NuScale SDAA includes the GTS for the NuScale Power Plant. The NuScale TS include requirements for specific valves to be tested in accordance with the IST program that satisfies 10 CFR 50.55a. The valves specified in the NuScale GTS include the RVVs, RRVs, RSVs, CVCS isolation valves, DHRS actuation valves, CIVs, main steam isolation valves (MSIVs), MSIV bypass valves, feedwater isolation valves (FWIVs), and FW regulation valves.

Technical Reports: The NuScale SDAA does not include technical reports for FSAR Section 3.9.6.

3.9.6.3 Regulatory Basis

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

- GDC 1, as it relates to pumps, valves, and dynamic restraints important to safety being designed, fabricated, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, as it relates to pumps, valves, and dynamic restraints important to safety to withstand the effects of natural phenomena combined with the effects of normal and accident conditions

- GDC 4, as it relates to designing pumps, valves, and dynamic restraints important to safety to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC 14, as it relates to designing pumps, valves, and dynamic restraints that form the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture

- GDC 15, as it relates to pumps, valves, and dynamic restraints that form the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

- GDC 37, “Testing of Emergency Core Cooling System,” as it relates to designing the ECCS to permit periodic functional testing to ensure leaktight integrity and the performance of its active components

- GDC 40, “Testing of Containment Heat Removal System,” as it relates to designing the containment heat removal system to permit periodic functional testing to ensure leaktight integrity and the performance of its active components

- GDC 43, “Testing of Containment Atmosphere Cleanup Systems,” as it relates to designing the containment atmospheric cleanup systems to permit periodic functional testing to ensure leaktight integrity and the performance of the active components

- GDC 46, “Testing of Cooling Water System,” as it relates to designing the cooling water system to permit periodic functional testing to ensure leaktight integrity and performance of the active components

- GDC 54, “Piping Systems Penetrating Containment,” as it relates to designing piping systems penetrating containment with the capability to test periodically the operability of the isolation valves and determine valve leakage acceptability

- 10 CFR Part 50, Appendix B, as it relates to QA in the design, fabrication, construction, and testing of safety-related pumps, valves, and dynamic restraints

- 10 CFR 50.55a(a) through (e), which incorporate the ASME BPV Code and ASME OM Code, as they relate to design, construction, testing, and inspection of pumps, valves, and dynamic restraints

- 10 CFR 50.55a(f) for pumps and valves and 10 CFR 50.55a(g) for dynamic restraints, as they relate to design and accessibility for performance of IST activities

- 10 CFR 52.137, “Contents of Applications; Technical Information,” specifies information to be provided by the applicant for NRC approval of an SDA for reactor design

SRP Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections. In addition, the following documents also provide policy guidance for the PST and IST programs:

- SECY-02-0067, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC),” dated April 15, 2002 (ML020700641), and associated Staff Requirements Memorandum, dated September 11, 2002 (ML022540755)

- SECY-04-0032, “Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses, and Acceptance Criteria,” dated February 26, 2004 (ML040230079), and associated Staff Requirements Memorandum, dated May 14, 2004 (ML041350440)

- SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” dated October 28, 2005 (ML052770257), and associated Staff Requirements Memorandum, dated February 22, 2006 (ML060530316)

3.9.6.4 Technical Evaluation

In accordance with 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” the NRC staff reviewed the design aspects of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints described in the NuScale SDAA. In addition to design aspects, the staff evaluated FSAR Section 3.9.6, and its associated sections to determine whether the FSAR provisions describe the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints sufficiently to satisfy the requirements of NRC regulations and the ASME OM Code as incorporated by reference in the regulations. The staff assessed the adequacy of the NuScale design to ensure that it will provide access to allow the performance of IST activities.

In its review of an SDAA, the staff evaluates whether the application provides assurance that the IST provisions of the ASME OM Code referenced in the FSAR can be performed and that the plant design provides access to permit the performance of IST activities pursuant to

10 CFR 50.55a(f). As part of a COL application review, the staff evaluates whether the COL applicant has fully described the IST program for pumps, valves, and dynamic restraints to demonstrate that the IST program will satisfy the NRC regulations when the program is developed and implemented. In its review of the NuScale SDAA, the staff evaluated the description of the IST program in the FSAR for design aspects of the program, including accessibility for the performance of IST activities, as well as to confirm that the description of the IST program will be acceptable for incorporation by reference in a COL FSAR, in support of a COL application.

FSAR Section 3.9.6 summarizes functional design, qualification, and PST and IST programs for valves to be used in a NuScale Power Plant for a COL applicant that references the NuScale SDA. As part of its review of the SDAA, the NRC staff conducted a regulatory audit with questions on the information provided in the original SDAA submittal. Based on issues discussed in the audit, the applicant revised the SDAA with respect to the requirements for the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints. The applicant also submitted information on the docket to clarify certain provisions in the FSAR. The staff reviewed the revised SDAA and supplemental information to verify that the PST and IST programs were fully described for reference by a COL applicant.

3.9.6.4.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

FSAR Section 3.9.6 describes the functional design and qualification provisions and IST program for the NuScale Power Plant. FSAR Section 3.9.6.1 specifies that the functional design and qualification of safety-related valves is performed in accordance with ASME Standard QME-1-2017, as endorsed in RG 1.100, Revision 3, with clarifications as described in FSAR Section 3.10.2. FSAR Section 3.10.2 indicates that ASME QME-1-2017 is used with the exceptions noted in RG 1.100, Revision 4, for the qualification of active mechanical equipment.

3.9.6.4.2 Inservice Testing Program for Pumps

FSAR Section 3.9.6.2 states that the NuScale Power Plant design does not include pumps that perform a specific function identified in paragraph ISTA-1100 of the ASME OM Code. Therefore, the staff's review of the description of the PST and IST programs to satisfy the ASME OM Code in the FSAR did not address pumps.

3.9.6.4.3 Inservice Testing Program for Valves

The NRC regulations in 10 CFR 50.55a incorporate by reference the ASME OM Code with regulatory conditions for the inservice testing of components in nuclear power plants. The NRC regulations in 10 CFR 50.55a(f) require that valves must be designed and provided with access to enable the performance of inservice testing of valves for assessing operational readiness set forth in the ASME OM Code (or NRC-accepted ASME OM Code Cases), incorporated by reference in 10 CFR 50.55a. The regulations in 10 CFR 50.55a(f)(4) state that valves that are within the scope of the ASME OM Code must meet the IST requirements set forth in the ASME OM Code, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations indicate that the IST requirements for valves that are within the scope of the ASME OM Code but are not classified as ASME BPV Code Class 1, 2, or 3 may be satisfied as an augmented IST program in accordance with 10 CFR 50.55a(f)(6)(ii).

FSAR Section 3.9.6 references the 2017 Edition of the ASME OM Code for the description of the IST program in support of the NuScale SDAA. The staff finds the reference to the 2017 Edition of the ASME OM Code in the FSAR description of the IST program for the NuScale SDA to be acceptable where implemented as incorporated by reference in 10 CFR 50.55a. For COL applicants proposing to implement the NuScale SDA, the NRC regulations in 10 CFR 50.55a(f)(4)(i) require that inservice tests to verify the operational readiness of pumps and valves with a function required for safety conducted during the initial IST Program interval must comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(b) from the time period specified in 10 CFR 50.55a(f)(4)(i) before the date scheduled for initial fuel loading under a COL issued under 10 CFR Part 52 or the optional ASME OM Code Cases listed in RG 1.192, subject to the limitations and modifications listed in 10 CFR 50.55a.

The NRC regulations in 10 CFR 50.55a require that ASME BPV Code Class valves must be designed and provided with access to enable the performance of inservice testing of the valves for assessing operational readiness set forth in the applicable editions and addenda of the ASME OM Code. FSAR Section 3.9.6.3.4 specifies that the design of the NuScale Power Plant allows for the ability to access valves for the performance of PST and IST activities as required by 10 CFR 50.55a and the ASME OM Code. Therefore, the staff has determined that the NuScale SDAA is consistent with the NRC regulatory requirements for design and accessibility of valves to perform the PST and IST activities specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a.

In addition to complying with the provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a, the NRC regulations in 10 CFR 50.55a(b)(3)(iii)(C) require that COL holders, whose initial fuel loading occurs on or after the date 12 months after August 17, 2017, shall monitor flow-induced vibration (FIV) from hydrodynamic loads and acoustic resonance during preservice testing or inservice testing to identify potential adverse flow effects on components within the scope of the IST program. FSAR Section 14.2.1.2, “Preoperational Test Phase Objectives,” states that one objective of the preoperational test phase is to perform inspections or testing for FIV loads on components that must maintain their structural integrity. The COL holder for a NuScale Power Plant will be responsible for addressing compliance with the NRC regulations in 10 CFR 50.55a(b)(3)(iii)(C) for FIV monitoring.

3.9.6.4.3.1 Motor-Operated Valves

The NuScale Power Plant design does not include MOVs that perform a specific function identified in paragraph ISTA-1100 of the ASME OM Code. Therefore, the staff’s review of the description of the PST and IST programs in the FSAR to satisfy the ASME OM Code did not address MOVs.

3.9.6.4.3.2 Inservice Testing Program for Power-Operated Valves Other than Motor-Operated Valves

The staff reviewed the description of the IST program for POVs other than MOVs provided in the NuScale SDAA. In particular, FSAR Section 3.9.6 indicates that POVs other than MOVs in the NuScale Power Plant include Air-Operated Valve (AOVs), Hydraulic Operated Valve (HOVs), and solenoid-operated valves (SOVs). In its review, the staff followed the guidance provided by the Commission for IST programs for new reactors. For example, SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084),” dated May 22, 1995 (ML003708005), includes several provisions to be applied to new reactors with passive emergency cooling systems to provide assurance of proper component performance. SECY-95-132 specifies that these reactor designs should incorporate provisions to test safety-related POVs under design-basis differential pressure and flow. In Staff Requirements Memorandum (SRM)-95-132, dated June 28, 1995 (ML003708019), the Commission approved those provisions and directed the staff to clarify the IST recommendations to demonstrate the design capability of safety-related POVs before installation, to verify valve capability during a preoperational test, and to periodically verify valve capability during the operational phase. In a public memorandum dated July 24, 1995 (ML003708048), the staff provided a consolidated list of the approved policy and technical positions for passive plant designs discussed in applicable Commission papers and their associated SRM. On March 15, 2000, the NRC issued Regulatory Issue Summary (RIS) 2000-03, “Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions” (ML003686003), to discuss the application of lessons learned from valve operating experience and research programs on POVs.

Based on this Commission guidance, the NRC conducted rulemaking to modify 10 CFR 50.55a to specify requirements to provide reasonable assurance that POVs will be capable of performing their safety functions. In addition to complying with the provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a, the NRC regulations in 10 CFR 50.55a(b)(3)(iii)(A) require that COL holders, whose initial fuel loading occurs on or after the date 12 months after August 17, 2017, shall periodically verify the capability of POVs to perform their design-basis safety functions. COL applicants and holders for a NuScale Power Plant will be responsible for addressing the provisions in 10 CFR 50.55a(b)(3)(iii) for new reactors, including the provision in item (A) for periodic verification of POV design-basis capability.

Appendix IV to the 2017 Edition of the ASME OM Code requires quarterly stroke time testing and preservice performance assessment testing for all AOVs within the scope of the IST program and periodic performance assessment testing for AOVs with high safety significance up to a maximum interval of 10 years.

FSAR Section 3.9.6 includes a request in accordance with 10 CFR 50.55a(z) to apply Appendix IV to the ASME OM Code (2017 Edition) to HOVs as an alternative to the ASME OM Code HOV testing requirements. Appendix IV to the 2017 Edition of the ASME OM Code requires quarterly stroke-time testing and preservice performance assessment testing for all AOVs within the scope of the IST program and periodic performance assessment testing for AOVs with high safety significance up to a maximum interval of 10 years. The staff finds the application of the IST provisions in Appendix IV to the ASME OM Code to the AOVs and HOVs in the NuScale Power Plant to provide an acceptable description of the IST program for AOVs and HOVs that satisfies 10 CFR 50.55a(b)(3)(iii)(A) and incorporates the lessons learned for POV performance as discussed in RIS 2000-03.

FSAR Section 3.9.6 describes the use of SOVs as valve subcomponents in POVs in the NuScale Power Plant. For example, paragraph (3), “Power-Operated Valve Tests,” in FSAR Section 3.9.6.3.2, specifies that HOV skid-mounted components include solenoid valves. In discussing POV performance tests, this paragraph indicates that when testing the two redundant, fail-safe hydraulic vent paths on each HOV, a flow device downstream of each solenoid valve verifies that both safety-related SOVs open on the valve stroke to the safe position. In addition, FSAR Table 3.9-17, “Active Valve List,” indicates in Note 2 that trip and reset valves (which are SOVs) are included within each RVV and RRV of the ECCS valve systems. Note 12 in FSAR Table 3.9-18 indicates that each ECCS valve system includes a trip and reset valves. Subsection ISTC in the ASME OM Code (2017 Edition) allows subcomponent valves (such as internal SOVs) to be demonstrated to perform adequately as part of testing of the main valve. The COL holder for a NuScale Power Plant will be responsible for satisfying 10 CFR 50.55a(b)(3)(iii)(A) for SOVs as valve subcomponents of specific POVs.

The NRC staff considers the description of the IST program for POVs other than MOVs to be consistent with the NRC regulations in 10 CFR 50.55a and Commission guidance. The NRC review of specific POVs, such as the ECCS valves and CIVs, and their IST provisions is discussed later in this SER section.

3.9.6.4.3.3 Inservice Testing Program for Check Valves

Paragraph (4), "Check Valve Tests," in FSAR Section 3.9.6.3.2, describes the IST program for check valves in the NuScale Power Plant. FSAR Section 3.9.6.3.2 indicates that there are four check valves for each NPM in the NuScale IST plan. FSAR Section 3.9.6.3.2 indicates that the check valves will be grouped when applying the ASME OM Code, Appendix II. FSAR Table 3.9-18 identifies the NuScale check valve test frequencies consistent with the ASME OM Code as incorporated by reference in 10 CFR 50.55a.

In addition to complying with the provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a, the NRC regulations in 10 CFR 50.55a(b)(3)(iii)(B) require that COL holders whose initial fuel loading occurs on or after the date 12 months after August 17, 2017, shall perform bidirectional testing of check valves within the IST program where practicable. FSAR Section 3.9.6.3.2 specifies that check valves will be exercised to both the open and closed positions regardless of their safety function position.

Based on its review, the staff finds that the description in FSAR Section 3.9.6.3.2 of the IST program for check valves is consistent with the ASME OM Code as incorporated by reference in 10 CFR 50.55a and therefore is acceptable.

3.9.6.4.3.4 Pressure Isolation Valve Leak Testing

Paragraph (2), "Valve Leakage Tests," in FSAR Section 3.9.6.3.2 describes leakage tests for various types of valves in the NuScale Power Plant. Paragraph (2) of Section 3.9.6.3.2 specifies that the NuScale design does not use pressure isolation valves that provide isolation between high- and low-pressure systems. FSAR Section 3.9.6.3.2 indicates that eight safety-related CVCS CIVs perform RCS isolation functions and specifies that NuScale TS 3.4.6, "Chemical Volume and Control System (CVCS) Isolation Valves," controls the RCS pressure isolation function. FSAR Section 3.9.6.3.2 indicates that the NuScale Power Plant design does not incorporate dedicated PIVs.

The staff finds the description in the SDAA of the function of the CIVs to provide pressure isolation to be acceptable for the NuScale IST program, because the plant design does not include PIVs. The staff describes its review of CIV leak testing later in this SER section.

3.9.6.4.3.5 Containment Isolation Valve Leak Testing

Paragraph (2) in FSAR Section 3.9.6.3.2 specifies that CIVs are leak tested in accordance with 10 CFR Part 50, Appendix J, "and paragraph ISTC-3620, "Containment Isolation Valves," of the ASME OM Code. The staff finds the reference in NuScale FSAR Section 3.9.6.3.2 to 10 CFR

Part 50, Appendix J, and the ASME OM Code as incorporated by reference in 10 CFR 50.55a for leak testing of CIVs in the NuScale IST program to be acceptable. The staff review of the CIV design is described later in this SER section.

3.9.6.4.3.6 Inservice Testing Program for Safety and Relief Valves

Paragraph (5), "Pressure Relief Device Tests," in FSAR Section 3.9.6.3.2 specifies that the PST and IST provisions for the pressure-relief devices are specified in Appendix I to the ASME OM Code. The NRC regulations in 10 CFR 50.55a incorporate by reference the testing provisions in Appendix I to the ASME OM Code for safety and relief valves. Therefore, the staff finds that the NuScale SDAA is consistent with the NRC regulatory requirements for safety and relief valves. The staff review of the RSV design is described later in this SER section.

3.9.6.4.3.7 Manually Operated Valves

The NuScale Power Plant design does not include manually operated valves that perform a specific function identified in paragraph ISTA-1100 of the ASME OM Code. Therefore, the staff's review of the description of the PST and IST programs in the FSAR did not address manually operated valves.

3.9.6.4.3.8 Pyrotechnic-Actuated Valves

The NuScale Power Plant design does not use pyrotechnic-actuated (squib) valves. Therefore, the staff's review of the description of the PST and IST programs in the FSAR did not address squib valves.

3.9.6.4.3.9 Rupture Disks

FSAR Table 3.9-18 specifies that the NuScale Power Plant includes two rupture disks as ASME OM Code, Category D, valves. Note 21 to Table 3.9-18 states that these nonreclosing devices open to relieve pressure inside the RXB to maintain structural integrity during design-basis events, and that these non-Code Class components meet the criteria of ISTA-1100. The staff has no objection to this approach.

3.9.6.4.3.10 Inservice Testing Program Tables

As part of its review of the IST program for the NuScale Power Plant, the staff evaluated whether the applicant has properly specified the provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a in the NuScale IST program tables. Based on the FSAR, the staff finds that the NuScale IST program tables incorporate the provisions of the ASME OM Code (2017 Edition) as incorporated by reference in 10 CFR 50.55a, with the applicable granted relief and authorized alternative by the NRC staff in this SER section.

3.9.6.4.4 Dynamic Restraints

FSAR Section 3.9.6 specifies that the NuScale Power Plant design does not use dynamic restraints (snubbers) that perform a specific function identified in ASME OM Code, paragraph ISTA-1100. Therefore, the staff's review of the description of the PST and IST programs in the FSAR did not address snubbers.

3.9.6.4.5 Relief Requests and Alternative Authorizations to the ASME OM Code

The NRC regulations allow an applicant or licensee to submit a request for relief from or an alternative to the provisions of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The NRC regulations in 10 CFR 50.55a(f)(5) and (6) specify the requirements for submittal of a request for relief from the ASME OM Code provisions and for the NRC to grant the requested relief. The NRC regulations in 10 CFR 50.55a(z) specify the requirements for submittal of a request for an alternative to the ASME OM Code provisions and for the NRC to authorize the requested alternative. In its SDAA, the applicant submitted one request for relief from the ASME OM Code provisions, and three requests for alternatives to the ASME OM Code. The NRC staff review of these requests is described below.

3.9.6.4.5.1 Cold Shutdown Definition Relief Request

FSAR Section 3.9.6.4.1 requests relief from the ASME OM Code with respect to the use of the term “cold shutdown” in the ASME OM Code (2017 Edition) under 10 CFR 50.55a(f)(5) as use of the OM Code term is impractical for NuScale operating modes. For example, Section 3.9.6.4.1 specifies that the NuScale TS do not include a Mode defined as “cold shutdown” as used in the ASME OM Code. In particular, Table 1.1-1, “Modes,” in SDAA Part 4, Volume 1, “US460 Generic Technical Specifications,” lists the five Modes for the NuScale Power Plant as follows: Mode 1 (Operations), Mode 2 (Hot Shutdown), Mode 3 (Safe Shutdown), Mode 4 (Transition), and Mode 5 (Refueling). FSAR Section 3.9.6.4.1 indicates that the NuScale Power Plant modes of operation differ from the standard TS for other PWRs. For example, Mode 3 (Safe Shutdown) for the NuScale Power Plant occurs with the reactivity condition of K_{eff} of less than 0.99, and all reactor coolant temperatures less than 216 degrees C (420 degrees F). The applicant also stated that containment and containment isolation operability are required at temperatures greater than or equal to 93 degrees C (200 degrees F). As more appropriate for the NuScale Power Plant, the applicant referenced “cold shutdown outage” as defined in paragraph ISTA-2000 of the ASME OM Code (2017 Edition) that applies to each nonrefueling outage period in which the cold shutdown mode, as defined by the plant TS, is entered. The applicant proposed that the “safe shutdown condition with reactor coolant temperatures < [93.3 degrees C] 200 °F,” where the NPM is stable, important safety systems are not required, and cold shutdown testing can commence for the NuScale Power Plant, is equivalent to the “cold shutdown outage” condition as defined in the ASME OM Code (2017 Edition).

The NRC regulations in 10 CFR 50.55a(f)(6) specify that the Commission will evaluate determinations that Code requirements are impractical and may grant relief and impose alternative requirements. The staff has determined that the applicant has justified that the requirement for the use of a “cold shutdown” testing interval in the ASME OM Code (2017 Edition) is impractical for the design and operating modes of the NuScale Power Plant. The staff finds that the alternative proposed by the applicant for the use of “safe shutdown with all reactor coolant temperatures < [93.3 degrees C] 200 °F” is acceptable for providing assurance that the NuScale Power Plant is in a safe-shutdown condition during the performance of IST activities consistent with the “cold-shutdown outage” definition in the ASME OM Code (2017 Edition).

Therefore, for the reasons described above, the staff concludes that the “cold shutdown definition” relief request in FSAR Section 3.9.6.4.1 may be granted in that it satisfies 10 CFR 50.55a(f)(6) because the alternative requirements are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden that could result if the requirements were imposed on the facility.

3.9.6.4.5.2 Active Hydraulically Operated Valve Alternative Authorization

FSAR Section 3.9.6.4.3 includes a request by the applicant to apply Appendix IV of the ASME OM Code (2017 Edition) for HOVs as an alternative to the ASME OM Code requirements under 10 CFR 50.55a(z)(1) as providing an acceptable level of quality and safety. In this FSAR section, the NuScale applicant notes that pursuant to 10 CFR 50.55a(b)(3)(iii)(A), new reactor licensees must periodically verify the capability of POVs to perform their design-basis safety functions. Comprehensive valve testing is required for active POVs to provide reasonable assurance that these components can perform their inservice functions during all design-basis conditions. FSAR Section 3.9.6.4.3 states that Appendix IV provides a level of safety equivalent to Subsection ISTC in the ASME OM Code and 10 CFR 50.55a(b)(3)(iii)(A) in demonstrating that HOVs can perform their safety function under design-basis conditions.

In the past, the ASME OM Code as incorporated by reference in 10 CFR 50.55a required stroke-time testing of all POVs within the scope of the ASME OM Code on a quarterly interval to assess their operational readiness in nuclear power plants. Valve operating experience and testing programs revealed significant weaknesses in the capability of stroke-time testing to identify performance issues with certain POVs. Therefore, ASME prepared Appendix III, "Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Water-Cooled Reactor Nuclear Power Plants," to the ASME OM Code (2009 Edition), and Appendix IV to the ASME OM Code (2017 Edition) to apply diagnostic testing to assess the operational readiness of MOVs and AOVs, respectively. As noted above, the NuScale Power Plant does not include any MOVs within the scope of the ASME OM Code. To incorporate lessons learned from valve operating experience and testing programs, Appendix IV to the ASME OM Code (2017 Edition) requires quarterly stroke-time testing and preservice performance assessment testing for all AOVs within the scope of the IST program and periodic performance assessment testing for AOVs with high safety significance up to a maximum interval of 10 years. As indicated above in this SER section, the NRC issued RIS 2000-03 to discuss the lessons learned from valve operating experience and research programs for POVs in nuclear power plants. The application of Appendix IV to the ASME OM Code (2017 Edition) as part of the IST program for POVs in the NuScale Power Plant incorporates the lessons learned from valve operating experience and research programs described in RIS 2000-03.

The NRC regulations in 10 CFR 50.55a(z) allow applicants and licensees to submit requests for the use of alternatives to specific 10 CFR 50.55a requirements when authorized by the NRC staff. In 10 CFR 50.55a(z), the regulations specify that the applicant or licensee must demonstrate that (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on its review, the staff finds that the request by the applicant to apply Appendix IV to the 2017 Edition of the ASME OM Code as an alternative to the IST requirements for HOVs in Subsection ISTC of the 2017 Edition of the ASME OM Code will improve the IST activities to provide increased assurance of the operational readiness of POVs in the NuScale Power Plant by the performance of initial diagnostic testing of all HOVs with periodic diagnostic testing of high-safety significant HOVs, in addition to the current ASME OM Code requirements for quarterly HOV stroke-time testing. Therefore, the staff concludes that the Appendix IV alternative request proposed in FSAR Section 3.9.6.4.3 for the HOVs in the NuScale Power Plant may be authorized as it provides an acceptable level of quality and safety in satisfying the requirements in 10 CFR 50.55a(z)(1).

3.9.6.4.5.3 Inadvertent Actuation Block Valve Test Frequency Alternative Authorization

FSAR Section 3.9.6.4.2 states that the inadvertent actuation block (IAB) valve test frequency authorization request is submitted under 10 CFR 50.55a(z)(1) as providing an acceptable level of quality and safety in lieu of the ASME OM Code requirements for the IAB valve. The NuScale applicant notes that pursuant to 10 CFR 50.55a(b)(3)(iii)(A), new reactor licensees must periodically verify the capability of POVs to perform their design-basis safety functions. Comprehensive valve testing is required for active power-operated relief valves (PORVs) to provide reasonable assurance that these components can perform their inservice functions during all design-basis conditions. The IAB valve is a subcomponent of the ECCS Reactor Recirculation Valve (RRV). Since it is an additional design feature not found on typical PORVs, additional clarification is needed to define testing that will provide an acceptable level of quality and safety compared to the ASME OM Code requirements. Testing of the IAB valve includes a performance assessment test that is part of the comprehensive testing of the ECCS RRV. ASME OM Code, Mandatory Appendix IV, provides a framework for testing ECCS RRV subcomponents. Therefore, the applicant submitted this alternative request for the IAB valve testing.

FSAR Section 3.9.6.4.2 notes that ECCS valves are required to meet ASME OM Code, Subsection ISTC, paragraph ISTC-3100, “Preservice Testing,” at conditions as near as practicable to those expected during subsequent inservice testing. Section 3.9.6.4.3 also notes that ECCS valves are required to meet ASME OM Code, Subsection ISTC, paragraph ISTC-3200, “Inservice Testing,” when the valves are required to be operable to fulfill their required functions. Appendix IV to the ASME OM Code (2017 Edition) includes paragraph IV-3300, “Preservice Testing,” and paragraph IV-3400, “Inservice Testing,” for provisions related to stroke testing, performance assessment testing, fail-safe testing, leak testing, and position verification testing, as applicable.

As a proposed alternative to the ASME OM Code requirements, FSAR Section 3.9.6.4.2 includes a request that (1) the preservice testing for the IAB valves in the RRVs for the ECCS valve system will meet the requirements of paragraph ISTC-3100, and (2) the IST frequency for the IAB valves for the initial and subsequent NPMs will be implemented as described in the FSAR to provide an equivalent level of safety for the IAB valves. For preservice testing of the RRV IAB valves, FSAR Section 3.9.6.4.2 specifies that all IAB valves shall be tested (1) to verify the IAB minimum analyzed closing threshold pressure and (2) to verify the IAB valve opening release pressure to be within the analyzed range. For the proposed IST alternative method, FSAR Section 3.9.6.4.3 specifies that IAB valves shall be tested (1) to verify the IAB minimum analyzed closing threshold pressure and (2) to verify the IAB opening release pressure to be within the analyzed range. For the alternative IST frequency for IAB valves, FSAR Section 3.9.6.4.2 specifies the following: (1) for the first NPM (initial NPM of the initial NuScale Power Plant) during the first refueling outage (RFO), all IAB valves will be tested, (2) for the first NPM during the second RFO, one IAB valve will be sample tested, (3) for the follow-on NPMs during their first RFO (if prior to the second RFO for the first NPM), one IAB valve will be sample tested, (4) for all NPMs after the second RFO of the first NPM, IAB valve test frequency shall be established per the requirements of ASME OM Code (2017 Edition), Appendix IV, paragraph IV-

3410, and (5) Performance Assessment Testing will be performed as an alternative to ASME OM Code, Appendix IV, paragraph IV-3420, “Stroke Testing.”

FSAR Section 3.9.6.4.3 states that during plant shutdown, the RVVs and RRVs are exercise tested, fail-safe tested, and position verification tested. This testing demonstrates the safety functions for the ECCS main valve and trip valve but does not verify the threshold and release pressures of the RRV IAB valves. In this instance, the IAB valves are treated as skid-mounted components as defined in the ASME OM Code. However, the IAB valve function is not demonstrated during the ECCS valve exercise testing. FSAR Section 3.9.6.4.2 indicates that it is not practicable to test the IAB valves during normal plant operation. In accordance with the specified IST schedule, the IAB valves will be removed during the RFO and bench tested to verify threshold and release pressures to demonstrate IAB valve functionality. The applicant asserted that the IST alternate testing and frequencies will provide reasonable assurance of the satisfactory performance of the IAB valve and an equivalent level of safety to ASME OM Code, Subsection ISTC, paragraph ISTC-3510, “Exercising Test Frequency.” Note 16 in FSAR Table 3.9-18 indicates that FSAR Section 3.9.6.4.2 will be used to determine the ECCS IAB test method and frequency.

The NRC regulations in 10 CFR 50.55a(z) allow applicants and licensees to submit requests for the use of alternatives to specific 10 CFR 50.55a requirements when authorized by the NRC staff. In 10 CFR 50.55a(z), the regulations specify that the applicant or licensee must demonstrate that (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff reviewed the request by the applicant to apply an alternative IST method and frequency for the RRV IAB valves from the ASME OM Code requirements to determine whether the requirements in 10 CFR 50.55a(z) are satisfied. As noted in FSAR Section 3.9.6.4.2, the applicant groups the ECCS valves on a plantwide (multimodule) basis to optimize testing, examination, and maintenance activities. When six NPMs are installed in a NuScale power plant, three RFOs will occur annually, and a significant amount of data will be developed from the testing of the IAB valves in accordance with the IST frequency proposed by the applicant. For example, all RRV IAB valves will be removed and bench tested during the first RFO of the first NPM with sampling of IAB valves for future RFOs of the first NPM and subsequent NPMs. The staff finds that the proposed IST method and frequency for the IAB valves is an acceptable alternative to the ASME OM Code requirements for performance assessment testing and stroke testing that provides reasonable assurance of the capability of the IAB valves to perform their safety functions. Therefore, the staff concludes that the alternative request for IST methods and frequency proposed in FSAR Section 3.9.6.4.3 for the IAB valves in the NuScale Power Plant may be authorized as it provides an acceptable level of quality and safety in satisfying the requirements in 10 CFR 50.55a(z).

3.9.6.4.5.4 Emergency Core Cooling System Valve Alternative Authorization

In FSAR Section 3.9.6.4.4, the applicant submitted an ECCS valve alternative authorization request under 10 CFR 50.55a(z)(1) as providing an acceptable level of quality and safety lieu of the ASME OM Code testing requirements for the ECCS valves in the NuScale reactor.

In FSAR Section 3.9.6.4.4, the applicant notes that pursuant to 10 CFR 50.55a(b)(3)(iii)(A), new reactor licensees must periodically verify the capability of POVs to perform their design-basis safety functions. In addition, the applicant states that comprehensive valve testing is required for active PORVs to provide reasonable assurance that these components can perform their inservice functions during all design-basis conditions. In the NuScale Power Plant, the ECCS valves have additional design features beyond typical PORVs. ASME OM Code Mandatory Appendix IV specifies requirements for comprehensive testing of the ECCS RRVs and RVVs; however, the applicant indicates that Subsection ISTC of the ASME OM Code does not provide a means for utilizing this method. Thus, the applicant submitted an alternative request in lieu of the ASME OM Code requirements for the ECCS valves.

The applicant states that the ECCS valves are PORVs with additional features that are subject to testing pursuant to the requirements of Subsection ISTC. The ECCS valves in the NuScale Power Plant are high safety-significant components that do require comprehensive testing to provide reasonable assurance that the valves will perform under design-basis conditions. Unlike AOVs or MOVs, no Mandatory Appendix exists to assess the operational readiness of certain active PORVs in water-cooled reactor nuclear power plants. A testing methodology for ECCS valves is required to provide reasonable assurance that ECCS valves will actuate under all design-basis conditions.

The applicant proposes an alternative to apply Mandatory Appendix IV for ECCS RRVs and RVVs that meet the criteria of paragraph ISTA-1100 of the ASME OM Code, Subsection ISTA. The applicant asserts that the application of Mandatory Appendix IV to ECCS RRVs and RVVs will provide a level of safety equivalent to Subsection ISTC and as required by 10 CFR 50.55a(b)(3)(iii)(A). This alternative will facilitate a NuScale Power Plant licensee to periodically verify the capability of ECCS valves to perform their design-basis safety functions. This is accomplished through the comprehensive testing of these components pursuant to Mandatory Appendix IV to provide reasonable assurance that they will fulfill their safety functions under all design-basis conditions. The applicant outlines the details of the scope of this request in FSAR Section 3.9.6.4.4.

As described in FSAR Section 6.3.2.2, the ECCS RRV and RVV (Section 6.3.2.2) are PORVs that do not share design similarities with pneumatically operated valves. Therefore, the differences need to be considered when applying Mandatory Appendix IV criteria for comprehensive testing of RRVs and RVVs. Performance assessment tests of RRVs and RVVs under the alternative are applied to Paragraph IV-3300 for Preservice Testing. These tests are different from what would be performed for an AOV; however, the specified tests are applicable to the ECCS design and provide reasonable assurance that RRVs and RVVs will perform their intended safety function under design basis conditions.

The applicant stated that the ECCS valves are grouped plant-wide (multi-module) to optimize testing, examination, and maintenance activities. This test frequency, combined with the population in each of the valve groups, provides assurance that there is sufficient data to confirm RRV and RVV performance. When six NPMs are installed and operating, three refueling outages will occur annually. The applicant asserts that the frequencies established by the criteria of OM Code Mandatory Appendix IV provide reasonable assurance for the satisfactory performance of the RRVs and RVVs and an equivalent level of performance testing as initially established in 10 CFR 50.55a(b)(3)(iii)(A).

The NRC regulations in 10 CFR 50.55a(z) allow applicants and licensees to submit requests for the use of alternatives to specific 10 CFR 50.55a requirements when authorized by the NRC staff. In 10 CFR 50.55a(z), the regulations specify that the applicant or licensee must demonstrate that (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Based on its review, the NRC staff finds that the request by the applicant to apply alternative testing for the ECCS valves from the ASME OM Code requirements satisfies 10 CFR 50.55a(z).

3.9.6.4.6 Specific Valve Review

Several sections of the FSAR specify provisions for various safety-related valves in the NuScale Power Plant design. For example, Section 5.2.2 describes the overpressure protection features of each NPM, including the design and operation of the RSVs and RVVs. Section 6.2.4 describes the CNTS, including the design and operation of the CIVs. Section 6.3 describes the ECCS, which provides core cooling during and after AOOs and postulated accidents, including design and operation of the RVVs and RRVs. The staff reviewed the functional design, qualification, and IST provisions for the safety-related valves in these FSAR sections as discussed in the following paragraphs.

3.9.6.4.6.1 Emergency Core Cooling System Valves

FSAR Section 6.3 specifies that the ECCS serves three fundamental purposes: (1) to function as part of the RCPB, (2) to cool the reactor core in situations when it cannot be cooled by other means, such as a LOCA inside the CNV, and (3) to provide low-temperature overpressure protection (LTOP) for the RPV. The ECCS valves include two RVVs at the top of the RPV and two RRVs on the side of the RPV above the active fuel level. FSAR Section 5.2.2.2.2, “Low Temperature Overpressure Protection System,” specifies that the RVVs are designed with sufficient capacity to prevent RCPB pressure from exceeding the limiting pressure when below the LTOP enabling temperature, such that the RPV is maintained below brittle fracture stress limits during operating, maintenance, testing, or postulated accident conditions.

The NRC staff reviewed the FOAK design of the NuScale ECCS valves based on the applicable regulations for a new reactor with passive means to accomplish the safety functions of emergency core cooling. Under 10 CFR 52.137(b), an SDA applicant that uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design except for site-specific elements and must meet the requirements of 10 CFR 50.43(e). The NRC regulations in paragraph (e) of 10 CFR 50.43, “Additional Standards and Provisions Affecting Class 103 Licenses and Certifications for Commercial Power,” require that applications that propose nuclear reactor designs which use simplified, inherent, passive, or other innovative means to accomplish safety functions will be approved only if (1) the performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof, (2) interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof, and (3) sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. The applicant for a NuScale SDA provided information to support compliance with 10 CFR 50.43(e) for the NuScale Power Plant.

As part of the SDAA review of the NuScale ECCS valve system, the NRC staff conducted a detailed review and audit of the ECCS valve design and associated tests to evaluate the FOAK design of the NuScale ECCS valves in support of compliance with 10 CFR 52.137(b) and 10 CFR 50.43(e).

The NRC staff review and audit of the ECCS valve system is described in this section of the SE. During the audit, the NRC staff requested that the applicant describe the specific tests and their results to demonstrate compliance with 10 CFR 50.43(e) for the SDA design of the ECCS valve system, including performance of the RVVs and RRVs, including the applicable venturi, and performance of the IAB valve used with the RRVs, consistent with the specifications in the SDAA. The applicant stated that an ECCS Valve Proof Test Program was completed with specific tests of the ECCS valves in support of the SDAA. The applicant reported that the ECCS Valve Proof Test Program incorporated design changes, such as removal of the RVV IABs, the addition of venturis, and modification of IAB threshold and release pressures from the previous Design Certification Application (DCA) program for the ECCS valve system in the NuScale US600 design.

During the audit, the applicant provided the following information:

1. Two ECCS test programs were conducted prior to the ECCS Valve Proof Test Program for the NuScale SDAA. The first program included proof-of-concept testing to support the design and development of the ECCS valves, and was performed in 2015 at the Target Rock facility. The second test program included demonstration testing of the ECCS valve design to support system functionality for the US600 DCA and was performed in 2019 at the Target Rock facility. The ECCS Valve Proof Test Program built upon the 2019 test program, and provides assurance that the ECCS valve designs will function as required by the design specification, and will meet the functional qualification requirements of the ASME QME-1 Standard. The ECCS Valve Proof Test Program evaluated aspects of future full qualification activities while also providing key performance data for safety analyses insights with testing performed according to the NuScale test specification.

   As part of its audit, the NRC staff reviewed the applicant’s proprietary documents describing the test requirements, test specifications, and test results for the ECCS Valve Proof Test Program for the NuScale SDAA. For example, the NRC staff reviewed the ECCS Valve Proof Testing Requirements Document, NPM-20, ER-ER-B020-5287, Revision 5, dated March 24, 2021, which specifies the testing requirements for the ECCS Valve Proof Test Program. The purpose of the ECCS Valve Proof Test Program was to provide assurance that the ECCS valve design will function as required by the design specification, and will meet the functional qualification requirements in the ASME QME-1 Standard.

   The NRC reviewed additional proprietary documents from the ECCS Valve Proof Test Program. Non-proprietary summaries of these documents are as follows:

   > Test Specification – ECCS Valve Proof Testing, TSD-TSD-B020-7651, Revision 3, dated March 30, 2021, specifies the testing requirements for the ECCS Valve Proof Test Program to evaluate the functional performance and design characteristics of the ECCS valves at representative operating conditions. The applicant developed the testing techniques from accepted industry standards, including IEEE 344-2004 and ASME QME-1-2017.
   >
   > The Final Test Report for the Reactor Vent Valve Assembly (ECCS-RVV-ETV), NuScale ECCS Valve Proof Testing, FTR-123192, Revision 0, dated November 9, 2022, Curtiss-Wright TRP No. 10444, Revision A, dated October 10, 2022, describes the tests conducted on the RVV Engineering Test Valve Model ECCS-RVV-ETV. For this segment of the program, Valve Model ECCS-RVV-ETV underwent a thorough series of tests including flow capacity testing, seat leakage testing, functional testing (reset,

passive opening, and actuation), jet diffuser testing, and fundamental frequency testing. Valve Model ECCS-RVV-ETV met the acceptance criteria for these tests without unexpected rework or design changes. The design, material selection, and theory of operation for Valve Model ECCS-RVV-ETV proved to be well suited for the operating conditions and anticipated inservice modes in a NuScale Power Module (NPM).

The Final Test Report for the Reactor Recirculation Valve Assembly (ECCS-RRV-ETV), NuScale ECCS Valve Proof Testing, FTR-123133, Revision 0, dated October 17, 2022, Curtiss-Wright TRP No. 10443, Revision A, dated September 28, 2022, describes the tests conducted on the RRV Test Valve Model ECCS-RRV-ETV, including the IAB Valve ECCS-IAB-001. For this segment of the program, Valve Model ECCS-RRV-ETV underwent a thorough series of tests including flow capacity testing, seat leakage testing, functional testing (reset, passive opening, IAB functional, and actuation), and fundamental frequency testing. Valve Model ECCS-RRV-ETV met the acceptance criteria for these tests without unexpected rework or design changes. The design, material selection, and theory of operation for Valve Model ECCS-RRV-ETV proved to be well suited for the operating conditions and anticipated inservice modes in an NPM.

The Final Test Report for the Trip/Reset Valve Assembly (ECCS-TRV-ETV), NuScale ECCS Valve Proof Testing FTR-120070, Revision 0, dated July 14, 2022, Curtiss Wright TRP No. 10445, Revision A, dated July 14, 2022, describes the tests conducted on the Trip/Reset Valve Model ECCS-TRV-ETV. The ECCS-TRV-ETV was also used to actuate the ECCS-RRV-ETV and ECCS-RVV-ETV during their respective functional tests. For this segment of the program, Valve Model ECCS-TRV-ETV underwent a thorough series of tests including seat leakage testing and fundamental frequency testing, and a complete modal analysis of the valve structure to detect and characterize critical resonant frequencies. Valve Model ECCS-TRV-ETV met the acceptance criteria for these tests. This testing of Valve Model ECCS-TRV-ETV identified two fundamental frequencies less than 100 Hertz, and an improvement opportunity for the valve design to facilitate future maintenance activities.

2. In its response to NRC staff questions during the audit, the applicant described the results of the proof testing of the various valves within the ECCS valve system. This information included the significant fundamental frequencies (modes) identified during the testing, and the plans to evaluate the modal response against the final in-structure response spectra. In light of valve leakage during the tests, the applicant stated that limiting leakage is not a safety function of the ECCS valve, but described the ECCS valve design specification that identifies a requirement that the valve supplier perform cycle testing along with ASME QME-1 testing to determine acceptable replacement life of valve component parts and the sealing capability of the valve assembly. The applicant described the Cv flow capacity testing and pressure differential ratio factor xT flow capacity testing for the applicable ECCS valves, and the correlation of the results to plant conditions. The applicant discussed the plans for individual production testing of each IAB valve for the RRVs in the ECCS valve system based on the IAB adjustments necessary during the valve proof testing. The applicant discussed the stroke-time test results, such as fast opening strokes that were addressed by sensitivity calculations, with future work planned to address the stroke-time test results during the qualification program.

3. The applicant did not specifically include boric acid testing when demonstrating compliance with 10 CFR 50.43(e) for the NPM-20 in the SDAA. The US600 DCA Demonstration Test Program for the ECCS valve system had included the performance of boric acid effects

testing to demonstrate that the ECCS valve system would operate properly with boric acid solutions as part of its safety function. In the DCA program, the applicant evaluated the ECCS valve system using boric acid solution with concentrations, temperatures, and pressures representative of the plant conditions. These test results demonstrated that the presence of boric acid did not have a noticeable impact on valve performance for the US600 reactor as compared to water with no boric acid. Based on the boric acid effects test results for the ECCS valve system in the DCA program, the applicant used water without boric acid for proof testing of the ECCS valve system for the SDA application. The applicant states that the functional design and qualification of safety-related valves, including the ECCS valve system, will be performed in accordance with ASME Standard QME-1-2017, which is accepted in RG 1.100 (Revision 4) with applicable regulatory positions.

FSAR Section 3.9.6.3.2(3), "Power-Operated Valve Tests," the applicant describes the activities to satisfy 10 CFR 50.55a(b)(3)(iii)(A) for all POVs to be used in the NuScale Power Plant with the applicable justification. The NRC staff finds FSAR Section 3.9.6.3.2(3) to be acceptable.

The NRC regulations in 10 CFR 50.55a(f)(4)(i) require that the holder of a COL under 10 CFR Part 52 must implement the latest edition of the ASME OM Code incorporated by reference in 10 CFR 50.55a no more than 18 months before the date scheduled for initial loading of fuel. NRC Regulatory Issue Summary (RIS) 2012-08 (Revision 1), "Developing Inservice Testing and Inservice Inspection Programs Under 10 CFR Part 52," discusses the process for a COL applicant to request to use the edition of the ASME OM Code that was specified in the original application. During the audit, the NRC staff requested the applicant to describe plans to inform future COL applicants of this NRC guidance. The applicant stated that the FSAR Section 3.9.6 references the 2017 Edition of the ASME OM Code. The applicant noted that RIS 2012-08, Revision 1, discusses how a COL applicant may request the use of an ASME OM Code edition specified in the original application. The applicant stated that licensees referencing the US460 design will have the option to either use the edition of the ASME OM Code in the US460 design, or use a later edition. The applicant recommended that applicants select an ASME OM Code edition to cover the startup of all six modules under same edition. As clarification of the response, the NRC staff notes that the applicant or licensee referencing the SDAA must meet the NRC regulatory requirements for using a specific ASME OM Code edition, or a later edition as required by 10 CFR 50.55a.

FSAR Section 3.9.3.3, "Pump and Valve Operability Assurance," states that the ASME OM Code specifies the IST requirements for components that are within the scope of the ASME OM Code. The staff considers the FSAR provision to be acceptable.

FSAR Section 3.11.2.2, "Qualification of Mechanical Equipment," specifies that for mechanical equipment located in a mild environment, acceptable environmental design is demonstrated by conformance with the design and purchase specifications for the equipment. During the audit, the NRC staff requested the applicant to describe the intent of this sentence in that the SDA application specifies that mechanical equipment will be qualified by the ASME QME-1 Standard as accepted in RG 1.100. The applicant stated that the intent of the sentence is to reflect that the approach used for mechanical equipment in a mild environment conforms to GDC 4 and applicable DSRS 3.11. The applicant stated that safety-related active mechanical equipment located in a mild environment are subject to seismic qualification as well as functional qualification requirements as described in RG 1.100, Revision 4.

During the audit, the NRC staff requested the applicant to describe how the testing planned under FSAR Section 3.9.6.4.2, "Inadvertent Actuation Block Test Frequency Alternate

Authorization," will be coordinated with the testing planned under FSAR Section 3.9.6.4.4, "Emergency Core Cooling System Valve Alternate Authorization," in the NuScale FSAR. In its audit response, NuScale describes its plans to coordinate the testing described in FSAR Section 3.9.6.4.2 for the IAB valve, and FSAR Section 3.9.6.4.4 for the ECCS valve alternate testing. The staff found the NuScale plans to provide for adequate qualification and testing of the IAB valve.

During the audit, the NRC staff requested the applicant to describe the installation of venturis in the RVVs and RRVs such that the method and frequency of testing provide reasonable assurance of their operational readiness. The applicant stated that the valve venturis are passive internal valve components with a fixed geometry. The applicant stated that installation of the venturis in the RVVs and RRVs does not result in a different methodology or frequency of testing of any preservice or inservice test as described in FSAR Section 3.9.6 or Table 3.9-18. The staff found the NuScale plans to provide adequate qualification and testing for the RVVs and RRVs with their venturis.

The applicant stated that the safety-related valves in the NuScale Power Plant are subject to environmental qualification requirements. As the scope of 10 CFR 50.49 is "important to safety" components, the NRC staff requested during the audit whether there were any non-safety-related but important to safety valves in the NuScale US460 standard design within the scope of 10 CFR 50.49. The applicant stated that the US460 standard design has valves with a safety classification of B2 (nonsafety-related, non-risk-significant) with augmented quality requirements for environmental qualification. The staff considered this response regarding augmented quality requirements to be acceptable.

The NRC staff determined that the ECCS Valve Design Demonstration Testing performed by NuScale satisfied 10 CFR 52.137(b) and 10 CFR 50.43(e) to demonstrate the safety features of the ECCS valve system described in the NuScale SDAA. A COL applicant referencing the NuScale Power Plant US460 standard design will need to address the follow-up items from the ECCS valve tests as part of the qualification testing under ASME Standard QME-1-2017 as accepted in RG 1.100. For example, QME-1 qualification testing will include verification of the assumed compressible flow parameters for the RVVs.

3.9.6.4.6.2 Containment Isolation Valves

FSAR Section 6.2.4, "Containment Isolation System," specifies that the containment boundary is formed by the CNV and CIVs and the passive containment isolation barriers that are used to prevent release through the penetrations in the CNV. Section 6.2.4 indicates that there are eight mechanical penetrations through the CNV top head, with two hydraulically operated PSCIVs in series outside of the CNV in lines connected to the RCPB or open to the atmosphere inside of the CNV. Section 6.2.4 also indicates that there are four mechanical penetrations through the CNV top head, with a single hydraulically operated SSCIV in lines outside of the CNV for piping inside of the CNV for a closed piping system and not connected to the RCPB or the atmosphere inside of the CNV. Section 6.2.4 specifies that the PSCIV design has a configuration of two valves (with separate actuators and ball-valve obturators) contained in a single body. Section 6.2.4 indicates that the PSCIVs will include a design feature to allow excess pressure caused by heatup of fluid between its two valves to be released into the CNV. The SSCIVs use a single ball-valve design. The MSIVs are specified as single SSCIVs. The FWIVs are specified as SSCIVs, but also have a feedwater isolation check valve housed in the same valve body. FSAR Figure 6.2-6b, "Feedwater Isolation Valve with Nozzle Check Valve and Actuator Assembly," shows a nozzle check valve in the same valve body with the FWIV. Section 6.2.4

specifies that hydraulic actuators with nitrogen gas cylinders are used to operate both the PSCIV and SSCIV designs. [Refer to SER Section 6.2.4]

Based on the FOAK valve design for the NuScale CIVs, the staff evaluated the CIV design to determine whether NuScale has provided sufficient information in support of the NuScale SDAA. From its review, the staff found that the design and operation of the PSCIVs and SSCIVs represent a new application for CIVs used in nuclear power plants. However, the staff determined that the design and operation of the PSCIVs and SSCIVs do not represent a significant safety question that requires design demonstration testing for the NuScale SDAA. In addition, the FSAR requires that PSCIVs and SSCIVs be qualified in accordance with ASME Standard QME-1-2017, which is endorsed in RG 1.100, Revision 4, to provide reasonable assurance of the capability of the PSCIVs and SSCIVs to perform their safety functions. Based on its review, the staff determined that NuScale has provided sufficient information in the CIV design documentation to be used in the NuScale Power Plant in support of the NuScale SDA.

Based on its review, the staff finds the description in the NuScale SDAA for the design, qualification, and testing of the CIVs to be acceptable for the NRC review of the NuScale SDAA.

3.9.6.4.6.3 Reactor Safety Valves

FSAR Section 5.2.2 specifies that each NPM is provided with overpressure protection features to protect the RCPB, including the primary side of auxiliary systems connected to the RCS, and the secondary side of the SGs. Section 5.2.2 indicates that, during normal operations and AOOs, two ASME BPV Code, Section III, safety valves provide integrated overpressure protection for the RCPB. In particular, two pilot-operated RSVs are installed above the pressurizer volume on the top of the RPV head to provide overpressure relief for the RCS directly into containment. Section 5.2.2 also specifies that the LTOP system consists of the RVVs and provides overpressure protection during low-temperature conditions. [Refer to SER Section 5.2.2

FSAR Section 5.2.2.4.1, “Reactor Safety Valves,” specifies that the RSVs are safety-related, Seismic Category I, Quality Group A, components. FSAR Section 5.2.2.6, “Applicable Codes and Classification,” indicates that the RSVs are designed in accordance with ASME BPV Code, Section III, Subarticle NB-3500, “Valve Design,” and function to satisfy the overpressure protection criteria described in ASME BPV Code, Section III, Article NB-7000, “Overpressure Protection.” FSAR Section 5.2.2.4.1 specifies that the RSVs are a pilot-operated valve design with general drawings provided in Figure 5.2-1, “Reactor Safety Valve Simplified Diagram,” and Figure 5.2-2, “Reactor Safety Valve Pilot Valve Assembly Simplified Diagram.”

From its review, the staff found that the design and operation of the RSVs to be used in the NuScale Power Plant are similar to overpressure protection valves used in current nuclear power plants. Therefore, the staff determined that the design and operation of the RSVs do not represent a significant safety question that requires design demonstration testing for the NuScale SDAA. The staff found that the FSAR indicates that the RSVs will be certified in accordance with the ASME BPV Code as incorporated by reference in the NRC regulations and qualified in accordance with ASME Standard QME-1-2017, which is endorsed by RG 1.100, Revision 4, to provide reasonable assurance of the capability of the RSVs to perform their safety functions. Based on its review, the staff determined that NuScale has provided sufficient information in the RSV design documentation to be used in the NuScale Power Plant in support of the NuScale SDAA.

The staff reviewed specific provisions provided in the NuScale SDAA for the design and qualification of the RSVs. For example, FSAR Section 5.2.2.2.2 includes COL Item 5.2-2 specifying that a COL applicant that references the NuScale Power Plant US460 standard design will provide a certified overpressure protection report in compliance with ASME BPV Code, Section III, Subarticles NB-7200, “Overpressure Protection Report,” and NC-7200, “Overpressure Protection Report,” to demonstrate that the RCPB and secondary system are designed with adequate overpressure protection features, including LTOP features. The NRC staff finds that this COL item is consistent with the NRC regulatory requirements in 10 CFR 50.55a that incorporate by reference the ASME BPV Code, Section III.

The staff notes that FSAR Section 5.2.2.4.1 specifies that the two RSVs are pilot-operated relief valves designed to maintain pressure below 110 percent of design pressure, with each RSV sized to provide 100 percent of the required relief capacity. FSAR Section 5.2.2.9, “System Reliability,” indicates that the RSVs are considered passive devices with respect to accident analyses. This SE addresses the staff review of the applicant’s assumptions in its accident analyses of the potential failure of the RSVs as part of the NRC evaluation of FSAR Chapter 15, “Transient and Accident Analyses.” With respect to its review of the NuScale IST program, the staff notes that FSAR Table 3.9-17 includes the RSVs as active valves in the NuScale Power Plant. In addition, RSVs are active valves in accordance with the ASME OM Code as required by 10 CFR 50.55a. FSAR Table 3.9-18 specifies the IST provisions for the RSVs consistent with the ASME OM Code requirements for safety valves as incorporated by reference in 10 CFR 50.55a in the NRC regulations.

Based on its review, the staff finds the provisions specified in the NuScale SDAA for the functional qualification of the RSVs to be consistent with the NRC regulatory guidance and therefore to be acceptable.

3.9.6.4.6.4 Main Steam Isolation Valves

FSAR Table 3.9-18 specifies the IST provisions for main steam isolation valves (MSIVs) and main steam isolation bypass valves (MSIBVs), and also secondary MSIVs and MSIBVs. The MSIVs and MSIBVs are discussed in FSAR Section 6.2.4. The secondary MSIVs and MSIBVs are discussed in FSAR Section 10.3.2.

Note 10 in FSAR Table 3.9-18 states that the MSIVs and MSIBVs are hydraulic operated to open, and nitrogen gas to close. These valves are located on the two main steam nozzle penetrations on the CNV head and are intended to satisfy the requirements of GDC 57. One actuator is located in a single valve body that is welded to a ASME BPV Code Class 2 piping. The valves close automatically on an MPS signal or loss of power to isolate the main stream line and preserve DHRS inventory (the MSIBV is normally closed). When the valve is deenergized, parallel hydraulic vent paths open allowing fluid to vent from the valve actuator. This allows the nitrogen gas cylinder to overcome hydraulic pressure and close the valve. The nitrogen cylinder is sealed and its pressure monitored by plant instrumentation, with alarms and indication available in the MCR. The exercise test and performance assessment test (Note 16 in FSAR Table 3.9-18) determines the state of the nitrogen cylinder (pressure, temperature), the state of the obturator (stroke time and diagnostics), and the state of each hydraulic vent path (by testing each vent path individually). These valves have a CITF that allows leakage testing (Note 15 in FSAR Table 3.9-18) locally in the direction of steam flow and DHRS isolation. These valves cannot be full-stroke or part-stroke exercised during plant operation because closing the valves interrupts steam flow resulting in possible SG pressure and level transients and may initiate a turbine or NPM trip.

Note 19 in FSAR Table 3.9-18 states that the secondary MSIVs and MSIBVs are nonsafety-related, non-risk-significant backup isolation valves to the safety-related MSIVs and MSIBVs, and are credited in the safety analysis. These valves have the same design pressure and temperature as the RCS. These valves cannot be full-stroke or part-stroke exercised during plant operation because closing the valves would interrupt steam flow resulting in possible SG pressure and level transients and may initiate a turbine or NPM trip. The FSAR accident analyses rely on the nonsafety-related secondary MSIVs for mitigating an SG tube rupture (SGTR) event.

Based on its review, the staff finds the description in the NuScale FSAR for the design, qualification, and testing of the MSIVs and MSIBVs, and secondary MSIVs and MSIBVs to be acceptable.

3.9.6.4.6.5 CVCS Flow Restricting Venturis

In its response to RAI-10157-R1 (Q9.3.4-3), the applicant provided details of flow restricting venturis that are installed in each CVCS line inside containment. The applicant stated that the venturis are not assigned a Quality Group because they "are not part of the ASME Code pressure boundary." This statement appears inconsistent with other designs, such as AP1000, where all mechanical components (whether they are pressure retaining or not) are assigned to a quality group. Staff evaluation for this area of review can be found in SER Section 9.3.4.

3.9.6.4.7 Augmented Testing Program

In addition to complying with the provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a, the NRC regulations in 10 CFR 50.55a(b)(3)(iii)(D) require that COL holders whose initial fuel loading occurs on or after the date 12 months after August 17, 2017, shall assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the RTNSS for applicable reactor designs. The NRC described its policy for new reactors with passive emergency cooling systems in several Commission papers and staff memoranda, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors (ALWR) Designs," dated April 2, 1993 (ML003708021); SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 (ML003708068); and SECY-95-132, with their applicable SRM, dated July 21, 1993; June 30, 1994; and June 28, 1995 (ML003708056, ML003708098, and ML003708019, respectively), and an NRC staff public memorandum dated July 25, 1995 (ML003708048). For example, SECY-93-087 indicates that passive reactor designs include active systems that are not safety-related to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal that serve as the first line of defense in the event of transients or plant upsets to reduce challenges to the passive systems. SECY-93-087 states that uncertainties remain concerning the performance of the unique passive features and the overall performance of core and containment heat removal because of the lack of proven operational performance history. SECY-93-087 indicates that the staff's review of passive designs requires an evaluation of not only the passive safety systems, but also the functional capability and availability of the active systems that are not safety-related to provide significant defense in depth and accident and core damage prevention capability.

FSAR Section 3.9.6.5 and Table 3.9-19 describe an augmented testing program for specific valves. In particular, FSAR Section 3.9.6.5 specifies that components not required by ASME OM Code, paragraph ISTA-1100, but with augmented requirements. FSAR Section 3.9.6.5

notes that these components provide a BDBE function, nonsafety-related backup of a safety-related function for containment isolation, and an ASME Code class / non-code class break to meet Regulatory Guide 1.26 provisions. FSAR Section 3.9.6.5 specifies that these components will be tested to the intent of the ASME OM Code and applicable addenda, as endorsed by 10 CFR 50.55a(f), or where the NRC has authorized an alternative in accordance with 10 CFR 50.55a, commensurate with the augmented requirements for those components. FSAR Section 3.9.6.5 indicates that FSAR Table 3.9-19 includes the augmented test requirements for specific valves. The staff finds this description of the augmented testing program to be consistent with the Commission policy on components that are not safety related but provide a first line of defense in the event of plant transients, and therefore, is acceptable.

3.9.6.4.8 Initial Test Programs

FSAR Section 14.2, “Initial Plant Test Program,” lists the series of preoperational and startup tests to be conducted by a COL holder for a NuScale Power Plant. Many of the initial test program (ITP) tests involve various types and categories of valves to be installed in the NuScale Power Plant.

The NRC staff has evaluated the adequacy of the ITP tests in SER Chapter 14. A COL holder for a NuScale Power Plant will be responsible for addressing the performance of the applicable valves during the ITP tests.

3.9.6.5 Combined License Information Items

Table 3.9.6-1, “NuScale COL Information Items for Section 3.9.6,” in this SER section lists COL information items and their descriptions related to FSAR Section 3.9.6. As discussed below, the staff has determined that these COL information items provide appropriate requirements for a COL applicant to fully describe its PST and IST programs in the COL application.

Table 3.9.6-1: NuScale COL Information Items for Section 3.9.6

Item No.	Description	FSAR Section
COL Item 3.9-8	An applicant that references the NuScale Power Plant US460 standard design will establish Preservice and Inservice Testing Programs. These programs are to be consistent with the requirements in the latest edition and addenda of the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code incorporated by reference in 10 CFR 50.55a.	3.9.6.3
COL Item 3.9-9	An applicant that references the NuScale Power Plant US460 standard design will develop specific test procedures to allow detection and monitoring of power-operated valve assembly performance sufficient to satisfy periodic verification [of] design basis capability requirements.	3.9.6.3.2
COL Item 3.9-10	An applicant that references the NuScale Power Plant US460 standard design will develop specific test procedures to allow detection and monitoring of emergency core cooling system valve assembly performance sufficient to satisfy periodic verification of design-basis capability requirements.	3.9.6.3.2

These COL items are acceptable in specifying that a COL applicant that references the NuScale Power Plant US460 standard design should develop a PST and IST program (1) to satisfy 10 CFR 50.55a(f) requirements specific to a COL applicant, (2) to incorporate the provisions of the applicable ASME OM Code edition, (3) to develop specific test procedures for POV periodic verification, and (4) to develop specific test procedures for ECCS valves for periodic verification of their design-basis capability requirements. The NRC staff finds that these COL items provide assurance that a COL applicant will develop PST and IST programs that will satisfy the NRC regulatory requirements in accordance with the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The staff also finds that these COL items will provide assurance that a COL applicant will develop test procedures to satisfy the NRC regulatory requirements in 10 CFR 50.55a for periodic verification of the design-basis capability of POVs and the ECCS valve system to perform their safety functions. Therefore, the staff finds that these COL items are acceptable for a COL applicant for a NuScale Power Plant to address as part of its application.

3.9.6.6 Conclusion

Based on the SDAA description and COL items, the NRC staff concludes that the applicant has demonstrated the functional design, qualification, and IST provisions for the pumps, valves, and dynamic restraints (as applicable) for the NuScale Power Plant consistent with GDC 1, 2, 4, 14, 15, 37, 40, 43, 46, and 54 in Appendix A to 10 CFR Part 50; Appendix B to 10 CFR Part 50; 10 CFR 50.55a; and 10 CFR 52.137 of the NRC regulations. The staff concludes that the NuScale SDAA provides assurance that the functional design, qualification, and IST provisions for valves in the NuScale Power Plant referenced in the SDAA can be performed and that the NuScale SSCs provide access to permit the performance of testing pursuant to 10 CFR 50.55a. The staff notes that the NRC regulations in 10 CFR 50.55a(f)(4)(i) require that inservice tests to verify the operational readiness of pumps and valves with a function required for safety conducted during the initial IST Program interval must comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(b) from the time period specified in 10 CFR 50.55a(f)(4)(i) before the date scheduled for initial fuel loading under a COL issued under 10 CFR Part 52 or the optional ASME OM Code Cases listed in RG 1.192, subject to the limitations and modifications listed in 10 CFR 50.55a.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

3.10.1 Introduction

The purpose of this section is to review the criteria, procedures, and methods the applicant will use for seismic and dynamic qualification to ensure the functionality of mechanical and electrical equipment (including I&C).

3.10.2 Summary of Application

FSAR: FSAR Section 3.10 addresses the acceptance criteria, codes and standards, procedures, and methods applied to the seismic and dynamic qualification of mechanical and electrical equipment (including instrumentation) to provide reasonable assurance that they will withstand the effects of postulated events and accidents and still be capable of performing their functions under the full range of normal, transient, seismic, and accident loadings. The equipment to be qualified includes that necessary for safe shutdown, emergency core cooling, containment heat removal, containment isolation, or for mitigating the consequences of accidents or preventing a significant release of radioactive material to the environment. Also included is equipment in the reactor protection system, the engineered safety features (ESFs), and other support systems that directly or indirectly support the performance of one or more of the above safety functions.

The qualification of electrical equipment is performed according to the Institute of Electrical and Electronics Engineers (IEEE) Std. 344-2013. The qualification of active mechanical equipment is conducted according to ASME QME-1-2017. The qualification includes analysis, testing, or a combination of analysis and testing. The methods for analysis and testing are also described. Analyses typically include static coefficient analysis and dynamic analysis. The methods of analysis and testing of supports of equipment and instrumentation are also discussed. Finally, FSAR describes the documentation of the equipment qualification (EQ) records.

ITAAC: SDAA, Part 8, Table 2.4-2, lists the relevant ITAAC for Section 3.10. SER Section 14.3 discusses NuScale ITAAC.

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

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

3.10.3 Regulatory Basis

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

- GDC 1 and GDC 30, “Quality of Reactor Coolant Pressure Boundary,” as related to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed

- GDC 2 and Appendix S to 10 CFR Part 50, as related to qualifying equipment to withstand the effects of natural phenomena, such as earthquakes

- GDC 4, as related to qualifying equipment to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC 14, as related to qualifying equipment associated with the reactor coolant boundary to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- 10 CFR Part 50, Appendix B, as related to qualifying equipment using the QA criteria provided

The guidance in SRP Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” lists the acceptance criteria for meeting the above requirements, as well as review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria to confirm that the above requirements have been adequately addressed:

- RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Revision 1

- RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Revision 3

- RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” Revision 4

3.10.4 Technical Evaluation

The staff performed its review of FSAR Section 3.10, related to the seismic and dynamic qualification of mechanical and electrical equipment, in accordance with the criteria and procedures delineated in SRP Section 3.10, Revision 4, issued March 2016. The SRP contains six acceptance criteria. The following subsections discuss the staff’s review of the consistency of FSAR with these criteria.

The seismic qualification methodology, described in FSAR Section 3.10, will be used for both mechanical and electrical equipment.

3.10.4.1 Qualification of Electrical and Mechanical Equipment and Supports

This first set of SRP acceptance criteria is divided into three areas: (1) the qualification of equipment functionality, (2) the design adequacy of supports, and (3) the verification of seismic and dynamic qualification. Each of these areas is evaluated below.

A. Qualification of Equipment Functionality

The qualification of equipment functionality includes 14 criteria (i through xiv), which are discussed below.

i. FSAR Section 3.10.2.1, “Qualification by Testing,” states, “Seismic qualification of mechanical and electrical equipment by testing is performed in accordance with the requirements of IEEE 344-2013.“ It further states, “The testing also simulates the effects of aging, such as the fatigue effects of five OBEs plus the loadings associated with normal operation for the design life of the equipment prior to simulating the effects of an SSE, which is equivalent to two SSEs, with 10 stress cycles each. Single-frequency and multi-frequency tests are used for seismic qualification.” This statement is consistent with the criterion. Moreover, the applicant will use testing or analysis to qualify equipment as stated in FSAR Section 3.10.2.2. The staff finds that this is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.i.

ii. SRP Section 3.10, Acceptance Criterion II.1.A.ii, states, “Equipment should be tested in the operational condition. Functionality should be verified during and/or after the testing, as applicable to the loadings simulating those of plant normal operation, such as thermal and flow-induced loading, if any, should be concurrently superimposed upon the seismic and other pertinent dynamic loading to the extent practicable.” The applicant is using ASME QME-1-2017 as described in RG 1.100 for the seismic qualification of active mechanical equipment. Following ASME QME-1-2017 as described in RG 1.100 is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.ii.

iii. FSAR Section 3.10.2.1, states, “The seismic testing consists of subjecting the equipment to vibratory motion that simulates the vibratory motion postulated to occur at the equipment mounting location. The testing conservatively considers the multi-dimensional effects of the postulated earthquake.” The staff finds the information in FSAR Section 3.10.2.1 acceptable because it is consistent with

SRP Section 3.10, Acceptance Criterion II.1.A.iii. Simulating the vibratory motion postulated to occur at the equipment mounting location characterizes the required seismic and dynamic input motions.

iv. FSAR Section 3.10.1.2, “Performance Requirements for Seismic Qualification,” states, “The EQRF identifies and evaluates the test response spectrum (TRS) and required response spectrum (RRS) for the seismic qualification are also identified in the EQRF. The RRS is bounded by the TRS to demonstrate the conservative qualification of equipment.” The staff finds FSAR Section 3.10.1.2 consistent with SRP Section 3.10, Acceptance Criterion II.1.A.iv. In bounding the RRS, the TRS will resemble and envelop the required spectrum over the critical frequency range.

v. FSAR Section 3.10.2.1, states, “the purpose of multi-frequency testing is to provide a broadband test motion that can produce a simultaneous response from multiple modes of a multi-degree-of-freedom system, which can malfunction as a result of modal interactions. It is preferable to perform multi-frequency testing rather than single-frequency testing because of the usually broad frequency content of the seismic and dynamic load excitation. However, single-frequency testing, such as sine beats, may be used in the following situations:

- When seismic ground motion is filtered due to a single predominant structural mode.
- When it can be shown that the anticipated response of the equipment is sufficiently represented by a single mode.
- When the input has enough duration and intensity to cause the excitation of the applicable modes to the required magnitude, causing the TRS to bound the corresponding spectra.
- When the resultant floor motion consists of a single predominant frequency.”

FSAR Section 3.10.2.1 is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.v, and is therefore acceptable.

vi. FSAR Section 3.10.2.1, states, “the test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.”

FSAR Section 3.10.2.1, satisfactorily addresses the testing requirements for equipment because it is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.vi. The orientation and application of the input motions are as described in the SRP.

vii. FSAR Section 3.10.2.1, states, "the equipment mounting in the test setup simulates the equipment mounting in service and does not cause non-representative dynamic coupling of the equipment to its mounting fixture." The staff finds FSAR Section 3.10.2.1 acceptable because it is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.vii.

viii. FSAR Section 3.10.3, "Methods and Procedures for Qualifying Supports of Mechanical and Electrical Equipment and Instrumentation," states, "the mountings are designed to avoid extraneous dynamic coupling. The equipment mounting considered in the analysis or testing is identified in the EQRF."

The staff finds FSAR Section 3.10.3 acceptable because it is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.viii.

ix. FSAR Section 3.10.2.1, states, "the loads include forces imposed by piping onto the equipment." The staff finds FSAR Section 3.10.2.1 acceptable because it is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.ix. When testing is performed in accordance with IEEE Std. 344-2013, stresses in valve bodies and pump casings are limited to the particular material's elastic limit when the pump or valve is subject to the combination of normal operating loads, SSE, and other applicable dynamic loads.

x. SRP Section 3.10, Acceptance Criterion II.1.A.x, states, "If the dynamic testing of a pump or valve assembly proves to be impracticable, static testing of the assembly is acceptable provided that the end loadings are conservatively applied and are equal to or greater than postulated event loads, all dynamic amplification effects are accounted for, the component is in the operating mode during and after the application of loads, and an adequate analysis is made to show the validity of the static application of loads." Because the applicant does not use this option, satisfying SRP Section 3.10, Acceptance Criterion II.1.A.x, is not necessary.

xi. The applicant is not using in situ application of vibratory devices to simulate the seismic and dynamic vibratory motions on a complex active device. Therefore, the applicant does not need to show that an acceptable test can be completed, as specified in SRP Section 3.10, Acceptance Criterion II.1.A.xi.

xii. SRP Section 3.10, Acceptance Criterion II.1.A.xii, states, "The test program may be based on selective testing of a representative number of components according to type, load level, size, and the like on a prototype basis." FSAR Section 3.10.2.1, states, "Seismic qualification of mechanical and electrical equipment by testing is performed in accordance with the requirements of IEEE 344-2013. The guidance of IEEE Std. 344-20013 allows this option." The staff finds that following IEEE Std. 344-2013, as stated in FSAR Section 3.10.2.1, satisfies SRP Section 3.10, Acceptance Criterion II.1.A.xii.

xiii. SRP Section 3.10, Acceptance Criterion II.1.A.xiii, states, "Selection of damping values for equipment to be qualified by analysis should be made in accordance with RG 1.61 and ANSI/IEEE Std. 344-1987. Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism." The staff reviewed the damping values and finds that

Acceptance Criterion II.1.A.xiii is satisfied. The damping values used for the analysis of the Seismic Category I and II SSCs are based on RG 1.61, Revision 1. The staff finds this acceptable for use in any subsequent dynamic analysis for the NuScale design.

xiv. FSAR Section 3.10.2.3 states the following:

> When testing or analysis alone are not practical to sufficiently qualify equipment, combined testing and analysis methods are used. The requirements of IEEE 344-2013 are used to perform equipment qualification by combined testing and analysis. Operability and structural integrity of components are demonstrated by calculating component deflections and stresses under various loads. These results are then compared to the allowable levels, per the applicable codes.

The methods and requirements of ASME QME-1-2017 as described in RG 1.100 are also used for the seismic qualification of active mechanical equipment, as stated in FSAR Section 3.10.1.1. Subsection QR-7312, “Dynamic Loading,” of ASME QME-1-2017 states that qualification of active mechanical equipment for dynamic loadings such as, but not limited to, vibration and seismic loadings, should consider the requirements and general approaches outlined in Nonmandatory Appendix QR-A and IEEE Std. 344. The staff finds the use of the analytical approach consistent with SRP Section 3.10, because ASME QME-1-2017 and IEEE Std. 344 will be used, and they contain analysis approaches consistent with SRP Section 3.10, Acceptance Criterion II.1.A.xiv.

B. Design Adequacy of Supports

i. SRP Section 3.10 indicates that analyses or tests should be performed for all supports of mechanical and electrical equipment to ensure their structural capability. FSAR Section 3.10.3 indicates that NuScale will use testing or analysis to qualify Seismic Category I mechanical and electrical equipment to demonstrate structural integrity. This is consistent with SRP Section 3.10. Testing or analysis is used to qualify Seismic Category I mechanical and electrical equipment to demonstrate its structural integrity, including the structural integrity of its anchorage, and its ability to withstand seismic excitation corresponding to the RRS. This is consistent with SRP Section 3.10, Acceptance Criterion II.1.B.i.

ii. SRP Section 3.10 indicates that the analytical results should include the required input motions to the mounted equipment as obtained and characterized in the manner stated in Acceptance Criterion II.1.A.iii, and the combined stresses of the support structures should be in accordance with the criteria specified in SRP Section 3.9.3. As described earlier in the section, the staff concluded that the methodology provided by the applicant satisfies Acceptance Criterion II.1.A.iii. The staff review of the application of the ASME BPV Code Class 1, 2, and 3 component load combinations criterion is documented in Section 3.9.3 of this SER and found to be acceptable. Therefore, the staff concluded SRP Section 3.10, Acceptance Criterion II.1.B.ii, can be satisfied.

iii. SRP Section 3.10 states, "Supports should be tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response in the test at the equipment mounting location should be monitored and characterized in the manner stated in subsection II.1.A.iii above. In such a case, equipment should be tested separately for functionality, and the actual input motion to the equipment in this test should be more conservative in amplitude and frequency than the monitored response from the support test." FSAR Section 3.10.3 states, "Installed equipment (or equivalent equipment with the same inertial mass effects and dynamic coupling to the equipment mounting) is used for qualification of supports for electrical equipment and instrumentation, which includes electrical cabinets, control consoles, electrical panels, and instrument racks."

iv. The stresses and deflections are compared to the applicable codes and regulations. When testing is not practical, equipment may be analyzed to confirm their structural integrity. The analysis accounts for the complexity of the supports and accurately represent the response to seismic excitation and vibratory motions." The staff finds that SRP Section 3.10, Acceptance Criterion II.1.B.iii, is satisfied, since the applicant uses a dummy weight consistent with this section of the SRP.

v. SRP Section 3.10 states that Acceptance Criteria II.1.A.iii through II.1.A.xiii apply when tests are conducted on the equipment supports. FSAR Section 3.10.3 states that testing or analysis is used to qualify Seismic Category I mechanical and electrical equipment to demonstrate its structural integrity, including the structural integrity of its anchorage, and its ability to withstand seismic excitation corresponding to the RRS for the equipment's mounting configuration. The qualification of supports for electrical equipment and instrumentation, which include electrical cabinets, control consoles, electrical panels, and instrument racks, uses the installed equipment or a dummy weight to simulate the inertial effects and dynamic coupling to the support. The analysis accounts for the complexity of the supports and accurately represent the response to seismic excitation and vibratory motions. The stresses and deflections are compared to the applicable codes and regulations. The staff finds that the applicant has satisfied SRP Section 3.10, Acceptance Criterion II.1.B.iv, because the TRS closely resembles and envelops the RRS over the critical frequency range.

C. Verification of Seismic and Dynamic Qualification

SRP Section 3.10, Acceptance Criterion II.1C, states, "The seismic and dynamic qualification testing performed in accordance with ANSI/IEEE Std. 344-1987, as endorsed by RG 1.100, Revision 2, as part of an overall qualification program should be performed in the sequence indicated in Section 6 of IEEE Std. 323-1974 (endorsed with exceptions by RG 1.89)." FSAR Section 3.10.1.1 notes that the requirements of IEEE Std. 344-2013 endorsed by RG 1.100, Revision 4, will be implemented. The use of these updated standards is required by RG 1.100, Revision 4. Therefore, the applicant is permitted by the RG to overrule the revision level of SRP Section 3.10, Acceptance Criterion II.1.C.

Based on the above evaluation of SRP Section 3.10, Criteria i through xiv for the qualification of equipment functionality, the design adequacy of supports, and the verification of seismic and dynamic qualification, the staff finds that the SDAA is either consistent with the criteria or the criteria do not apply to the NuScale method for qualification of the equipment. Thus, since the qualification is consistent with the SRP criteria, which the staff considers an acceptable means to meet GDC 1, 2, 4, 14, and 30, the staff finds that the SDAA meets these GDC for the seismic and dynamic qualification of mechanical and electrical equipment.

3.10.4.2 Qualification of Regulatory Guide 1.97 Instrumentation (SRP Section 3.10, Acceptance Criterion 2)

As stated in FSAR Section 3.10.1.2, the qualification of instrumentation is addressed in FSAR Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." FSAR Section 3.11 and Appendix 3C, "Methodology for Environmental Qualification of Electrical and Mechanical Equipment," describe the environmental conditions of the mechanical and electrical equipment, including the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including seismic events. This includes instrumentation covered in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." However, seismic qualification of RG 1.97 instrumentation meets the criteria of SRP Section 3.10 since it is still categorized as mechanical and electrical equipment. FSAR Section 3.10.2 states, "Seismic Category I instrumentation and electrical equipment are qualified by type testing or by a combination of testing and analysis." Therefore, the qualification of RG 1.97 instrumentation is consistent with the criterion in SRP Section 3.10, Acceptance Criterion II.2.

3.10.4.3 Qualification of Equipment Using an Experience-Based Approach (SRP Section 3.10, Acceptance Criterion 3)

The applicant did not propose to qualify equipment using an experience-based approach. This is acceptable, as the experience-based approach is not a requirement but an option.

3.10.4.4 Equipment Qualification Records (SRP Section 3.10, Acceptance Criterion 4)

The results of seismic qualification testing and analysis, according to the criteria in FSAR Sections 3.10.1, 3.10.2, and 3.10.3, are included in the corresponding EQRFs. NuScale stated that the EQRFs are (1) created and maintained during the equipment selection and procurement phase for the equipment requiring qualification, (2) contain a detailed description of the equipment and its support structures, qualification methodology, test and analysis results, and (3) are updated and maintained current and auditable. The experience database containing plant EQRF data is maintained for the life of the plant. The EQRFs as required by the SDA include the following information: detailed equipment information to include location in building, supplier or vendor, make and model, and serial number. identification of the RCPB components.

- the type of support used to mount the equipment.
- the weight, dimensions, and physical characteristics of the equipment.
- the function of the equipment.
- the loads and load intensities for which the equipment is qualified.

- for equipment qualified by testing, the test procedures and methods, a description of the test, parameters of the test, and results of the test.
- for equipment qualified by analysis, the analytical methods, assumptions, and results.
- the equipment's natural frequencies.
- the methods used to qualify equipment for vibration-induced fatigue cycle effects, if applicable.
- identification of whether equipment is installed.
- the associated RRS or time history and the applicable damping for normal loadings and other dynamic loadings in conjunction with the specified seismic load.

The staff finds that the development of EQRF files, as described, meets the requirements of SRP Section 3.10, Acceptance Criterion II.4. The criterion states the following:

> GDC 1 and 10 CFR Part 50, Appendix B, Criteria XVII, "Quality Assurance Records," establish requirements for records concerning the qualification of equipment. To satisfy these requirements, complete and auditable records must be available, and the applicant must maintain them for the life of the plant. These files should describe the qualification method used for all equipment in sufficient detail to document the degree of compliance with the criteria of this SRP section. These records should be updated and kept current as equipment is replaced, further tested, or otherwise further qualified.

The EQRF files, as described above, satisfy these requirements because the files (1) contain a detailed description of the equipment and its support structures, qualification methodology, test and analysis results, (2) are updated and modified as new tests and analyses are performed, and (3) are maintained for the life of the plant.

3.10.4.5 Qualification of Valves in the Reactor Coolant Pressure Boundary (SRP Section 3.10, Acceptance Criterion 5)

SRP Section 3.10, Acceptance Criterion II.5, specifies that the qualification program for valves that are part of the RCPB should include testing or testing and analyses demonstrating that these valves will not experience leakage, or an increase in leakage, as a result of any loading or combination of loadings for which the valves must be qualified. Section 3.9.6 of this SER documents the review of the functional qualification of valves in the RCPB; however, the seismic testing of the valve actuator should be in accordance with SRP Section 3.10, Acceptance Criterion II.1.C, which states the following:

> The seismic and dynamic qualification testing performed in accordance with ANSI/IEEE Std 344-1987, as endorsed by RG 1.100, Revision 2, as part of an overall qualification program should be performed in the sequence indicated in Section 6 of IEEE Std 323-1974 (endorsed with exceptions by RG 1.89). However, RG1.100. Revision 4, supersedes ANSI/IEEE Std 344-1987, with ANSI/IEEE Std 344-2013 and IEEE Std 323-1974 with IEEE Std 323-2003, as specified by IEEE Std 344-2013, "For undated references, the latest edition of the referenced document (including any amendments or corrigenda) applies."

The applicant, in FSAR Section 3.10, stated the following:

> Electrical and mechanical equipment including instrumentation (with exception of piping) and their associated supports classified as Seismic Category I, are demonstrated through qualification to withstand the full range of normal and accident loadings. The equipment to be seismically and dynamically qualified includes the following: electrical equipment, including instrumentation and some post-accident monitoring equipment; [and] active, safety-related mechanical equipment, such as control rod drive mechanism and some valves, that perform a mechanical motion to accomplish their safety function and other nonactive mechanical components, that maintain structural integrity.

The EQRFs address the requirements for active valves and dampers. The structural integrity and operability of active valves and dampers are qualified by a combination of analyses and tests. ASME QME-1-2017 is used with the exceptions noted in RG 1.100, Revision 4, for the qualification of active mechanical equipment.

3.10.4.6 Equipment Qualification Program Implementation Documentation (SRP 3.10, Acceptance Criterion 6)

NuScale stated that an EQRF is developed for each piece of electrical equipment and instrumentation classified as Seismic Category I. FSAR Section 3.11 and Appendix 3C describe the environmental conditions of the mechanical and electrical equipment, including the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including seismic events. The equipment specifications define the performance requirements for the electrical equipment and instrumentation. The EQRF also identifies the test response spectrum (TRS) and required response spectrum (RRS) for the seismic qualification. The RRS is bounded by the TRS to demonstrate the conservative qualification of equipment.

For seismic Category I active mechanical equipment, the performance requirements are defined in the corresponding equipment requirements specification. EQRFs address the requirements for active valves and dampers. Nonactive Seismic Category I mechanical equipment has a single performance requirement—to maintain structural integrity. Section 3.9 of this report provides additional staff review of information on the structural integrity of pressure-retaining components, their supports, and reactor core support structures.

The applicant satisfied Criteria II.6.A through II.6.C of SRP Section 3.10 as described below:

A. SDAA Part 2 meets the criteria of SRP Section 3.10, Acceptance Criterion II.6, because it contains a description of the qualification testing and analysis, does not use earthquake experience data in the qualification process, and presents information on the administrative control of the qualification.

B. SDAA Part 2 contains the following:

    i. a list of all systems required to perform the functions defined in the second paragraph of Subsection I of SRP Section 3.10

    ii. no requirement for in-plant testing in Section 3.10

C. EQRFs contain the following:

i. the list of systems required to perform the functions defined in the second paragraph of Subsection I of SRP Section 3.10

ii. the list of equipment, and its supports, associated with each system and any other equipment required in accordance with the second paragraph of Subsection I of SRP Section 3.10

iii. the summary data sheets for each piece of equipment (i.e., each component) listed

iv. a detailed description of the experience database similar to SRP Section 3.10, Acceptance Criterion II.6.A.ii, for in-scope equipment not covered in the SDAA

Based on the staff's review of the qualification standards, performance requirements, and procedures for equipment seismic qualification as described above, the staff concludes that SDAA Part 2 is consistent with the guidelines of SRP Section 3.10 for the documentation of the equipment qualification program implementation and is, therefore, acceptable.

3.10.5 Combined License Information Items

There are no COL Items for this area of review.

3.10.6 Conclusion

The staff concludes that the criteria, procedures, and methods the applicant will use for seismic and dynamic qualification to ensure the functionality of mechanical and electrical equipment (including I&C) will meet the guidance in SRP Section 3.10, thereby meeting the regulations of 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 14, and 30, and 10 CFR Part 50, Appendix B and Appendix S, with respect to seismic and dynamic qualification of components.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Introduction

Mechanical, electrical, and I&C equipment associated with systems that are essential for emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, or equipment otherwise essential in preventing significant release of radioactive material to the environment is reviewed to determine whether they are required to be environmentally qualified to meet their intended design function related to safety.

The environmental qualification (EQ) of mechanical and electrical equipment important to safety is required by 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4, to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

FSAR Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," provides the methodology for EQ of equipment and identifies the equipment that is within the scope of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." Included in FSAR Section 3.11 is a description of the approach used by the applicant to environmentally qualify electrical and mechanical equipment.

The objectives of the staff's review are to confirm that the applicant meets the requirements in 10 CFR 52.137(a)(13); to review the applicant's EQ program for compliance with 10 CFR 50.49 and to check that the set of equipment to be environmentally qualified includes, as appropriate, safety-related equipment, equipment that is not safety related whose failure under postulated environmental conditions could prevent satisfactory performance of specified safety functions, and instrumentation to monitor parameters specified in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 5, issued May 2019 (ML18136A762).

For mechanical equipment, the staff's review evaluates whether the applicant's EQ program incorporates provisions to demonstrate that nonmetallic parts of active mechanical components are designed and qualified to be compatible with the postulated environmental conditions, including those associated with a LOCA.

3.11.2 Summary of Application

FSAR: FSAR Section 3.11 is summarized below:

The applicant stated that the approach to EQ of electrical and mechanical equipment meets the applicable requirements of 10 CFR Part 50, Appendices A and B, and 10 CFR 50.49. Specifically, with regard to 10 CFR Part 50, Appendices A and B, the applicant stated that its EQ program meets the requirements of GDC 1, 2, 4, and 23 in Appendix A, and Criteria III, XI, and XVII in Appendix B. The applicant defined the scope of equipment for which EQ is required to include equipment essential for emergency reactor shutdown, core cooling, containment isolation, containment and reactor heat removal, and any equipment necessary to prevent a significant radioactive release to the environment. Also, FSAR Appendix 3C, "Methodology for Environmental Qualification of Electrical and Mechanical Equipment," describes the methodology used by the applicant to environmentally qualify electrical and mechanical equipment. FSAR Table 3.11-1, "List of Environmentally Qualified Equipment Located in Harsh Environments," provides the list of equipment located in a harsh environment to be environmentally qualified.

In FSAR Table 3C-1, "Environmental Qualification Zones," the applicant identified areas of the plant that could be subjected to a harsh environment following an accident. Further, FSAR Table 3.11-1 describes plant equipment and the area where the equipment is located and whether that area could be subjected to a harsh environment.

FSAR Section 3.11 and Appendix 3C describe the NuScale process for the EQ of nonmetallic parts of mechanical equipment (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). Nonmetallic materials are designed to meet the applicable environmental and service conditions and are qualified in accordance with Appendix QR-B, "Guide for Qualification of Nonmetallic Parts," of ASME Standard QME-1-2017. FSAR Section 3.11 and Appendix 3C do not address the functional or seismic qualification of mechanical equipment that may be considered part of "equipment qualification"; FSAR Section 3.9.6 and Section 3.10, respectively, address these topics.

ITAAC: COL Item 14.3-1 and COL Item 14.3-2 provide the requirements for an applicant to develop methodology for the ITAAC specific for the plant's structures, systems, and components within their scope. SER Section 14.3 discusses the relevant ITAAC.

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

Technical Reports: There are no Technical Reports associated with this area of review.

Topical Reports:

- TR-0915-17565-NP-A, Revision 4, “Accident Source Term Methodology Topical Report” (ML20057G132) dated February 2020

3.11.3 Regulatory Basis

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

- 10 CFR 50.49 requires that the applicant establish a program for qualifying electrical equipment important to safety located in a harsh environment.

- GDC 1 requires, in part, that components important to safety be designed, fabricated, erected, and tested to quality standards, commensurate with the importance of the safety function to be performed.

- GDC 2 requires, in part, that components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function.

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

- GDC 23, “Protection System Failure Modes,” requires that protection systems be designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criterion III, “Design Control,” requires that measures be established to ensure that applicable regulatory requirements and the associated design bases are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to ensure that appropriate quality standards are included in design documents and that deviations from established standards are controlled. A process shall also be established to determine the suitability of equipment that is essential to safety-related functions and to identify, control, and coordinate design interfaces between participating design organizations. Where a test program is used to verify the adequacy of a specific design feature, it shall include suitable qualification testing of a prototype unit under the most adverse design conditions.

- 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires a test control plan to be established to ensure that all tests needed to demonstrate a component’s capability to perform satisfactorily in service be identified and performed in accordance with written procedures that incorporate the requirements and acceptance limits contained in applicable design documents.

- 10 CFR Part 50, Appendix B, Criterion XVII, “Quality Assurance Records," requires that sufficient records be maintained to furnish evidence of activities affecting quality.

- 10 CFR 52.137(a)(13) requires an application for a standard design approval to include "[t]he list of electric equipment important to safety that is required by 10 CFR 50.49(d)." The NRC understands that the SDA applicant may not be able to establish qualification files for all applicable components.

The guidance in DSRS Section 3.11 lists the acceptance criteria adequate to meet the above requirements for the EQ of nonmetallic parts of mechanical equipment, as well as review interfaces with other DSRS/SRP sections. In addition, the following guidance documents provide acceptance criteria confirming that the above requirements have been adequately addressed:

- RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," issued June 1984 (ML003740271), provides the principal guidance for implementing the requirements and criteria of 10 CFR 50.49 for EQ of electrical equipment that is important to safety and located in a harsh environment.

- NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," issued July 1981 (ML031480402), Category I guidance may be used if relevant guidance is not provided in RG 1.89.

- RG 1.63, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Plants," Revision 3, issued February 1987 (ML003740219), endorses IEEE Std. 317, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

- RG 1.73, Revision 1, "Qualification Tests for Safety-Related Actuators in Nuclear Power Plants," issued October 2013 (ML13210A463), endorses IEEE Std. 382, "IEEE Trial Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations."

- RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 5, issued April 2019 (ML18136A762) provides guidance acceptable to the staff for the EQ of the post-accident monitoring (PAM) equipment described in Subsection I, Item 1(F), as well as instruments and controls for the equipment described in Subsection I, Items 1(a) to 1 (e), of DSRS Section 3.11.

- RG 1.100, Revision 4, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," issued May 2020 (ML19312C677), endorses, with exceptions and clarification, ASME Standard QME-1-2017 for the qualification of nonmetallic parts of active mechanical equipment.

- RG 1.156, Revision 1, "Qualification of Connection Assemblies for Nuclear Power Plants," issued July 2011 (ML111730464), endorses IEEE Std. 572, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations."

- RG 1.158, Revision 1, "Qualification of Safety-Related Vented Lead-Acid Storage Batteries for Nuclear Power Plants," issued March 2018 (ML17256A104), endorses IEEE Std. 535, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for

Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the EQ of Class 1E lead storage batteries and should be used in conjunction with NUREG-0588 and RG 1.89, as appropriate, for evaluating the EQ of lead storage batteries.

- RG 1.180, Revision 2, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," issued December 2019 (ML19175A044), provides guidance acceptable to the staff for determining electromagnetic compatibility for I&C equipment during service. These criteria, as supplemented by those in RG 1.89, should be used to evaluate the environmental design and qualification of safety-related I&C equipment. New digital systems and new advanced analog systems may require susceptibility testing for electromagnetic interference/radiofrequency interference and power surges, if the environments are significant to the equipment being qualified.

- RG 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000 (ML003716792), provides guidance acceptable to the staff for determining the radiation dose and dose rate for equipment during postulated accident conditions. These criteria, as supplemented by those of RG 1.89, should be used to evaluate the accident source term used in the environmental design and qualification of equipment important to safety.

- RG 1.211, Revision 0, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants," issued April 2009 (ML082530205), endorses IEEE Std. 383, "Standard for Type Test of Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations."

- Appendix QR-B of ASME QME-1-2017 provides guidance for the EQ of nonmetallic parts of active mechanical equipment.

3.11.4 Technical Evaluation

3.11.4.1 Environmental Qualification of Electrical and I&C Equipment

The staff reviewed FSAR Section 3.11, which describes the applicant's approach to complying with 10 CFR 50.49 for the EQ of equipment located in a harsh environment and identifies equipment that is within the scope of 10 CFR 50.49. The staff evaluated whether the information presented in FSAR Section 3.11 is sufficient to support the conclusion that all items of equipment that are important to safety are capable of performing their design safety functions under (1) normal environmental conditions (e.g., startup, operation, refueling, and shutdown), (2) AOOs (e.g., plant trip and testing), and (3) design-basis accidents (DBAs) (e.g., LOCA and HELB) and postaccident environmental conditions.

The specific equipment within the scope of EQ requirements is mechanical, electrical, and I&C, including digital I&C equipment associated with systems that are (1) essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or (2) otherwise essential in preventing significant release of radioactive material to the environment. The EQ equipment includes the following:

- equipment that initiates the above functions automatically

- equipment the failure of which can prevent the satisfactory accomplishment of one or more of the above safety functions

- electrical equipment important to safety covered by 10 CFR 50.49(b)(1)

- certain PAM equipment

3.11.4.1.1 Compliance with 10 CFR 50.49

FSAR Appendix 3C describes the methodology used to develop the EQ program. The applicant identified all equipment in the scope of 10 CFR 50.49 in FSAR Table 3.11-1, to establish the EQ list for electrical and I&C equipment, according to provisions in 10 CFR 50.49(j). The equipment included in this table is based on the guidelines provided according to provisions in 10 CFR 50.49(b)(1), 10 CFR 50.49(b)(2), and 10 CFR 50.49(b)(3):

- 10 CFR 50.49(b)(1)—safety-related electrical equipment that is relied on to remain functional during and following design-basis events (DBEs) to ensure that certain functions are accomplished

- 10 CFR 50.49(b)(2)—electrical equipment that is not safety-related, whose failure under the postulated environmental conditions could prevent satisfactory accomplishment of the safety functions of the safety-related equipment

- 10 CFR 50.49(b)(3)—certain PAM equipment as described in RG 1.97, Revision 5

The applicant explained in FSAR Section 3.11.1.1, "Equipment Identification," that equipment important to safety is classified in three categories: (1) equipment that is relied on to detect and mitigate a design-basis event (DBE), (2) equipment with credited function relied upon for its ability to achieve or maintain a safe shutdown condition for a DBE that produces a harsh environment, and (3) certain PAM equipment subject to EQ, as required by 10 CFR 50.49 (b)(3).

The staff reviewed FSAR Table 3.11-1, to verify that electrical supporting safety systems that are not safety related but are required to support a safety function were appropriately categorized under the EQ program. For example, to meet the requirements of GDC 50, the Electric Penetration Assemblies (EPAs) must be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. Since they are required to function during and after a LOCA DBE to maintain containment integrity, EPAs must be environmentally qualified.

FSAR Table 3C-1, "Environmental Qualification Zones," provides the equipment location zones of the NuScale US460 SDAA and the zone classification of the environmental conditions whether is harsh or mild. FSAR Table 3C-2, "Designated Harsh Environment Areas," provides the basis for the designation of harsh environments.

The service condition environments are divided into two categories: (1) harsh environment and (2) mild environment. FSAR Section 3.11.1.2, "Definition of Environmental Conditions," defines harsh environments as any significant change from normal that has the potential to result in environmental or radiation-induced common-cause failure mechanisms. A harsh environment has environmental conditions that exist during and after a DBE that can result in severe or elevated effects of pressure, temperature, humidity, radiation, flooding, or chemistry, including pH control. FSAR Section 3.11.1.2 defines a mild environment as plant areas where the environment at no time would be significantly more severe than the environment that would occur during normal plant operation, including AOOs. The staff finds that the definition of “mild environment” is consistent with the definition in 10 CFR 50.49(c), which states that “[a] mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.” The staff also finds the definition of harsh environment is acceptable because it is consistent with IEEE Std. 323-1974 as endorsed by RG 1.89, Revision 1.

In 10 CFR 50.49(b)(1)(ii), the NRC defines DBEs as “conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed.” The environmental conditions considered for the EQ program, described in FSAR Section 3.11.1.2, include normal, AOOs, and accident and postaccident environments resulting from DBEs, consistent with the requirements in 10 CFR 50.49(b)(1)(ii). The applicant provided the applicable environmental parameters required in 10 CFR 50.49(e) in FSAR Section 3C-2, “Scope,” which includes environmental parameters of pressure, temperature, relative humidity, radiation, chemical conditions, spray/wetting, and submergence. RG 1.180 specifies electromagnetic compatibility design requirements for electromagnetic and radio-frequency interference and power surges for equipment and is independent of the EQ Program. All equipment important to safety that will be qualified undergoes aging analysis to identify aging mechanisms that significantly increase the equipment’s susceptibility to DBA conditions as described in FSAR Appendix 3C, Section 3C.4.1.3, “Qualified Life Objective.” The staff finds that the applicant addressed the environmental parameters in 10 CFR 50.49(e), including temperature, pressure, humidity, radiation, and aging.

Regarding chemical effects, the staff reviewed the information in FSAR Section 3.11.4.1, “Chemical Environments,” which states the following:

> Appendix 3C defines applicable chemical environments for normal and AOO conditions. The chemical environments from the most limiting design basis event are also considered in the qualification of the equipment and are presented in Appendix 3C.

In 10 CFR 50.49(d)(7), the NRC states, “Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.” RG 1.89, Section C.5.a, states, “Synergistic effects known at this time are effects resulting from the different sequence of applying radiation and (elevated) temperature.” The applicant considered synergistic effects as described in FSAR Appendix 3C, Section 3C.5.3, “Synergistic Effects.” Synergistic effects can be categorized into two groups: (1) test sequence effects and (2) radiation dose rate effects. The applicant stated, “The possibility that significant synergistic effects may exist is addressed by using the worst-case aging sequence, conservative accelerated aging parameters and conservative, DBE test levels to provide confidence that any synergistic effects are enveloped.” The staff finds acceptable the applicant’s use of the parameters in FSAR Appendix 3C, Section 3C.5.3, to address synergistic effects, including the worst-case aging sequence.

In addition, the applicant considered power supply voltage and frequency variation in equipment design as required in 10 CFR 50.49(d)(2). FSAR Appendix 3C, Table 3C-5, "Environmental Qualification Program Margin Requirements," provides margins for power supply voltage and frequency. The staff finds that the information in Table 3C-5 addresses the requirements of voltage and frequency variations as described in 10 CFR 50.49(d)(2).

FSAR Appendix 3C, Section 3C.6, “Qualification Methodology,” describes the methods for qualifying electrical equipment important to safety. The methods to be used are (1) type testing, (2) qualification by analysis, (3) qualification by operating experience, and (4) combined qualification, using a combination of the first three methods. The staff finds that these methods are acceptable for EQ since they are specified in 10 CFR 50.49(f).

FSAR Table 3.11-1 lists the electrical and I&C equipment that requires qualification because it is located in a harsh environment, as required in 10 CFR 52.137(a)(13). The table describes the equipment, its location, EQ environment, operational time, EQ category, and PAM.

FSAR, Section 3.11.1.3, “Equipment Post-Accident Operating Time,” discussed the equipment operating times for the equipment important to safety subject to 10 CFR 50.49. The applicant notes that operating times for the equipment will vary, based on the credited function of the equipment. FSAR Table 3.11-1 provides the specific operating time for each of the equipment subject to EQ. FSAR Table 3C-4 provided the four different post-accident operating times for equipment subject to harsh environments. The staff finds the approach acceptable and is consistent with RG 1.89, Revision 1.

FSAR, Section 3.11.2.3, “Justification for Using Latest Institute of Electrical and Electronics Engineers Standards Not Endorsed by a Regulatory Guide,” provides a description and justification for using the latest IEEE Standards not endorsed by current RGs for qualification of the equipment. The applicant statement implied that multiple IEEE Standards not endorsed by the NRC were used in the development of the methodology of the equipment subject to 10 CFR 50.49. Based on staff’s request during an audit, the applicant revised Section 3.11.2.3.2 to include details of the justification and specific information regarding the use of IEEE Standards not endorsed in current RGs.

During the audit, the staff identified revisions to the FSAR, Table 3.11-1. Some of the revisions related to the removal and additions of some of the equipment from the list, reclassification of qualification programs, changes in EQ categories, and PAM type reclassification. The staff requested the applicant to provide justifications of the proposed changes and how the proposed changes are in accordance with the requirements of 10 CFR 52.137 (a)(13). The applicant clarified during the audit that the proposed changes are related to reclassification of certain equipment in other FSAR chapters. In addition, some equipment was classified as electrical, when the equipment is a mechanical qualification. The applicant also identified additional equipment important to safety that needed to be included in FSAR, Table 3.11-1.

Based on the staff’s review of the applicant’s EQ program described in FSAR Section 3.11, the staff finds that the program includes the qualification criteria (mild versus harsh environments, qualified life, operability time), design specification (normal and abnormal operating conditions for temperature or radiation), qualification methods (type test and combination of testing and analysis), and documentation needed to support electrical and I&C equipment. Therefore, the staff concludes that the applicant meets the requirements of 10 CFR 52.137(a)(13) because the SDAA includes the list of the equipment subject to 10 CFR 50.49(e).

3.11.4.1.2 Conformance to Regulatory Guide 1.89

RG 1.89, Revision 1, is the guidance for implementing the requirements and criteria of 10 CFR 50.49 for EQ of electrical equipment that is important to safety and located in a harsh environment. RG 1.89, Revision 1, endorses IEEE Std. 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” which provides guidance for demonstrating the qualification of Class 1E equipment by including test procedures and analysis methods. When these qualification requirements are met, the electrical and I&C equipment that is important to safety will perform its design function under normal, abnormal, DBE, post-DBE, and containment test conditions. FSAR Section 3.11 states that electrical equipment identified in FSAR Table 3.11-1 will be environmentally qualified using the guidance in IEEE Std. 323-1974. FSAR Section 3.11 states that equipment located in a harsh environment will be qualified in accordance with IEEE Std. 323-1974.

FSAR Table 3.11-2, "NRC Guidance and Industry Standards for Environmental Qualification" notes that PAM equipment will be environmentally qualified in accordance with RG 1.97, Revision 5, which endorses IEEE Std. 497-2016, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

Qualification of electrical equipment and components in a mild location is based on the normal local environment and seismic event. The applicant's EQ program addressed the acceptability of important-to-safety electrical equipment located in a mild environment (not subject to 10 CFR 50.49). Mechanical and electrical equipment required to perform a design function related to safety located in mild environments is qualified in accordance with the provisions of GDC 4. IEEE Std. 323-2003 provides guidelines to qualify electrical equipment and components in mild locations.

FSAR Table 3.11-2, "NRC Guidance and Industry Standards for Environmental Qualification," references additional guidance and industry standards for qualification of equipment in mild environment. This table also references IEEE Standard 535-2013, "IEEE Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations," to be used as supplemental guidance to IEEE Standard 323-2003 to address aging of valve regulated lead acid batteries.

Based on the review discussed above, the staff concludes that FSAR conforms to the RG 1.89, Revision 1, requirements for following IEEE Std. 323-1974 for qualification of electrical equipment for a harsh environment.

3.11.4.1.3 Compliance with 10 CFR Part 50, Appendix A

As stated in 10 CFR 52.137(a)(3), an application for a standard design approval must include the design of the facility. The design information includes (1) the PDC for the facility (the GDC in Appendix A to 10 CFR Part 50 establish minimum requirements for the PDC), (2) the design bases and the relation of the design bases to the PDC, and (3) information on materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with an adequate margin for safety. The staff's review of EQ is discussed below.

3.11.4.1.4 General Design Criterion 1

GDC 1 addresses requirements for quality standards and records concerning the quality standards for design, fabrication, erection, and testing of components important to safety. Components in the GDC 1 scope must have auditable records to document that environmental design and qualification requirements have been met.

According to FSAR Section 3.11.3, "Qualification Test Results," and Section 3C.7, "Documentation," for quality standards, all qualification records will be documented and maintained in an auditable form for the entire installed life. Records will be kept concerning the quality standards for design, fabrication, erection, and testing of components, in accordance with 10 CFR 52.79(a)(10) and 10 CFR 52.80(a). The staff finds that this complies with the quality standards and records requirements of GDC 1 because the applicant followed documentation requirements specified in IEEE Std. 323-1974, as endorsed by RG 1.89, Revision 1, and provides assurance that the EQ will be recorded and kept in an auditable form.

3.11.4.1.5 General Design Criterion 2

GDC 2 addresses the design bases for components important to safety which must withstand the effects of the most severe natural phenomena without loss of capability to perform their safety function.

Components within the scope of GDC 2 are designed with consideration of the environmental conditions or stressors resulting from natural phenomena, as part of the environmental conditions outlined in 10 CFR 50.49(e). The applicant stated the following in FSAR Section 3.11:

> Components in the scope of this section are designed with consideration of the environmental conditions or effects resulting from natural phenomena as part of the environmental conditions evaluated, including their location within safety designed structures.

The staff finds that the information in FSAR Section 3.11 complies with the requirements of GDC 2 by including effects resulting from natural phenomena in the design and qualification; therefore, the staff finds that the applicant meets GDC 2.

3.11.4.1.6 General Design Criterion 4

GDC 4 requires that components important to safety be designed to protect against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures. Components must also be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

In 10 CFR 50.49(f), the NRC describes the methodology used to qualify equipment that can perform its safety functions, under specified conditions such as applicable normal, abnormal, and DBE service conditions during its qualified life. FSAR Section 3.11 states that mechanical and electrical equipment required to perform a design function related to safety located in mild environments is qualified in accordance with the provisions of GDC 4. For each piece of equipment selected for EQ, the environmental parameters and the qualification process are listed in the associated EQRF.

For electrical and mechanical devices located in mild environments, compliance with the environmental design provisions of GDC 4 are generally achieved and demonstrated by proper incorporation of all relevant environmental conditions into the design process, including the equipment specifications. FSAR Table 3C-1, “Environmental Qualification Zones - Reactor Building,” shows equipment location zones designated as harsh environment, and FSAR Table 3C-3, “Designated Mild Environment Areas,” shows equipment location zones designated as mild environment. The staff noted that reactor pressure vessel (RPV) was not identified in a safety classification zone in either FSAR Table 3C-1 or FSAR Table 3C-3. The applicant subsequently explained that the environment within the RPV during a design basis event would not be more severe than during normal operation, thus meeting the criteria for the definition of a mild environment (ML24346A161). Further, the applicant notes that components inside the RPV require compliance with GDC 4 and are designed to accommodate the environmental conditions within the RPV. The applicant also notes that any equipment or components supporting electrical equipment are included in SDAA Table 3.11-1 to ensure that they are environmentally qualified. Based on this, the staff confirmed that the RPV meets the definition of a mild environment as defined in 10 CFR 50.49.

The qualification approach complies with GDC 4. In FSAR Appendix 3C, the applicant describes the implementation of the program and the methodology for dynamic qualifications. Since all EQ equipment is tested and qualified for the requirements of 10 CFR 50.49(f) (i.e., by simulating the effects or analyzing test data for equipment failures) to withstand the aforementioned normal operations, maintenance, and postulated accidents, including LOCAs, the applicant stated that the equipment is protected against dynamic effects that may result from equipment failures. The staff finds that the methodology in Appendix 3C complies with the requirements of GDC 4 because it establishes the program for EQ of electrical and mechanical equipment. Appendix 3C states that equipment subject to a harsh environment will be environmentally qualified using IEEE Std. 323-1974 and equipment subject to a mild environment will be qualified using IEEE Std. 323-2003. Therefore, the staff finds that the equipment subject to EQ is designed against dynamic effects.

3.11.4.1.7 General Design Criterion 23

GDC 23 requires that protection systems be designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, steam, water, or radiation) are experienced.

FSAR Appendix 3C, Section 3C.5.1, “Aging,” describes the mechanisms that significantly increase the equipment’s susceptibility to the DBA. The applicant further stated that the qualification process must consider all significant types of degradation that can affect the ability of the equipment to perform its design function related to safety during or following exposure to harsh environmental conditions. Since the qualification methods used to test its protection systems include the aging analysis discussed in Section 3C.4.2, the staff finds that this complies with the requirements of GDC 23 because the equipment subject to EQ provides reasonable assurance that the equipment can perform during and after a DBE. EQ testing provides reasonable assurance that the equipment can perform its design safety functions.

3.11.4.1.8 Compliance with 10 CFR Part 50, Appendix B

According to 10 CFR 52.137(a)(19), an SDAA must include a description of the QAP applied to the design of the SSCs of the facility. Appendix B to 10 CFR Part 50 presents the requirements for QAPs for nuclear power plants. The description of the QAP for a nuclear power plant shall include a discussion of how the applicable requirements of Appendix B to 10 CFR Part 50 were satisfied.

Compliance with 10 CFR Part 50, Appendix B, Criterion III, requires that measures be established to ensure that applicable regulatory requirements and the associated design bases are correctly translated into specifications, drawings, procedures, and instructions. This criterion is applicable since it includes requirements for test programs that are used to verify the adequacy of a specific design feature. Such test programs include suitable qualification testing of a prototype unit under the most adverse design conditions. FSAR Section 3C.6, “Qualification Methodology,” notes that the qualification defines tests, inspections, performance evaluation, acceptance criteria, and required analysis to demonstrate that, when called upon, the qualified equipment can perform its credited function for the required post-accident operating time. The applicant’s EQ process under EQ program includes appropriate qualification testing under the most adverse design conditions to verify the adequacy of a specific design feature. The staff finds that EQ related testing under the most adverse design conditions complies with 10 CFR Part 50, Appendix B, Criterion III.

Compliance with 10 CFR Part 50, Appendix B, Criterion XI, requires development of a test control plan to ensure that all tests needed to demonstrate a component’s capability to perform satisfactorily in service be identified and performed in accordance with written procedures that incorporate the requirements and acceptance limits contained in applicable design documents. RG 1.89, Revision 1, which endorses IEEE Std. 323-1974, outlines a planned sequence of test conditions (test plan) that meet or exceed the expected or specified service conditions. The applicant used the criteria to establish test procedures for the EQ program. The staff finds that the applicant complies with the requirements of Criterion XI because the applicant followed the methodology for test controls described in IEEE Std. 323-1974, which is endorsed by RG 1.89, Revision 1.

Compliance with 10 CFR Part 50, Appendix B, Criterion XVII, “Quality Assurance Records,” requires that sufficient records be maintained to furnish evidence of activities affecting quality. The EQ records must include inspections, tests, audits, monitoring of work performance, and materials analysis. Records pertaining to QA must be identifiable and retrievable. Complying with 10 CFR 50.49(j) requires that records must be maintained to furnish evidence of activities affecting quality. FSAR Section 3.11.3, “Qualification Test Reports,” states that the summaries and results of qualification tests for electrical and mechanical equipment and components are documented in the EQRF per Appendix 3C. Section 3C.7 provides the description of the EQRF and follows the guidance of IEEE Std. 323-1974. The staff finds that the applicant complies with the requirements of Criterion XVII because the applicant followed the methodology described in IEEE Std. 323-1974, which is endorsed by RG 1.89, Revision 1.

3.11.4.2 Environmental Qualification of Mechanical Equipment

The objective of this review is to determine if the SDA contains provisions to demonstrate that nonmetallic parts of active mechanical components are designed and qualified to be compatible with the postulated environmental conditions, including those associated with a LOCA.

The staff reviewed the description in FSAR Section 3.11 and Appendix 3C of the EQ program for nonmetallic parts of mechanical equipment to be used in the NuScale design. The staff confirmed the consistency with the NRC regulations and guidance to support the acceptability for reference in a COL application.

3.11.4.2.1 Identify Safety-Related Mechanical Equipment Located in Harsh or Mild Environment Areas and Operating Times

FSAR Section 3.11 states that the EQ program described in that section includes EQ of active mechanical equipment that performs a design function related to safety. FSAR Section 3.2 describes the safety classification of SSCs.

FSAR Section 3.11.1.1 states that the list of equipment that is in harsh environments and required to be environmentally qualified is provided in FSAR Table 3.11-1. That table also lists the operating times for mechanical equipment located in harsh environments.

FSAR Appendix 3C.4, “Qualification Process,” states that the mild environmental zones within the plant are listed in FSAR Table 3C-3, “Designated Mild Environment Areas.” The staff finds this identification of applicable mechanical equipment to be acceptable.

3.11.4.2.2 Identify Nonmetallic Subcomponents of Mechanical Equipment

FSAR Section 3.11.2, “Qualification of Mechanical Equipment,” states that FSAR Table 3.11-1, provides a list of the mechanical components with nonmetallic or consumable parts that are located in areas with a harsh environment and require EQ. The staff finds that this information is consistent with the guidance in DSRS Section 3.11 and is acceptable.

3.11.4.2.3 Identify the Environmental Conditions and Process Parameters for which the Equipment Must Be Qualified

FSAR Section 3.11.2.2, states that environmental conditions considered in the design of the NuScale reactor include AOOs and normal, accident, and postaccident environmental conditions. FSAR Appendix 3C specifies the environmental parameters (e.g., radiation, temperature, chemical effects, humidity from steam, pressure, wetting, and submergence) applicable to the various environmental conditions in specific plant building and room locations. Service conditions include the process conditions anticipated or experienced by equipment during operation of the plant.

For mechanical equipment, the environmental design and qualification consider both the external environmental conditions and the internal operational service conditions of the equipment. FSAR Section 3.11 and Appendix 3C describe the external environmental conditions.

The staff finds that the identification of environmental conditions and process parameters for nonmetallic parts of mechanical equipment in FSAR Section 3.11 and Appendix 3, is consistent with the guidance in DSRS Section 3.11 and is acceptable.

3.11.4.2.4 Identify Nonmetallic Material Capabilities

FSAR Section 3.11.2.2 states that for mechanical equipment that must function during or following exposure to a harsh environment, demonstrating the non-metallic parts and components of the equipment are suitable for the postulated design basis environmental conditions, ensures compliance with the environmental design provisions of GDC 4. FSAR Table 3.11-1 lists the mechanical components that contain non-metallic or consumable parts located in harsh environments that require EQ.

The staff finds that the identification of nonmetallic material capabilities in FSAR Section 3.11 for nonmetallic parts of mechanical equipment is consistent with the guidance in DSRS Section 3.11 and is acceptable.

3.11.4.2.5 Evaluate Environmental Effects on the Nonmetallic Components

FSAR Section 3.11.2.2 and Appendix 3C.3 state that mechanical equipment that performs an active design function related to safety during or following exposure to harsh environmental conditions is qualified in accordance with ASME QME-1, Appendix QR-B, with the following exceptions:

a. QR-B5200, "Identification and Specification of Qualification Requirements," paragraph (g), material activation. The applicant stated that in accordance with QR-B5200, nonmetallic material will be qualified to perform its intended functions and, although activation energy might not be used for identification purposes per QR-B5200, the activation energy will be applied to the thermal aging equation for determining material degradation and qualification. The staff finds the applicant's proposal acceptable because it meets the intent of Appendix QR-B of ASME QME-1 in that the material activation energy is applied to the thermal aging equation.

b. QR-B5300, "Selection of Qualification Methods." The applicant noted that the last paragraph in QR-B5200 states, "The shelf life of all nonmetallics, and any applicable storage limitations, should be determined and recorded in the qualification documentation." The applicant stated that shelf life and preservation requirements are documented in accordance with ASME Standard NQA-1-2008, Requirement 13 and Subpart 2.2, in lieu of ASME QME-1-2017, Appendix QR-B5300, and that these requirements are not included in the EQRF but are documented separately. The staff finds the applicant's proposal acceptable because the material shelf life and preservation requirements will be identified as specified in the NQA-1-2008 documentation. The staff accepts the use of NQA-1-2008 in 10 CFR 50.55a(b)(3)(i) when it is applied consistently with the QA requirements in 10 CFR Part 50, Appendix B.

c. QR-B5500, "Documentation," paragraph (h), shelf life preservation requirements. The applicant stated that shelf life and preservation requirements are documented in accordance with NQA-1-2008, Requirement 13 and Subpart 2.2, in lieu of ASME QME-1-2017, Appendix QR-B5300, and that these requirements are not included in the EQRF but are documented separately. The staff finds the applicant's proposal acceptable because the material shelf life and preservation requirements will be identified as specified in the NQA-1-2008 documentation. The staff accepts the use of NQA-1-2008 in 10 CFR 50.55a(b)(3)(i) when it is applied consistently with the QA requirements in 10 CFR Part 50, Appendix B.

FSAR Section 3.11.1.1 states that for mechanical devices located in mild environments, compliance with the environmental design provisions of GDC 4 is generally achieved and demonstrated by proper incorporation of all relevant environmental conditions in the design process, including equipment specification compliance.

FSAR Section 3.11.2.2and Appendix 3C.3 state that mechanical equipment located in mild environments and required to perform a credited function is qualified in accordance with the provisions of GDC 4. For each piece of equipment selected for EQ, the EQRF lists the environmental parameters and the qualification process. The staff finds this description of compliance with GDC 4 for the capability of mechanical equipment located in mild environments to be acceptable.

3.11.4.2.6 Equipment Specification

FSAR Appendix 3C, Section 3C.4.1, “Design Specifications,” states that the equipment specification identifies the applicable codes and standards, required operating times, performance requirements, credited functions, service conditions, accepted methods of qualification, and acceptance criteria.

During the NuScale DCA review, the staff audited the process for design equipment specifications established by NuScale, and issued an audit report, dated December 20, 2019 (ML19331A397). The NuScale US460 design (SDA) commits to using the same process and standards that were used in the US600 design (DCA), with the exception of updating to a later QME-1 standard. On the basis of its detailed DCA audit of the NuScale’s processes, the staff finds the design equipment specification process to be applicable to the US460 design because the audit demonstrated the process that a COL applicant will use to establish the specifications and qualification of components by implementing the ASME QME-1 standard as accepted by the NRC in RG 1.100. In addition, the environmental conditions for the US600 design components and the US460 design components are sufficiently similar such that the process for design equipment specifications can be applied for both designs. Further, as specified in the SDA FSAR, a COL applicant for the US460 design will be required to implement the program specified in the ASME QME-1 standard as accepted by RG 1.100 for the qualification of components to perform their design-basis safety functions. Therefore, the NRC staff finds the NuScale process for developing equipment specifications to be acceptable for the NuScale SDA.

3.11.4.3 Environmental Qualification of the Radiation Environment

The staff reviewed FSAR Section 3.11 and supporting documentation to ensure that the radiological effects on electrical and mechanical equipment important to safety are in accordance with GDC 4 and 10 CFR 50.49. The subject information is found in FSAR Section 3.11 and Appendix 3C. Guidance for the staff’s evaluation applicable to the NuScale SDA design appears in Section 3.11 of the NuScale DSRS; in RG 1.89, Rev 1; and in Appendix I to RG 1.183, Revision 0.

As discussed in FSAR Section 3C.4.2.5, “Design Basis Event Radiation Conditions,” the radiation environment for EQ is based on the total integrated gamma and neutron dose during normal operation and the total integrated gamma and beta dose following an accident. Beta radiation is negligible during normal operation, and there will not be significant neutron radiation following a DBA because the reactor will be shut down and the normal operation neutron dose will dominate. In addition, as provided in FSAR Section 3.11.1.2, the basis for designating areas as a harsh environment is based on the total integrated dose (TID) from normal operation plus accidents. If the TID is greater than 100 gray (Gy) (1.0×10^4 rad), the equipment is considered to be in a harsh radiation environment. If the TID is greater than 10 Gy (1.0×10^3 rad), but less than 100 Gy (1.0×10^4 rad), the radiation environment is considered harsh for electronic equipment but mild for other equipment. This approach is consistent with DSRS 3.11, RG 1.89, RG 1.183, and 10 CFR 50.49, including 10 CFR 50.49(e)(4), in that the total dose over the installed life of the equipment during normal operation and accident conditions must be considered and the criteria for classifying equipment as in a mild or harsh environment is appropriate. Finally, this is consistent with GDC 4 because all significant types of radiation during normal and accident conditions are considered. Based on this, the staff finds this approach acceptable.

Normal operational dose includes neutron and neutron-induced gamma radiation from the fission process, gamma radiation from fission products, and gamma radiation from corrosion and activation products in the reactor coolant (e.g., nitrogen-16), as well as activated components. While RG 1.89 specifies that the EQ analysis should be based on an assumed 1-percent failed fuel percentage, the radiation source terms and dose rates proposed by NuScale are based on an assumed 0.066-percent failed fuel percentage. The 0.066-percent failed fuel percentage is consistent with the dose equivalent iodine-131 and xenon-133 values specified in TS Limiting Condition of Operation 3.4.8. Since the TS prohibit long-term operation above 0.066-percent failed fuel percentage, the staff evaluated NuScale's proposal and finds the use of 0.066-percent failed fuel acceptable since this is the maximum amount of radioactivity expected to be present in systems. FSAR Chapter 12, "Radiation Protection," provides information about the normal operation source terms used to develop the normal operation dose rate information provided in FSAR Appendix 3C, Table 3C-6. Chapter 12 of this SER presents the staff's review of these source terms.

FSAR Appendix 3C, Table 3C-6, provides the normal operation TIDs by EQ zone. The zones include all areas of the RXB and the control building (CRB). This includes zoning for inside of containment and under the bioshield. Equipment is designated an EQ zone based on the location that it is located. The EQ zoning for specific equipment is found in FSAR Table 3.11-1. The staff reviewed the normal operation TIDs based on the source terms, radiation protection design features, and radiation zoning provided in FSAR Chapter 12 and information provided to the staff during an audit (ML23067A298). The staff reviewed the normal operation TIDs for certain areas in Table 3C-6, and found them to be consistent with the information in FSAR Chapter 12, information audited by the staff, and staff confirmatory calculations. Staff also relied on past experience from the NuScale DCA review to inform the SDAA review, as applicable. Dose rates included consideration for penetrations through the pool wall into the reactor module bay. As a result of the review, the staff found the normal operation dose rates to be acceptable.

Regarding accident radiation doses to equipment, the accident source term methodology topical report TR-0915-17565, Revision 4 (ML20057G132), provides information on the radiation accident source terms and EQ dose methodology, in combination with the accident source term information in FSAR Chapters 3, 12, and 15. In 10 CFR 50.49(e)(4), the NRC requires, in part, that the radiation environment in the electric equipment qualification program must include the radiation environment associated with the most severe DBA during or following which the equipment is required to remain functional. In addition, 10 CFR Part 50, Appendix A, GDC 4, requires, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; including LOCAs. While RG 1.89 and RG 1.183, Appendix I, indicate that significant core damage should be assumed, in TR-0915-17565, NuScale indicated that there are no credible DBA scenarios in the NuScale design that result in core damage.

As described in SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," dated August 16, 2019 (ML19107A455), the staff has determined that core damage events do not need to be evaluated to address 10 CFR 50.49(e)(4) in the NuScale design. In addition, based on the approach described in SECY-19-0079 and also since the design of equipment under 10 CFR Part 50, Appendix A, GDC 4, for other parameters, such as pressure and temperature, are not evaluated for core melt accidents, GDC 4 is treated similar to 10 CFR 50.49(e)(4) and equipment is not evaluated to core damage radiological conditions, in the NuScale design. However, while core damage is not considered in addressing the requirements of 10 CFR 50.49(e)(4) and GDC 4, the staff notes that a core damage equipment survivability analysis is needed for equipment that is required to function to withstand core damage events, as required by 10 CFR 52.137(a)(23) (SDA equivalent of 10

CFR 52.47(a)(23)) and 10 CFR 50.44 and as provided in SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990 (ML003707849), SECY-93-087 (ML003760768), and the associated SRM for SECY-93-087 (ML003708056). The radiological conditions associated with equipment survivability and the equipment included within the equipment survivability analysis are discussed in FSAR Section 15 and Section 19.

Since core damage is not considered for 10 CFR 50.49 and GDC 4 in the NuScale design, the applicant proposed using the iodine spike design-basis source term as the bounding source term for EQ to meet 10 CFR 50.49(e)(4). Appendix B to the TR provides the methodology used in calculating the dose for equipment inside containment and under the bioshield. While the iodine spike source term is not based on a specific accident, a rapid increase (or spike) in reactor coolant radionuclide concentrations is known to occur following transients at nuclear power plants, and the spiking of iodine is discussed in RG 1.183. The iodine spike source term was evaluated by the staff and was found to be acceptable for use in the NuScale design for the accident source term for areas inside of containment and under the bioshield, as well as shine from those sources, in addition to areas outside of the NPM bay prior to the isolation of containment. The justification for the acceptability of the methodology for this source term, along with the conditions and limitations on the use of the iodine spike source term, is found in the staff’s SER (ML19290F633) dated October 2019, for TR-0915-17565. In addition, the NuScale SDAA Chapter 15 includes accident scenarios with lower coolant levels over the core for a longer duration than in the DCA. Therefore, the staff considered the potential impacts of lower coolant levels on the TIDs. Through information provided by the applicant and reviewed by staff during the audit, the staff finds that the radiation dose due to radiation shine from the shutdown core with low coolant levels was insignificant to the TID inside containment and under the bioshield, when considering the normal operation dose combined with the iodine spike source term.

In addition, the staff reviewed the accident TIDs provided in FSAR Table 3C-8. The staff found the dose rates consistent with the source term in FSAR Table 12.2-31 based on a comparison of the accident source terms and total integrated dose rates in the NuScale US600 and US460 designs and staff confirmatory calculations. For areas other than inside of containment and under the bioshield, the staff determined that the accident dose is expected to be low since there is no core damage and because the containment will be isolated shortly after initiation of the accident and the radioactive material will be confined to inside of these areas. There may be a brief release of radioactive materials following the onset of an accident, before containment is isolated. The dose from this is accounted for in Table 3C-8. Based on this, the staff found the postaccident radiation doses for areas outside of containment and under the bioshield to be acceptable, with the conditions and limitations found in the staff’s SER for TR-0915-17565.

The applicant designated each environmental zone as either mild or harsh based on the maximum normal operation and post-accident environmental conditions in that zone. FSAR Table 3C-1, “Environmental Qualification Zones,” Table 3C-2, “Designated Harsh Environment Areas” and Table 3C-3, “Designated Mild Environment Areas,” describes the harsh and mild radiation environmental zones in the RXB (including inside containment and under the bioshield) and CRB (there is no equipment requiring qualification in the RWB or other plant buildings, so EQ zoning is not provided for these areas). The staff notes that Zones RXBG-5E, RXBG-7E, and CRB-3E are harsh only for electronic equipment located in certain areas within the zones that may exceed 10 Gy (1.0 x 10^3 rad) TID because the TID in these areas does not exceed 100 Gy (1.0 x 10^4 rad). These dose criteria for harsh and mild radiation environments are consistent with the methodology discussed earlier in this section and are acceptable. The staff reviewed the normal and post-accident radiation TIDs for each zone and found the classifications of each environmental zone to be consistent with the TIDs provided and to therefore be acceptable from a radiation dose perspective.

FSAR Section 3C.5.3 specify that synergistic effects of environmental conditions (such as the effects of radiation and temperature) are also considered. Synergistic effects may be significant for certain materials and conditions. This is in accordance with the requirements of 10 CFR 50.49(e)(7), in that synergistic effects are considered when these effects are significant. Therefore, the staff found this to be acceptable.

Equipment located in harsh environmental zones is designed to perform under all appropriate environmental conditions. If the dose in a harsh environment is higher than the calculated dose, it could result in a TID (including consideration for postulated accidents) that exceeds the TID that the equipment in that area was designed to withstand. Therefore, FSAR Section 3.11.4.2 specifies that the radiation doses are monitored during plant life and compared to the calculated doses. If the measured doses are higher than the calculated doses, the equipment is evaluated to ensure that it remains qualified. To address the monitoring of equipment to ensure that the dose requirements are not exceeded, Section 3.11.4.2 includes the following COL item (COL Item 3.11-2):

> An applicant that references the NuScale Power Plant US460 standard design will ensure the Environmental Qualification Program cited in COL Item 3.11-1 includes a description of how equipment subject to program requirements will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh or mild environments will remain qualified if an actual environment parameter, such as temperature, pressure, humidity, radiation, or chemical exposure deviates from the acceptable range for which the component is qualified.

The staff evaluated the COL item and found it acceptable for the COL applicant to describe how equipment will be monitored and managed throughout the life of the plant.

FSAR Section 3C.5.1.2 indicates that radiation qualification testing can be performed in one exposure, simulating the TID (normal plus accident doses) or that it can be performed in separate exposures (one for normal and one for the accident dose). This allows flexibility in defining the specific test sequence, with consideration of factors, such as the applicability of test sequence synergistic effects, the relative magnitude of the normal dose compared to the accident dose, and the time to perform the irradiation. The staff evaluated this and determined that since both normal and accident doses are fully considered, it is acceptable to either perform the testing as one exposure (combining normal and accident TID) or perform the normal and accident TID testing separately. The staff finds this to be consistent with the requirements of 10 CFR 50.49 and GDC 4.

More information on the staff's evaluation of the applicant's source terms and dose rates in the NuScale design, which are used in the development of EQ TID rates, can be found in Chapter 12 of this SER. Additional information on the DBA source terms and EQ dose methodology following accidents can be found in TR-0915-17565.

Based on the above information, the staff finds the applicant's approach to the radiological aspects of EQ to be in accordance with 10 CFR 50.49 and GDC 4 and to be acceptable.

3.11.5 Combined License Information Items

Table 3.11-1 lists COL information item numbers and descriptions related to EQ from FSAR Table 1.8-1.

Table 3.11-1: NuScale COL Information Items for Section 3.11

Item No.	Description	FSAR Section
COL Item 3.11-1	An applicant that references the NuScale Power Plant US460 standard design will submit a full description of the Environmental Qualification Program and milestones and completion dates for program implementation.	3.11.3
COL Item 3.11-2	An applicant that references the NuScale Power Plant US460 standard design will ensure the Environmental Qualification Program cited in COL Item 3.11-1 includes a description of how equipment subject to program requirements will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh or mild environments will remain qualified if an actual environment parameter, such as temperature, pressure, humidity, radiation, or chemical exposures deviates from the acceptable range for which the component is qualified.	3.11.4.2
COL Item 3.11-3	An applicant that references the NuScale Power Plant US460 standard design will implement an Environmental Qualification Operational Program that incorporates the aspects in Section 3.11.5 specific to the environmental qualification of mechanical and electrical equipment. This program will include an update to Table 3.11-1 to include commodities that support equipment listed in Table 3.11-1.	3.11.5

During an audit (ML23067A300), the staff noted that the description for monitoring equipment located in harsh conditions described in COL Item 3.11-2 only discussed radiation and did not include the environmental parameters described in 10 CFR 50.49(e). The staff requested the applicant to explain how other environmental parameters described in 10 CFR 50.49(e) would be monitored throughout the life of the equipment. In response, the applicant revised COL Item 3.11-2 to include other environmental parameters including temperature, pressure, humidity, and chemical exposure, and provided FSAR markups. The staff finds this acceptable.

3.11.6 Conclusion

As described above, the staff has reviewed all of the relevant information applicable to FSAR Section 3.11 for the EQ program for mechanical and electrical equipment. Based on the above, the staff concludes that the applicant has provided reasonable assurance that the EQ program complies with 10 CFR 52.137(a)(13); 10 CFR 50.49(d); 10 CFR Part 50, Appendix B, Criteria III, XI and XVII; and GDC 1, 2, 4, and 23.

The staff concludes that the provisions in the NuScale SDAA and TR-0915-17565 for the EQ of mechanical and electrical equipment in the NuScale design, including the statements in the

COL Items listed in Table 3.11-1 of this report, are acceptable and meet the applicable NRC requirements and are consistent with guidance. This conclusion is based on the applicant having specified provisions in the FSAR that mechanical, electrical, and I&C equipment, including digital I&C equipment designated as important to safety, addressed in the EQ program is capable of performing its design functions under all normal environmental conditions, AOOs, and accident and postaccident environmental conditions.

3.12 ASME BPV Code Class 1, 2, and 3 Piping Systems and Associated Support Design

3.12.1 Introduction

This section covers the design and structural integrity of piping systems and supports used in Seismic Category I and non-Seismic Category I piping systems, the failure of which could potentially affect Seismic Category I systems. The staff's evaluation considered the adequacy of the structural integrity, as well as the functional capability, of piping systems. The review includes piping designed in accordance with the ASME BPV Code, Section III, Subsections NB, NC, and ND, as incorporated by reference in 10 CFR 50.55a (also referred to as ASME Class 1, 2, and 3 or QGs A, B, and C, respectively).

The review also includes buried piping, instrumentation lines, and interaction of non-seismic Category I piping with Seismic Category I piping. The following sections of this report provide the staff's evaluation of the adequacy of the SDAA piping analysis methods, design procedures, acceptance criteria, and verification of the design.

The staff's evaluation included the following:

- regulatory criteria
- applicable codes and standards
- methods to be used in the design of piping and pipe supports
- modeling of piping systems
- pipe stress analysis criteria
- pipe support design criteria

3.12.2 Summary of Application

FSAR: In FSAR Section 3.12, "ASME BPV Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports," the applicant described the methods of piping analysis and addressed the design of piping systems for loadings from normal operating conditions, system operating design transients, postulated pipe breaks, and seismic events. The section also includes loading combinations for piping analysis.

FSAR Section 3.12.1, "Introduction," states that NuScale has adapted the graded approach for piping design, which the staff proposed in the March 4, 2014, NRC white paper "Piping Level of Detail for Design Certification" (ML14065A067). The paper's graded approach to the piping analysis for SDA is consistent with the Commission's direction in SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," dated November 8, 1990 (ML003707889). The white paper, in conjunction with SECY-90-377, discusses requirements for preliminary and completed final piping design analyses (in this context, "final" (as opposed to "preliminary") piping design analysis refers to the completed, as-designed piping stress analysis for DCA and not the ASME-certified pipe stress analysis reports). The level of detail of the piping design for the DC is to be proportionate with the importance of the piping systems or piping segments to safety. This is also applicable for the level of information for an SDA.

ITAAC: ITAAC for the NPM appear in SDAA Part 8, Table 2.1-2. ITAAC No. 02.01.01 in this table provides for the ASME BPV Code Class 1, 2, and 3 piping systems to comply with ASME BPV Code, Section III, requirements through the completion of ASME BPV Code, Section III, design reports for the ASME BPV Code Class 1, 2, and 3 as-built piping systems. Section 14.3 of this SER discusses NuScale ITAAC.

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

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

3.12.3 Regulatory Basis

The applicant's piping and pipe support design criteria, including the analysis methods and modeling techniques, are acceptable if they meet the applicable codes and standards and are consistent with regulatory guidance documents, commensurate with the safety function to be performed. This will ensure that the piping design criteria meet the relevant requirements in 10 CFR Part 50 to ensure structural integrity and pressure boundary leakage integrity of piping and components, as well as structural integrity of pipe supports in nuclear power plants. The acceptance criteria are based on meeting the relevant requirements of the following regulations for piping systems, piping components, and their associated supports, as described below:

- 10 CFR 50.55a and GDC 1, as they relate to piping systems, pipe supports, and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed

- 10 CFR Part 50, Appendix B, which sets QA requirements for safety-related equipment

- GDC 2 and Appendix S to 10 CFR Part 50, with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC 4, with regard to piping systems and pipe support important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal as well as postulated events, such as a LOCA, and dynamic effects

- GDC 14, with regard to the RCPB being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- GDC 15, with regard to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design condition of the RCPB is not exceeded during any condition of normal operation, including AOOs

- 10 CFR 52.137(a)(22), which requires information necessary to demonstrate how operating experience insights have been incorporated into the plant design

The guidance in SRP Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports,” lists the acceptance criteria adequate to meet the above requirements, as well as 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:

- SECY-90-377 and its associated SRM (ML003707889 and ML003707892, respectively)
- SECY-93-087 and its associated SRM (ML003708021 and ML003708056, respectively).
- DSRS Section 3.7.2, Revision 0, “Seismic System Analysis,” issued June 2016 (ML15355A389)
- DSRS Section 3.7.3, Revision 0, “Seismic Subsystem Analysis,” issued June 2016 (ML15355A402)
- SRP Section 3.9.1, Revision 3, “Special Topics for Mechanical Components,” issued March 2007.
- SRP Section 3.9.2, Revision 3, “Dynamic Testing and Analysis of Systems, Structures, and Components,” issued March 2007 (ML070230008)
- SRP Section 3.9.3, Revision 3, “ASME BPV Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures,” issued April 2014 (ML14043A231)
- NRC white paper, “Piping Level of Detail for Design Certification,” dated March 4, 2014 (ML14065A067)
- NUREG/CR-1980, “Dynamic Analysis of Piping Using the Structural Overlap Method,” issued March 1981
- EPRI TR-1011955, Materials Reliability Program (MRP)-146, Revision 1, “Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines,” dated June 22, 2011

3.12.4 Technical Evaluation

The staff evaluated the structural design of piping and pipe supports using the industry codes and standards, RGs, and staff technical reports listed in the SRP. During the review, the staff also considered the level of detail for an SDA applicant as detailed in the white paper referenced above, and industrial practice and programs.

In FSAR Section 3.12.1, the applicant described the use of the graded approach in completing final and preliminary piping design analyses and identified the scope of the graded approach, as follows. It stated that preliminary piping analyses have been completed for the high-energy piping larger than DN 25 (NPS 1) in the NPM in order to support HELB evaluations. That includes Class 1 RCPB piping (greater than DN 25 (NPS 1)) inside containment, Class 2 MS and FW lines up to the first six-way restraint beyond the CIVs, and Class 2 DHRS lines. The applicant further stated that the preliminary and detailed piping analyses use all applicable loads mentioned in FSAR Section 3.12.5.3, “Loadings and Load Combinations.” The staff finds that the selection of piping systems for design pipe stress analysis and loads used, as shown in

FSAR Section 3.12.1, is acceptable because it is based on the safety function, piping size, and layout and also because the selection is consistent with the staff's discussion of level of detail for DCs in SECY-90-377 and the pertinent staff white paper.

The preliminary calculated pipe stresses and fatigue CUFs (applicable to Class 1 piping), meet allowable limits in ASME BPV Code, Section III and therefore are acceptable. The final piping design reports will be completed during ITAAC. Section 14.3 of this SER discusses NuScale ITAAC.

3.12.4.1 Codes and Standards

GDC 1 requires that SSCs important to safety be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed. When generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. In 10 CFR 50.55a, the NRC requires that certain systems and components of boiling- and pressurized-water nuclear power plants must meet certain requirements of the ASME BPV Code. The regulation specifies the use of the latest edition and addenda endorsed by the NRC and any limitations discussed in the regulations. In RG 1.84, the staff lists acceptable ASME BPV Code, Section III, Code Cases for design and materials acceptability and any conditions that apply to them.

In FSAR Section 3.12.2, "Codes and Standards," the applicant discussed the applicable codes and standards for the design of ASME Class 1, 2, and 3 piping systems.

3.12.4.1.1 ASME BPV Code

FSAR Section 3.12.2.1, "ASME Boiler and Pressure Vessel Code," indicates that safety-related piping is designed in accordance with the ASME BPV Code, Section III, 2017 Edition (no addenda). In using ASME BPV Code, Section III, the applicant stated that it followed the regulatory conditions found in 10 CFR 50.55a(b)(1). Hence, the application of ASME BPV Code, Section III, by the applicant is acceptable.

3.12.4.1.2 ASME BPV Code Cases

FSAR Section 3.12.2.2, "ASME BPV Code Cases," which states that ASME BPV Code Cases may be used if conditionally or unconditionally approved by RG 1.84, is acceptable to the staff.

3.12.4.1.3 Design Specifications

ASME BPV Code, Section III, requires that design specifications be prepared for ASME Class 1, 2, and 3 components, such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design inputs. The Code also requires a design report for ASME Class 1, 2, and 3 piping and components in Subsection NCA, "General Requirements for Division 1 and Division 2," paragraph NCA-3550, "Requirements for Design Output Documents."

The as-designed piping should be in accordance with the governing design specification. In the NuScale design, this requirement is accomplished via ITAAC No. 02.01.01 in SDAA Part 8,

Table 2.1-3. This ITAAC specifies that the ASME BPV Code Class 1, 2, and 3 piping systems comply with ASME BPV Code, Section III, requirements through the completion of ASME BPV Code, Section III, design reports, for the ASME BPV Code Class 1, 2, and 3 as-built piping systems.

According to FSAR Section 3.12.1, the requirements for the design, analysis, materials, fabrication, inspection, examination, testing, certification, packaging, shipping, and installation of piping systems within the NPM are documented in an ASME design specification for Class 1, 2, and 3 piping.

Based on its review, the staff finds the FSAR statements on piping design specifications acceptable because they are in accordance with ASME BPV Code, Section III, which is incorporated by reference in 10 CFR 50.55a.

3.12.4.1.4 Conclusions on Codes and Standards

Based on the review described above, the staff concludes that the piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

- The applicant satisfied the requirements of GDC 1 and 10 CFR 50.55a by specifying appropriate codes and standards for the design and construction of safety-related piping systems.
- The applicant stated in FSAR Section 3.12.2.2, "American Society of Mechanical Engineers Code Cases," that ASME BPV Code cases may be used for ASME Class 1, 2, and 3 piping if they are approved in RG 1.84.

3.12.4.2 Piping Analysis Methods

3.12.4.2.1 Experimental Stress Analysis Method

In FSAR Section 3.12.3.1, "Experimental Stress Analysis Method," the applicant stated that experimental stress analysis methods will not be used to qualify piping for the NuScale design. The staff finds this acceptable based on SRP Section 3.12, Acceptance Criterion II.A.i.

3.12.4.2.2 Modal Response Spectrum Method

FSAR Section 3.12.3.2, "Modal Response Spectrum Method," shows that the response spectrum analysis is performed using either the uniform support motion (USM) or the independent support motion (ISM) technique. Piping attached to the NPM uses the in-structure response spectra (ISRS) of the NPM. Piping attached to the RXB uses the ISRS of the RXB.

The staff evaluated the modal response spectrum method and documented the results of its evaluation in the following sections.

3.12.4.2.2.1 Development of Seismic Input and Floor Response Spectra

According to FSAR Section 3.12.3.2.1, "Development of In-structure Response Spectra," In-structure response spectra of the NPM are determined using dynamic analysis of a three-dimensional, finite element model of the NPM structural system as described in US460 NuScale Power Module Seismic Analysis Technical Report TR-121515. The staff reviewed TR-121515, and its acceptance is documented in Section 3.7 of this report. The applicant also states that "Uncertainties in the structural frequencies, which represent uncertainty or approximations of material and structural properties, are accounted for by peak broadening ±15 percent". FSAR Section 3.7 discusses the development of response spectra. Section 3.7.2 of this report discusses the staff's evaluation of the development of seismic input and floor response spectra.

3.12.4.2.2.2 Uniform Support Motion Method

FSAR Section 3.12.3.2.2, "Uniform Support Motion," shows that piping systems supported at multiple points within a structure may be analyzed using the USM method. This analysis method applies a single set of spectra at all support locations, which envelops all the individual response spectra for these locations and thus defines a uniform response spectrum (URS). SRP Section 3.7.3, "Seismic Subsystem Analysis," Acceptance Criterion II.9, indicates that the USM method is a conservative and acceptable approach for analyzing component items supported at two or more locations to calculate the maximum inertial response of the component. Therefore, the staff finds the USM method acceptable.

3.12.4.2.2.3 Modal Combination

FSAR Sections 3.12.3.2.3 through 3.12.3.2.7 describe the method of combining modal responses for response spectrum analysis of piping systems.

For piping systems with no closely spaced modes, periodic modal responses for modes at frequencies lower than zero period acceleration (ZPA) are obtained by using the SRSS method. This method is the recommended approach in RG 1.92, Revision 3, for combining modal responses for modes that are not closely spaced, and therefore, it is acceptable.

The applicant used the RG 1.92, Revision 3, definition of closely spaced modes, which is a function of the critical damping ratio, and combined their periodic modal responses by either the algebraic double sum method of RG 1.92, Revision 3, or the absolute double sum method of RG 1.92, Revision 1, issued February 1976 (ML003740290). These methods both conform to the current NRC guidance in RG 1.92, Revision 3, for combining closely spaced periodic modes and therefore are acceptable.

For calculating the remaining or residual rigid responses for the contribution of modes at frequencies higher than ZPA, the applicant used the missing-mass method or the static ZPA method in the piping seismic analysis. These methods are in the RG 1.92, Revision 3, guidance and therefore are acceptable to the staff.

In combining responses from periodic modes with responses from rigid modes, the applicant used the SRSS method, at a component level, from Regulatory Position C.1.5.1, Combination Method A, of RG 1.92, Revision 3, which is acceptable.

3.12.4.2.2.4 Directional Combination

In FSAR Section 3.12.3.2.8, "Directional Combination," the applicant showed that when performing seismic response spectrum analysis, it combined modal responses caused by seismic inputs in the three orthogonal directions utilizing the SRSS combination method described in RG 1.92, Revision 3, which, therefore, is acceptable to the staff.

3.12.4.2.2.5 Seismic Anchor Motion Analysis Method

The staff notes that, for piping systems that are anchored and restrained to floors and walls of structures that have differential movements during a seismic event, additional forces and moments resulting from the differential supporting structure movements are induced in the system.

FSAR Section 3.12.3.2.9, "Seismic Anchor Motion," indicates that maximum relative anchor and support displacements are obtained from the structural response calculations or from the applicable ISRS, which are then imposed on the supported piping in the most unfavorable combination using the static analysis method. This is known as seismic anchor motion (SAM) analysis.

FSAR Section 3.12.3.2.9, shows that when using the USM method for dynamic seismic inertia analysis, the responses from the dynamic analysis are combined with the responses from the static SAM analysis by the absolute sum method, which is recommended in SRP Section 3.9.2. It also shows that when using the independent support motion (ISM) method of dynamic seismic inertia analysis, to find the total response, the responses from the dynamic seismic analysis and from the static SAM analysis are combined by the SRSS method. This method is recommended in NUREG-1061, Volume 4, Section 2, and SRP Section 3.7.3.

Because, as discussed above, the applicant used NRC guidance in considering the effects of SAM in the NuScale piping analysis, the staff finds the applicant's method of SAM analysis acceptable.

3.12.4.2.3 Independent Support Motion Method

As an alternative to the USM method of seismic analysis, in FSAR Section 3.12.3.2.10, "Independent Support Motion Method," the applicant proposed to use the ISM response spectrum seismic analysis method for piping with multiple supports. As noted in SRP Section 3.7.3, both methods are acceptable to the staff. FSAR Section 3.12.3.2.10 shows that when the ISM method is used, all related criteria in NUREG-1061 will be followed. The staff finds the applicant's use of the ISM response spectrum method for seismic analysis of piping acceptable because the applicant's description of the ISM method is the same as the recommended method in SRP Section 3.7.3 and is found in NUREG-1061, Volume 4.

In the ISM method of piping analysis, the supports are divided into groups. A support group is defined by supports that have the same response spectrum. This usually means all supports attached on the same floor (or portions of a floor) elevation of a structure. During analysis, the specified response spectrum for each specific group is applied to all supports in that group, while supports in all other groups are held stationary. After the individual group responses are determined, they are combined by the absolute sum method, which is recommended in NUREG-1061, Volume 4. The applicant stated that in the ISM method, the damping values described in RG 1.61, Revision 1, are used, which, therefore, is acceptable to the staff.

As discussed above, the staff review finds the applicant's ISM method of piping analysis acceptable because the applicant used NRC guidance to perform ISM seismic response spectrum piping analysis.

3.12.4.2.4 Time-History Method

FSAR Section 3.12.3.4, “Time-History Method,” states that the time-history method may be used for seismic inertial dynamic analysis of piping and for other dynamic analyses of piping resulting from transient loadings such as water hammer, steam hammer, and loads from postulated pipe breaks. As described in SRP Sections 3.7.1, 3.7.2, and 3.7.3, as well as RG 1.92, the time-history method for seismic analysis is acceptable to the staff. The applicant stated that when using the time-history analysis, it relies on the modal superposition technique method. The staff notes that the modal superposition technique for time-history analysis is used for linear elastic dynamic analysis and is acceptable to the staff because the SRP acceptance criteria primarily address linear elastic analysis.

The applicant stated that when it uses the modal superposition time-history method of analysis to determine piping dynamic response, it used the procedures for combining modal responses provided in RG 1.92, with guidance from SRP Section 3.7.2. The applicant also stated that direct integration time history analysis is used to analyze seismic loads on piping systems. The analysis is executed according to the methodology described in TR-121515-P. The staff reviewed TR-121515-P and documented its acceptance in Section 3.7 of this report.

Because the applicant used methods recommended in NRC guidance, as discussed above, the staff finds the applicant’s time-history method for piping analysis acceptable.

FSAR Sections 3.7.1 and 3.7.2 describe the NuScale seismic analysis methods in detail. Section 3.7 of this SER presents the complete staff evaluation of the time-history seismic analysis methods.

3.12.4.2.5 Inelastic Analyses Method

FSAR Section 3.12.3.6, “Inelastic Analyses Method,” states that inelastic analysis methods are not used for any NuScale piping system analysis. The applicant’s decision not to use inelastic analysis methods is consistent with SRP Section 3.12, Acceptance Criterion II.A.v, and therefore, is acceptable to the staff.

3.12.4.2.6 Equivalent Static Load Method

FSAR Section 3.12.3.6, “Equivalent Static Load Method,” states that the equivalent static load method of seismic analysis is not used for ASME Class 1, 2, and 3 piping. This is acceptable to the staff based on the guidance in SRP Section 3.12.

3.12.4.2.7 Non-seismic/Seismic Interaction

FSAR Section 3.12.3.7, “Non-seismic/Seismic Interaction (II/I),” shows that when isolation of Seismic Category I piping from piping that is not required to be designed to Seismic Category I requirements is not feasible or practical, adjacent non-Category I piping is classified by the applicant as Seismic Category II and is analyzed in accordance with the same seismic design criteria applicable to the Seismic Category I piping. The applicant also showed that when non-seismic piping is attached to seismic Category I piping, the non-seismic piping up to the first anchor is included in the math model to account for its dynamic effects on the Seismic Category I piping. This portion of the non-seismic piping is designed not to cause a failure of the Seismic Category I piping. The applicant’s provisions for considering non-seismic to seismic interaction for piping are consistent with the staff’s recommendations in SRP Section 3.9.2, Acceptance Criterion II.2.K, and therefore, the staff finds the applicant’s position acceptable.

3.12.4.2.8 Category I Buried Piping

FSAR Section 3.12.3.8, “Seismic Category I Buried Piping,” states that the NuScale design does not include seismic piping that is directly buried in soil. The applicant’s position on buried piping is acceptable to the staff since buried piping is not part of the NuScale design.

3.12.4.2.9 Conclusions on Piping Analysis Methods

Based on its review described above, the staff concludes that the structural evaluations of ASME Class 1, 2, and 3 piping systems that are important to safety are acceptable because they satisfy the requirements of GDC 2 by specifying appropriate analysis methods for designing piping to withstand seismic loads.

3.12.4.3 Piping Modeling Technique

3.12.4.3.1 Computer Codes

FSAR Section 3.12.4.1, “Computer Codes,” lists ANSYS and AUTOPIPE for computer programs used in the design of NuScale piping. The FSAR also introduces COL Item 3.12-1, which allows a COL applicant that references the NuScale Power Plant US460 standard design to use other programs in addition to ANSYS and AUTOPIPE if the COL applicant implements a benchmark program using the models for NuScale Power Plant standard design. FSAR Section 3.12.4.1 also states that ANSYS and AUTOPIPE have been verified and validated to NUREG/CR-1677, Volumes I and II.

The staff finds the applicant’s piping benchmark program acceptable because it conforms to SRP Section 3.12, Acceptance Criterion II.B.iii, and the acceptance criteria in SRP Section 3.9.1, Subsection II.2. In SER Section 3.9.1, the staff finds the use of ANSYS and AUTOPIPE acceptable.

3.12.4.3.2 Decoupling Criteria

In FSAR Section 3.12.4.4, “Decoupling Criteria,” the applicant showed that for ASME Class 1, 2, and 3 pipe stress analysis, branch lines smaller than the main run of pipe can be decoupled from the analysis of the main run pipe and analyzed separately. The applicant stated that decoupling is performed for branch lines for which the routing is unknown. The applicant used the Welding Research Council (WRC) Bulletin (BL) 300, “Technical Position on Damping and on Industry Practice,” issued December 1984, criterion for decoupling, which states that if the ratio of run to branch pipe moment of inertia is 25 to 1 or more, the branch pipe may be decoupled from the run pipe. The applicant also showed that it had applied the WRC BL 300 restrictions in using this decoupling criterion. The WRC BL 300 decoupling criteria are acceptable to the staff because they have been accepted by the NRC in past DCAs (see NUREG-1793) and are widely used in nuclear piping design analysis.

FSAR Section 3.12.4.4, also includes provisions after decoupling and conditions for decoupling. It states that stress intensification factors and stress indices associated with the connection of the smaller line are considered in the analysis of the larger pipe, and the analysis includes a lump mass at the branch connection equal to at least half the mass of the branch line from the decoupling point to the branch line nearest support. Also, when the decoupled branch line is analyzed, the branch connection is modeled as an anchor for the branch line with stress intensification factors and stress indices associated with the type of connection. Displacements from the run pipe, caused by applicable loading conditions (e.g., seismic and thermal), are also applied at this anchor for the branch pipe stress analysis. In addition, if the run pipe is demonstrated to be dynamically rigid, by showing that its fundamental frequency is above the cutoff frequency, the envelope of response spectra of the nearest supports on both the run pipe and the decoupled branch pipe is applied at the connection for the branch piping analysis. If the run pipe is not determined to be rigid, FSAR Section 3.12.4.4 shows that the seismic input for the decoupled branch line is obtained by the analysis of the larger run pipe. This accounts for the amplification of the larger run pipe in the analysis of the branch line. These provisions, which are in addition to the WRC BL 300 decoupling criteria, are acceptable to the staff because they adequately account for the effects of the branch pipe on the stress analysis of the main run pipe and vice versa.

FSAR Section 3.12.4.4, has a subsection titled, “Overlap Region Methodology.” When the piping analysis cannot contain a full anchor-to-anchor model, a structural piping overlap can be used to terminate a pipeline model without an anchor. NUREG/CR-1980 presents the NRC’s guidance for the structural overlap method. NUREG/CR-1980, Section 2, “Conclusions and Recommendations,” contains conditions and criteria for using the structural overlap method and specifically requires that there should be at least four rigid restraints in each of three mutually perpendicular directions in the overlap region (including the ends). For axial restraints only, this requirement may be relaxed to a single restraint in any straight segment.

FSAR Section 3.12.4.4 states that if it is not feasible to analyze a piping system as a single model, then a structural overlap model is used. The applicant showed that when the structural overlap methodology is applied, the conditions and criteria in Section 2 of NUREG/CR-1980 are satisfied, and it included the above NUREG/CR-1980 specific requirement. The applicant also stated that piping system analyses, which include the overlap region, are required to show acceptable results for the piping components and supports in the overlap region. The staff finds the applicant’s structural overlap methodology acceptable because it follows the recommendations of NUREG/CR-1980, Section 2.

3.12.4.3.3 Conclusions on Piping Modeling Technique

Based on the review described above, the staff concludes that the applicant has met the requirements of Appendix B to 10 CFR Part 50 for the validity of computer programs used for the piping analysis of safety-related piping systems. The staff also concludes that the applicant has met GDC 1 by submitting information that demonstrates the applicability of the design methods used for the piping design analysis of ASME BPV Code Class 1, 2, and 3 piping.

3.12.4.4 Piping Stress Analysis Criteria

3.12.4.4.1 Seismic Input

In FSAR Section 3.12.5.1, “Seismic Input Envelope vs. Site-Specific Spectra,” the applicant stated that the seismic analysis of piping is performed using both the CSDRS and CSDRS-HF. CSDRS-HF was developed to address the high-frequency, hard rock sites in the central and eastern United States. FSAR Section 3.7.1.1 discusses the development of the CSDRS and CSDRS-HF. The applicant described the development of floor response spectra for the NuScale design in FSAR Section 3.7.2.5, where it stated that development of ISRS follows guidance in RG 1.122. Section 3.7.2 of this SER documents the staff’s evaluation and acceptance of FSAR Sections 3.7.1 and 3.7.2.

3.12.4.4.2 Design Transients

FSAR Section 3.9.1, discusses design transients and operating condition level categories, as defined in ASME BPV Code, Section III. FSAR Table 3.9-1 lists the design transients by ASME service level and includes the number of events over the design life of the plant for each transient. The number of cycles for each design transient is based on a plant life of 60 years. The transients are defined for the design purposes of safety-related equipment and are intended to provide a bounding representation of the NPM operation. Section 3.9.1 of this SER documents the staff's evaluation of this information.

3.12.4.4.3 Loadings and Load Combinations

The loadings and load combinations presented in the application should be sufficiently defined to provide the basis for ASME BPV Code Class 1, 2, and 3 analysis of piping and pipe supports for all applicable conditions. The acceptability is based on comparisons with positions in Appendix A to SRP Section 3.9.3 and with appropriate standards acceptable to the staff. FSAR Section 3.12.5.3, "Loadings and Load Combinations," discusses the loads and load combinations used for the structural evaluation of ASME Class 1, 2, and 3 piping. In the "Load Combinations" portion of Section 3.12.5.3, the applicant showed that in evaluating pipe stresses for NuScale piping, it used ASME BPV Code, Section III, methodology and equations, which include evaluations for service levels A, B, C, and D, as well as testing. FSAR Table 3.12-1, "Required Load Combinations for Class 1 Piping," and Table 3.12-2, "Required Load Combinations for Class 2 and 3 Piping," tabulate this information for the referenced piping systems.

In the "Seismic" portion of FSAR Section 3.12.5.3, the applicant stated that the operating basis earthquake (OBE) is defined as one-third of the SSE. Due to the selection of the OBE as one-third of the SSE, the OBE effects are not included as design loads (as allowed by 10 CFR 50 Appendix S), but the OBE cyclic effects are included in fatigue evaluations of ASME Code Class 1 piping. This portion of FSAR is in accordance with 10 CFR Part 50, Appendix S, and therefore is acceptable to the staff. This portion of FSAR also shows that the OBE cycle effects are considered in the fatigue evaluation of Class 1 piping, which conforms to the positions stated in SRM-SECY-93-087 and guidance in SRP Section 3.7.3 and therefore is acceptable to the staff.

In FSAR Tables 3.12-1, 3.12-2, and 3.12-3, "Required Load Combinations for Class 1, 2, & 3 Supports," the applicant stated that dynamic loads other than HELBs and SSE loads, which should be combined by the SRSS method, are combined considering the time phasing of the events in accordance with NUREG-0484, Revision 1. The staff finds that the applicant's dynamic load combination method is acceptable because it is in accordance with the guidance found in SRP Section 3.9.3, which is designed to comply with GDC 4 and which states that the appropriate method for combining dynamic loads should be in accordance with NUREG-0484, Revision 1.

The staff reviewed the proposed loads, load combinations, and stress limits given in the FSAR sections and tables discussed above and concludes that appropriate combinations of operating design transients and accident loadings have been specified to provide a conservative design envelope for the design of piping systems. The staff finds that the load combinations and stress limits conform to the guidelines in SRP Section 3.9.3 and the Commission position in Item 9 of SRM-SECY-93-087 about the elimination of the OBE. Therefore, the staff finds that the load combinations for the NuScale piping design are acceptable.

The staff also compared the listed condition loadings, equations, and stress limits in FSAR Tables 3.12-1 and 3.12-2 with those of ASME BPV Code, Section III. The staff concluded that the applicant's position complies with the requirements of ASME BPV Code, Section III, as incorporated by reference in 10 CFR 50.55a, and thus is acceptable.

Based on the above review, the staff finds that the applicant has defined appropriate loads and load combinations for the stress analysis of piping.

3.12.4.4.4 Damping Values

FSAR Section 3.12.3.4, "Damping Values," states that a constant damping value of 4 percent is used in the SSE loads seismic analysis of the NuScale piping systems and that frequency-dependent damping is not used. This is acceptable to the staff because Table 3 in RG 1.61, Revision 1, specifies the value of 4-percent damping for SSE.

In FSAR Section 3.12.3.4, the applicant also showed that composite modal damping is used when the piping analysis includes modeling of pipe supports or other structural elements that have different damping values as recommended in RG 1.61. FSAR Section 3.12.3.4, states that if the analysis of a piping model includes other nonpiping components (such as supports or structural elements) that have different damping values per RG 1.61, then composite modal damping values are determined using Equation (1) or Equation (2) from SRP 3.7.2, Acceptance Criterion 13. The staff finds this acceptable because this approach is the same as those in SRP Section 3.7.2, Revision 4, Acceptance Criterion II.5.D.13. The applicant also stated that when this method is used, damping shall not exceed 20 percent. SRP Section 3.7.2 also states this limit, and therefore, it is acceptable.

Based on the review described above, the staff finds acceptable the applicant's position on damping values used in the piping analysis.

3.12.4.4.5 Combination of Modal Responses

FSAR Section 3.12.3.2, "Modal Response Spectrum Method," states that the combination of modal response components is treated differently in RG 1.92, Revision 3, depending on whether a given mode includes only periodic components, only rigid components, or both periodic and rigid components. Combining the periodic and rigid response components in accordance with procedures of RG 1.92 Revision 3 for all modes provides the total system response to the URS. Section 3.7.2.7 of this report documents the staff's evaluation of the applicant's combination of modal responses.

3.12.4.4.6 High-Frequency Modes

According to RG 1.92, Revision 3, the missing mass method for calculating the contribution of high-frequency modes (above ZPA) is acceptable for both response spectrum analysis and modal superposition time-history analysis.

In FSAR Section 3.12.3.2.6, "Residual Rigid Response," the applicant showed that the residual rigid response for response spectrum analysis is obtained using the missing mass method described in Regulatory Position C.1.4.1 of RG 1.92, Revision 3. It also showed that, alternatively, the static ZPA method in RG 1.92, Revision 3, can be used to include the contribution of high-frequency modes.

In FSAR Section 3.12.3.3, the applicant showed that when the time-history method is used for seismic analysis of NuScale piping, the modal superposition method is utilized for contribution of mass above the ZPA frequency, the missing mass method of Regulatory Position C.1.4.1 of RG 1.92, Revision 3, is used. The applicant also states that direct integration time history analysis is used to analyze seismic loads on piping systems. The staff finds this acceptable.

Based on the above, the staff finds that the applicant has properly accounted for the contribution of high-frequency modes in the seismic analysis of piping.

3.12.4.4.7 Fatigue Evaluation for ASME BPV Code Class 1 Piping

In FSAR Section 3.12.5.4, "Fatigue Evaluation of ASME Code Class 1 Piping," the applicant stated that ASME Class 1 piping is to be evaluated for the effects of fatigue resulting from thermal transients, hydraulic transients, and external (cyclic) loads such as earthquakes.

For seismic consideration in the fatigue evaluation of Class 1 piping, the applicant's method conforms to the positions stated in SRM-SECY-93-087 and guidance in SRP Section 3.7.3. With the elimination of the OBE (being one-third of the SSE or less), according to SECY-93-087, the requirement is to use two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range), which as the SECY states, is equivalent to the cyclic load basis of one SSE and five OBE events, as recommended in SRP Section 3.9.2, when accounting for differences in the structural damping between the OBE and SSE and for a 60-year (instead of a 40-year) plant life. SRP Section 3.7.3 states that, alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with IEEE Std. 344-2013, Annex D. Following the guidance from SRP Section 3.7.3 and using 20 full SSE vibratory cycles at one-third the maximum SSE amplitude, the applicant determined that 312 fractional amplitude SSE cycles are required. The staff verified that number by performing the following calculation using Annex D of IEEE Std. 344:

(PCE_{max}) = (No. of fractional cycles) × (Percentage of maximum peak cycles)$^{2.5}$

Given data: 20 full SSE vibratory cycles at 1/3 the maximum SSE amplitude, which translates to equivalent maximum peak cycles, $(PCE_{max}) = 20$; and

(Percentage of maximum peak cycles) = 1/3.

By substituting the given data into the above equation and solving for number of fractional cycles, n:

$n = 20/(1/3)^{2.5} = 311.8$ or 312 fractional amplitude SSE cycles.

The staff reviewed the applicant's position on use of seismic cycles for Class 1 piping fatigue evaluation, as shown above, and finds it acceptable as it is consistent with the staff's guidance in SRP Section 3.7.3, Revision 4.

In FSAR Section 3.12.5.4, the applicant showed that the fatigue evaluation of ASME Class 1 piping considers the effects of the reactor coolant environment and follows the guidance in RG 1.207. SRP Section 3.12, Acceptance Criterion II.C.xix, indicates that the guidance in RG 1.207 is an appropriate means of characterizing the effects of environment on fatigue design. Because the NuScale piping design addresses the effects of environment on fatigue life in conformance with the guidance in RG 1.207, the staff finds this acceptable.

Based on the above, the staff finds acceptable the applicant's methodology for the fatigue evaluation of Class 1 piping.

3.12.4.4.8 Fatigue Evaluation for ASME BPV Code Class 2 and 3 Piping

In FSAR Section 3.12.5.5, "Fatigue Evaluation of ASME Code Class 2 and 3 Piping," to account for fatigue in Class 2 and 3 piping, the applicant showed that it complies with the ASME BPV Code Class 2 and 3 piping fatigue requirements. Stress range reduction factors are applied to the allowable stress range resulting from thermal expansion if the piping is subjected to a total number of equivalent full-temperature cycles greater than 7,000 as provided in NC/ND-3611.2(e). The staff finds this acceptable because fatigue evaluation of NuScale ASME Class 2 and 3 piping meets the requirements of ASME BPV Code, as incorporated by reference in 10 CFR 50.55a.

3.12.4.4.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

Thermal stratification, cycling, and striping (TASCS) are thermal mechanisms that have caused significant damage to power plant pressure boundary components, most commonly, fatigue cracking of piping. NRC BL-88-08 requested licensees to identify and evaluate the piping systems connected to the RCS that were susceptible to TASCS to ensure that the piping will not be subjected to unacceptable thermal stresses. The bulletin recommended nondestructive examinations of potentially affected pipes to ensure that no flaws exist, as well as the development and implementation of a program to provide continuing assurance of piping integrity. Ways to provide this assurance include designing the system to withstand the cycles and stresses from valve leakage, instrumenting the piping to detect adverse temperature distributions and establishing appropriate limits, and providing a means to monitor pressure differentials that may indicate valve leakage.

While NuScale was not a recipient of this bulletin, the operating experience described in the bulletin should be incorporated in the design in accordance with 10 CFR 52.137(a)(22). SRP Section 3.12 includes criteria related to this bulletin, to the extent that the issue applies to a given design.

In FSAR Section 3.12.5.6, "Thermal Oscillations in Piping Connected to the Reactor Coolant System," the applicant stated that it used the screening criteria and evaluation methodology of EPRI TR-103581, "Thermal Stratification, Cycling, and Striping (TASCS)," issued July 1999, to assess unisolable piping connected to the RCS to identify TASCS in the NuScale design.

The applicant listed and screened the following lines that are connected to the NuScale RCS:

- the CVCS RCS discharge piping
- the CVCS RCS injection piping
- the PZR spray lines
- the RPV high-point degasification piping
- the ECCS hydraulic lines

Customarily, licensees of U.S. nuclear power plants use guidance found in the EPRI MRP-146, to address NRC BL-88-08. The EPRI TR-103581 TASCS report was an earlier EPRI report to

MRP-146 to assist utilities in addressing BL-88-08. As the applicant stated in FSAR Section 3.12.5.6, since the issuance of EPRI TR-103581, EPRI has issued MRP-146 with updated guidance for the assessment of TASCS addressed in BL-88-08, which has led to changes in the thermal oscillation and stratification screening criteria from what was documented in EPRI TR-103581. MRP-146 provides a model for predicting and evaluating thermal cycling for PWR stagnant lines, which has been shown by benchmarking results, using operating experience, to be effective in predicting the location of thermal cycling in a branch line attached to the RCS. EPRI has committed to keeping the guidance current through future MRP revisions based on owner operating experience (ML120120028).

MRP-146 is an EPRI proprietary document. The applicant cited and referenced three publicly available documents, which discuss updated MRP-146 screening criteria for TASCS. The applicant used these documents to determine screening criteria for the NuScale design, so that the assessment of whether a line is susceptible to thermal stratification or cycling is consistent with current industry practice. The applicant documented its TASCS screening for the NuScale lines connected to the RCS, as follows:

- The RCS discharge and injection lines are not stagnant during normal operation. According to MRP-146, these lines are screened out of further evaluation.

- The PZR spray lines are not stagnant during normal operation. FSAR Section 5.4.5, “Pressurizer,” shows that a reduced spray flow is continuously maintained during normal operation to minimize stresses on spray line components from thermal transients. Therefore, these lines are not stagnant and, according to MRP-146, are screened out of further evaluation.

- The RPV high-point degasification line is a vapor-filled, up-horizontal line with no potential for in-leakage. According to MRP-146, this line is screened out of further evaluation.

- The ECCS hydraulic lines are normally stagnant and have horizontal portions, but they are smaller than DN 25 (NPS 1). According to MRP-146, these lines are screened out of further evaluation.

The applicant concluded that the evaluated RCS connected lines satisfy the TASCS screening criteria and therefore do not require further evaluation.

Based on the review described above, the staff finds that the actions taken by the applicant addressed NRC BL-88-08 and the requirement in 10 CFR 52.137(a)(22) related to operating experience, because it used methodology and criteria consistent with industry practice found in EPRI MRP-146, which has been used in a previous DCA approved by the staff (see “Advanced Power Reactor 1400 (APR1400) Final Safety Evaluation Report,” dated March 28, 2018 (ML18087A364)).

3.12.4.4.10 Thermal Stratification

The phenomenon of thermal stratification can occur in long runs of horizontal piping when two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Under such stratified flow conditions, the top of the pipe may be at a much higher temperature than the bottom. This thermal gradient produces pipe deflections, support loads, pipe bending stresses, and local stresses. NRC BL-79-13, Revision 2, discusses the effects of thermal stratification in operating reactors in FW lines, and NRC BL-88-11 discusses these effects in PZR surge lines.

NRC BL-79-13 addresses the effect of thermal stratification that can lead to cracking of the FW line. Thermal stratification could occur in horizontal sections of piping when the incoming FW flow rate is low, and there is a large temperature difference between the incoming FW and the SG coolant, which results in a density difference.

FSAR Section 3.12.5.7.3, “Feedwater Line Stratification,” states that the FW line is designed to minimize adverse loading resulting from thermal stratification because the SG FW nozzle, located on the FW inlet plenum, and the adjacent FW line are either vertical or angled downward from the horizontal and therefore minimize the potential for thermal stratification.

NRC BL-88-11 and SRP Section 3.12 discuss the potential for stresses induced by thermal stratification in the PZR surge line. In particular, BL-88-11 requested that licensees at the time establish a program that would monitor the surge line for the effects of thermal stratification beginning with hot functional testing. In FSAR Section 3.12.5.7.1, “Pressurizer Surge Line Stratification,” the applicant noted that the NuScale power plant design does not have a PZR surge line. Thus, BL-88-11 is not applicable to the NuScale design.

3.12.4.4.11 Safety Relief Valve Design, Installation, and Testing

In FSAR Section 3.12.5.8, “Safety Relief Valve Design, Installation, and Testing,” the applicant stated that the design and installation of safety and relief valves for overpressure protection consider the recommendations in Appendix O to ASME BPV Code, Section III, Division 1. The applicant stated that the NuScale relief valves, which discharge into containment, are considered an open discharge system configuration. The applicant also discussed that dynamic structural analysis is performed for piping systems where the relief valve is discharging to closed systems and can also be performed for discharge to open systems or atmosphere. For open system discharge, in lieu of dynamic analysis, the applicant stated that a static analysis may be performed using a dynamic load factor (DLF). SRP Section 3.9.3 allows both these methods, and therefore, they are acceptable.

For application of the static method of analysis, the staff notes that ASME BPV Code, Section III, Appendix O, requires that the calculated reaction force and moments caused by discharge thrust be multiplied by the DLF, based on the relief/safety valve opening time and system dynamic characteristics. ASME BPV Code, Section III, Appendix O, does not provide the details of how to consider dynamic characteristics, while ASME BPV Code B31.1 provides additional instruction on how to perform these calculations. The ASME BPV Code B31.1, Nonmandatory Appendix II, takes a similar approach to calculating the reaction force resulting from discharge thrust. Use of ASME BPV Code B31.1, Nonmandatory Appendix II, is the standard industry approach, which the NRC has accepted for these calculations. Considering system dynamic characteristics, valve installation period, and the time a valve takes to operate from fully closed to fully open, Appendix II determines a DLF (minimum of 1.1 and maximum of 2.0) based on its Figure II-3-2, “Dynamic Load Factors for Open Discharge Systems,” which is in turn based on curves from *Introduction to Structural Dynamics*, by J.M. Briggs (McGraw-Hill Book Co., 1964).

When the static method of analysis is used, the applicant suggested using either a DLF of 2.0 or guidance from the ASME BPV Code B31.1, Appendix II, to calculate an appropriate load factor.

According to SRP Section 3.9.3, Subsection II.2, for pressure-relief device design and installation, the applicant should use the design criteria for pressure-relief installations specified in ASME BPV Code, Section III, Division 1, Appendix O. SRP Section 3.9.3, Acceptance Criterion II.2.C, also specifies that a maximum DLF of 2 may be used in lieu of a dynamic analysis to determine the DLF.

Based on the staff's review summarized above, the applicant's design for safety relief valve installation conforms to the staff's recommendation in SRP Section 3.9.3, and therefore, the staff finds the applicant's approach acceptable. Section 3.9.6 of this report documents the review of valve testing.

3.12.4.4.12 Functional Capability

The staff reviewed PVP-2002-1254, "Simulation of Excessive Deformation of Piping due to Seismic and Weight Loads". The paper indicated that the mode of failure was considered due to superposition of bending moment arising from vertical load produced by weight and horizontal seismic inertia force. To clarify the conditions leading to such failure, elastic-plastic analysis using the general FEM Code was carried out on the test#37 piping model. The analytical model and method were verified as effective means of study by comparison of analytical results with those obtained experimentally. In deadweight stress and excitation and its dominant frequency. The parameter determinations were made under condition of variation. The revised ASME new seismic stress criteria to prevent fatigue failure with ratcheting of piping components under seismic stress should also be shown effective for preventing excessive progressive deformation. The staff also observes that where the ASME BPV Code of record for a given plant is before the 1992 Edition with 1994 Addenda or after the 2004 Edition with 2005 Addenda, the Level D stress limits in the ASME BPV Code are considered sufficient to ensure piping functional capability consistent with NUREG-1367. Therefore, the applicant's use of the ASME BPV Code, 2017 Edition (no addenda), is in itself sufficient to address the primary concern related to this acceptance criterion in SRP Section 3.12. The applicant's reference to NUREG-1367, Section 9.1, which includes several other provisions to confirm functional capability, provides additional confidence that functional capability will be maintained.

3.12.4.4.13 Combination of Inertial and Seismic Anchor Motion Effects

FSAR Section 3.12.3.2.9 states that analyses of piping systems due to SAMs are performed statically. The system response due to inertia effects and due to anchor motions are combined by the absolute sum method for USM analysis, when combining of the results is necessary. This position is consistent with the guidance identified in NUREG-1061, Vol. 4, Therefore, the staff finds this method acceptable.

3.12.4.4.14 Operating-Basis Earthquake as a Design Load

FSAR Section 3.12.5.3 stated that because the OBE has been set as one-third of the SSE, the OBE is not considered as a design load for the NuScale plant. However, the fatigue evaluation of Class 1 piping did consider the cyclic effect of the OBE. The ASME Code's position needs to consider Levels A, and B service loads in the fatigue analysis. OBE is Level B service load. Therefore, the staff finds this position acceptable.

3.12.4.4.15 Welded Attachments

In some cases, welded pipe attachments are needed to transfer pipe loads to pipe supports for the structural qualification of the pipe pressure boundary in accordance with the ASME BPV Code. SRP Section 3.12 states that the applicant can use accepted Code Cases listed in RG 1.84.

FSAR Section 3.12.5.10, “Welded Attachments,” states that welded attachments for ASME Class 1, 2, and 3 piping are permitted provided the effects of the attachment on the piping are considered in accordance with ASME Code, Section III Nonmandatory Appendix Y.

Although the nonmandatory appendices to ASME BPV Code, Section III, are not incorporated by reference into 10 CFR 50.55a, the staff observes that the technical provisions for welded attachments of Appendix Y are the same as those in ASME BPV Code Cases N-122-2, N-318-5, N-391-2, and N-392-3. ASME annulled these Code Cases after Appendix Y was added, but they remain accepted by the staff without conditions in RG 1.84, which is currently incorporated by reference in 10 CFR 50.55a. The staff finds the use of ASME BPV Code, Section III, Appendix Y, for the evaluation of integral pipe welded attachments acceptable, given that this appendix provides industry-accepted guidance for ensuring the quality of these welded attachments and that the staff previously approved the technical content of this appendix in RG 1.84, which was incorporated into 10 CFR 50.55a.

3.12.4.4.16 Modal Damping for Composite Structures

FSAR Section 3.12.3.5 and Section 3.12.3.2.2 contain the applicant’s discussion and position on modal damping for composite structures. The staff’s review for its acceptance is documented in Sections 3.7.2 and 3.12.3.4 of this report.

3.12.4.4.17 Minimum Temperature for Thermal Analyses

According to SRP Section 3.12, Acceptance Criterion II.C.xvii, the stress-free reference temperature for a piping system is defined as a temperature of [21 degrees C] 70 degrees F. For piping systems that operate at temperatures above [21 degrees C] 70 degrees F, a thermal expansion analysis should be performed in accordance with ASME BPV Code, Section III. The SRP also states that if a higher stress-free reference temperature is selected, the applicant should justify the higher temperature. The NRC will review this justification on a case-by-case basis to confirm that the higher temperature is suitable for the piping configuration, design support loads, piping displacement, and other factors.

In FSAR Section 3.12.5.11, “Minimum Temperature for Thermal Analyses,” the applicant stated that ASME BPV Code, Section III, does not require thermal analysis for Class 2 and 3 piping if the operating temperature is 65 degrees C (150 degrees F) or less. FSAR Section 3.12.5.3 also identifies that 21 degrees C (70 degrees F) is the stress-free reference temperature for thermal analysis of piping systems.

The staff notes that ASME Code NC/ND-3652.2 provides requirements for thermal expansion analysis. However, NC/ND-3673.1 states that all systems shall be analyzed for adequate flexibility by a structural analysis unless the following conditions are met:

- the operating temperature of the piping system is at or below 150°F (65°C) and the piping is laid out with inherent flexibility as provided in [NC/ND]-3672.7.

Referenced Section NC/ND-3672.7 states:

Piping system shall be designed to have sufficient flexibility to prevent pipe movements from causing failure from overstress of the pipe material or anchors, leakage at joints, or detrimental distortion of connected equipment resulting from excessive thrusts and moments. Flexibility shall be provided by changes of direction in the piping through the use of bends, loops, or offsets; or provisions shall be made to absorb thermal movements by utilizing expansion, swivel, or ball joints, or corrugated pipe.

The applicant updated FSAR Section 3.12.5.11 to address the flexibility issue identified in NC/ND-3672.7. On the basis of the Code compliance, the staff finds this acceptable.

3.12.4.4.18 Intersystem Loss-of-Coolant Accident

According to SRP Section 3.12, Acceptance Criterion II.C.xvii, to the extent practicable, low-pressure systems should be designed to withstand full RCS pressure. Meeting this acceptance criterion provides assurance that overpressurization of low-pressure piping systems because of RCPB isolation failure will not result in rupture of the low-pressure piping.

In FSAR Section 3.12.5.12, "Intersystem Loss-of-Coolant Accident," the applicant stated that piping systems that normally operate at low pressure that interface with the RCS and are subjected to the full RCS pressure are designed for the design pressure of the RCS. This statement by the applicant is acceptable because it meets the SRP Section 3.12 acceptance criterion as it gives assurance that over-pressurization of low-pressure piping systems resulting from RCPB isolation failure will not cause failure of the low-pressure piping.

3.12.4.4.19 Effects of Environment on Fatigue Design

FSAR Section 3.12.5.13, "Effects of Environment on Fatigue Design," states that the fatigue evaluation of ASME Class 1 piping considers the effects of the reactor coolant environment and follows the guidance in RG 1.207. SRP Section 3.12, Acceptance Criterion II.C.xix, indicates that the guidance in RG 1.207 is an appropriate means of characterizing the effects of environment on fatigue design. Because the NuScale piping design addresses the effects of environment on fatigue life in conformance to the guidance in RG 1.207, the staff finds this acceptable.

3.12.4.4.20 Conclusions on Piping Stress Analysis Criteria

Based on the review described above, the staff concludes that, with regard to pipe stress analysis criteria in the NuScale SDAA, the applicant has followed NRC guidance provided in SRP Section 3.12 and other guidance listed in Section 3.12.3 of this SER to meet acceptance criteria that are based on the relevant requirements of the following Commission regulations:

- GDC 1 and 10 CFR 50.55a, with regard to piping systems being designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed and with appropriate quality control

- GDC 2 and 10 CFR Part 50, Appendix S, with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC 4, with regard to piping systems important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions

- GDC 14, with regard to the RCPB of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- GDC 15, with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded

3.12.4.5 Piping Support Design Criteria

3.12.4.5.1 Applicable Codes

In FSAR Section 3.12.6.1, "Applicable Codes," the applicant discussed the codes it used for the design of pipe supports. It indicated that the classification of pipe supports for ASME class piping is the same as that for piping. According to FSAR Section 3.12.6.1," for ASME classified supports and for seismic Category I supports, the applicant used ASME BPV Code, Section III, Subsection NF. FSAR Section 3.12.6.1 also shows that the additional stress limit criteria of RG 1.124, Revision 3, and RG 1.130, Revision 3, are met "[f]or Class 1 lineartype and plateandshelltype supports." Because the applicant's pipe support design codes for safety-related and Seismic Category I supports conform to SRP recommendations and regulatory guidance, the staff finds them acceptable.

Based on its review described above, the staff finds that the applicant has appropriately used applicable codes for pipe support design.

3.12.4.5.2 Jurisdictional Boundaries

SRP Section 3.12, Acceptance Criterion II.D.ii, states that the jurisdictional boundaries between pipe supports and interface attachment points should comply with ASME BPV Code, Section III, Subsection NF. Paragraph NF-1131 states that the jurisdictional boundary between components, including piping systems, and supports shall meet the requirements of NB-1132, NC-1132, ND-1132, or NE-1132, as applicable to the class of component.

According to FSAR Section 3.12.6.2, "Jurisdictional Boundaries," piping supports having welded attachments to the piping follow the jurisdictional boundary guidance in NB/NC/ND-1132 and therefore are acceptable.

FSAR Section 3.12.6.4, "Pipe Support Baseplate and Anchor Bolt Design," states that all Class 1 and 2 pipe supports are supported by the containment vessel (CNV). FSAR Section 3.12.6.2 shows that for pipe supports that attach to the surface of the CNV, the support boundary is at the surface of the CNV, and the weld is considered part of the CNV and conforms to the requirements of the CNV. This is acceptable because it is in accordance with NC-1132.2(b), which states that attachments, welds, and fasteners with a pressure-retaining function shall be considered part of the component.

According to FSAR Section 3.12.6.2, for ASME Class 1, 2, and 3 piping systems in the design, pipe supports are attached to the NPM and not a building structure, Class 1, 2, and 3 instrument line supports are installed in one or more of the following configurations: welded or bolted to a Subsection NF support structure, welded to a vessel, or attached to the reactor building structure. In this FSAR section, the applicant showed that the jurisdictional boundary requirements for these supports follow guidance in ASME BPV Code, Section III, Subsection NF, and therefore are acceptable.

Based on its review above, the staff finds the applicant's position on jurisdictional boundaries for pipe supports acceptable.

3.12.4.5.3 Loads and Load Combinations

SRP Section 3.9.3, Subsection II.1, provides acceptance criteria for component and component support design. This SRP section states that the design and service loading combinations should be sufficiently defined to provide the basis for the design of ASME Class 1, 2, and 3 components and component supports for all conditions. It also states that the acceptability of the combination of design and service loadings applicable to the design of ASME Class 1, 2, and 3 components and component supports is judged by comparison with positions stated in Appendix A to SRP Section 3.9.3.

The loads on pipe supports are reaction loads at support locations resulting from the piping stress analysis, which uses the loads and load combinations presented in FSAR Section 3.12.5.3. The staff's evaluation of Section 3.12.5.3 appears above in Section 3.12.4.4.3. FSAR Section 3.12.6.3, "Loads and Load Combinations," states that the pipe support load combinations are shown in Table 3.12-3. Nomenclature for the acronyms of the abbreviated loads in Table 3.12-3 can be found in FSAR Table 3.9-2. FSAR Section 3.12.5.3 presents a full description of the loads.

The staff reviewed the loads and load combinations listed by the applicant in FSAR Table 3.12-3, and finds them acceptable because they conform to the guidelines in Appendix A to SRP Section 3.9.3.

3.12.4.5.4 Pipe Support Baseplate and Anchor Bolt Design

In FSAR Section 3.12.6.4, the applicant stated that the NuScale design baseplates are not used for any Class 1, 2 or 3 pipe supports.

3.12.4.5.5 Use of Energy Absorber and Limit Stops

Because FSAR Section 3.12.6.5, "Use of Energy Absorbers and Limit Stops," shows that the NuScale ASME Class 1, 2, or 3 piping design does not use energy absorbers or limit stops, the staff finds it acceptable based on SRP Section 3.12, Acceptance Criterion II.D.v.

3.12.4.5.6 Use of Snubbers

Because FSAR Section 3.12.6.6, "Use of Snubbers," shows that the NuScale ASME Class 1, 2, or 3 piping design does not use snubbers, the staff finds it acceptable based on SRP Section 3.12, Acceptance Criterion II.D.vi.

3.12.4.5.7 Pipe Support Stiffness

In FSAR Section 3.12.6.7, "Pipe Support Stiffness," the applicant showed that either actual support stiffness is used in the piping analysis for all supports or all supports are modeled with rigid stiffness. The exception is that if variable spring supports are used, their actual stiffness is modeled in the piping analysis regardless of the method used for the remainder of the supports. The staff notes that, in general, rigid pipe supports are modeled in the piping analysis using a very high stiffness default in the analysis program. This is referred to as “rigid” stiffness. The applicant also showed that when the “rigid” stiffness is used, support deflection is checked to verify the rigidity. Each support modeled as rigid is checked with the deflection in the restrained directions to a maximum of 1.59 mm (1/16 in.) for SSE loadings and a maximum of 3.18 mm (1/8 in.) for other loadings. In addition, when evaluating pipe support deflections, any dynamic flexible elements of the attaching components or building structure are also considered.

The staff reviewed the applicant’s procedure for pipe support stiffness presented in FSAR Section 3.12.6.7 and found it acceptable because it is reasonable and consistent with industry practices documented in WRC BL 353, “Position Paper on Nuclear Plant Pipe Supports,” issued May 1990.

3.12.4.5.8 Seismic Self-Weight Excitation

In FSAR Section 3.12.6.8, the applicant showed that the pipe support seismic analysis included the effect of the SSE where the pipe support structure is considered as a self-weight excitation. Dynamic analysis is performed for the seismic inertial response of the support mass similar to that used in the piping dynamic seismic analysis, or alternatively, the equivalent static analysis procedure found in FSAR Section 3.7.3, is used to determine the support seismic response resulting from self-weight excitation. The staff reviewed the equivalent static load method in FSAR Section 3.7.3.1.2, compared it with the equivalent static load method of SRP Sections 3.9.2 and 3.7.2, and found it equivalent to the method in the SRP. The applicant showed that support self-weight SSE response, the piping inertial load SSE response, and loads from SAMs are combined by absolute summation, which is recommended in SRP Section 3.9.2. Damping values for welded and bolted structures are taken from RG 1.61.

Based on the review discussed above, the staff found the information in FSAR Section 3.12.6.8 acceptable because it is consistent with the SRP guidance.

3.12.4.5.9 Design of Supplementary Steel

In FSAR Section 3.12.6.9, “Design of Supplementary Steel,” the applicant stated that ASME Class 1, 2, and 3 pipe supports in the design are designed to Subsection NF of the ASME Code. Pipe support members used in connecting ASME Class 1, 2, and 3 piping to the supporting vessels, component supports, or RXB structures are within the jurisdictional boundaries of Subsection NF of the ASME Code (Section 3.12.6.2). Supplemental structural steel is not used in pipe support members. The staff finds this acceptable.

3.12.4.5.10 Consideration of Friction Forces

According to SRP Section 3.12, Acceptance Criterion II.D.x, the design of sliding type supports, such as guides or box supports, should include evaluation of the friction loads induced by the pipe on the support. Friction force on a pipe support is determined by the applied pipe force normal to the support member surface multiplied by an appropriate coefficient of friction (CoF).

In FSAR Section 3.12.6.10, “Consideration of Friction Forces,” the applicant presented its approach for the consideration of frictional forces on pipe supports. The applicant used a minimum CoF of 0.3. Its reference for this value is WRC BL 353. The staff notes that the 0.3

CoF for steel-to-steel is a reasonable value, which has been used in currently operating nuclear plants and which the staff has approved in DCAs (see NUREG-1966, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design," issued April 2014 (ML14099A519)). Therefore, the staff accepts the CoF value of 0.3 for steel-to-steel applications.

FSAR Section 3.12.6.10 also shows that friction forces on pipe supports are considered from all applicable loads, including deadweight/buoyancy loads, thermal expansion loads, loads from anchor or support movement (resulting from temperature or pressure), and other applicable signed loads, such as those from relief or safety valve discharge to an open system. In addition, FSAR Table 3.12-3 shows that friction forces are to be considered for all applicable loading conditions. The applicant's consideration of friction forces on pipe supports is acceptable to the staff because it provides assurance that all applicable loads will be considered in calculating the frictional load on pipe supports and frictional loads on supports will be considered in all applicable loading conditions.

3.12.4.5.11 Pipe Support Gaps and Clearances

According to SRP Section 3.12, Acceptance Criterion II.D.xi, pipe support gaps must account for the diametrical expansion of the pipe as the result of pressure and temperature.

FSAR Section 3.12.6.11, "Pipe Support Gaps and Clearances," specifies a nominal cold condition gap of 1.59 mm (1/16 in.) radially for rigid-guide-type pipe supports. It also states that deadweight pipe supports are to be in contact with the pipe in the direction of gravity with a 3.18-mm (1/8-in.) gap above the pipe when providing vertical restraint (in that direction). To check and avoid pipe binding through a sliding-type support, the applicant provided an equation, which calculates the combined radial pipe growth resulting from temperature and pressure. The staff reviewed the applicant's equation and finds it acceptable because it is derived using a standard engineering approach.

Based on the review discussed above, the staff finds the applicant's method for specifying pipe support gaps and clearances acceptable because the method follows the guidance in SRP Section 3.12, Acceptance Criterion II.D.xi.

3.12.4.5.12 Instrumentation Line Support Criteria

In FSAR Section 3.12.6.12, "Instrumentation Line Support Criteria," the applicant stated that the design loads, load combinations, and acceptance criteria for instrumentation line supports are similar to those used for pipe supports. Design loads include deadweight, thermal, and seismic loads. The staff noted that the use of pipe support design criteria for instrumentation line supports provides a conservative design and uses standards developed by professional societies, which are acceptable to the staff, as discussed in Section 3.12.4.5 above.

3.12.4.5.13 Pipe Deflection Limits

In FSAR Section 3.12.6.13, "Pipe Deflection Limit," the applicant stated that where standard piping supports or standard piping support parts are used, the manufacturer's recommended deflection limits are followed. In the NuScale Power design, spring supports are not used for ASME Code Class 1, 2, and 3 piping. Where rods or struts are used, an installation tolerance of 1 degree is applied to the manufacturer's swing angle limit. Correspondingly, the installation tolerances of these types of supports are 1 degree.

The staff reviewed the applicant's approach to deflection limits and finds it acceptable because the use of manufacturers' recommendations to limit pipe deflection provides confidence that pipe deflection will not cause the failure of the supports or cause an unanalyzed condition in the piping stress analysis. Also, the installation tolerance is acceptable, because it increases confidence that the component movement will remain within intended design limits of the component supports, thus ensuring the functionality of supports.

3.12.4.5.14 Clamp-Induced Local Pipe Stress Evaluation

FSAR Section 3.12.6.11 also states that the NuScale Power Plant does not use any specialized stiff pipe clamps that would induce high local stresses on the pipe, as discussed in NRC Information Notice 83-80, "Use of Specialized 'Stiff' Pipe Clamps," dated November 23, 1983. The staff finds this acceptable based on SRP Section 3.12, Acceptance Criterion II.D.xiv.

3.12.4.5.15 Conclusions on Piping Support Design Criteria

Based on the review above, the staff concludes that, with regard to pipe support design criteria in the NuScale SDAA, the applicant has followed NRC guidance in SRP Section 3.12 and other guidance listed in SER Section 3.12.3 to meet acceptance criteria based on the relevant requirements of the following Commission regulations:

- GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with these requirements and general engineering practice

- GDC 2 and GDC 4 by designing and constructing the safety-related pipe supports to withstand the effects of normal operation, as well as postulated accidents such as LOCAs and the effects of the SSE

- GDC 14 by following the ASME BPV Code requirements that the RCPB of the primary piping systems be designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture

- GDC 15 by following the ASME BPV Code requirements that the reactor coolant piping systems be designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded

- 10 CFR Part 50, Appendix S, by providing reasonable assurance that the safety-related piping systems are designed to withstand the effects of earthquakes with an appropriate combination of other loads of normal operation and postulated accidents with an adequate margin for ensuring their safety functions

3.12.5 Combined License Information Items

Table 3.12-1 lists COL information item numbers and descriptions related to ASME BPV Code Class 1, 2, and 3 piping systems and associated supports design, from FSAR Section 3.12.

Table 3.12-1: NuScale COL Information Items for Section 3.12

Item No.	Description	FSAR Section
COL Item 3.12-1	An applicant that references the NuScale Power Plant US460 standard design may use a piping analysis program other than the programs listed in Section 3.12.4; however, the applicant will implement a benchmark program using the models for the NuScale Power Plant US460 standard design.	3.12.4.3
COL Item 3.12-2	An applicant that references the NuScale Power Plant US460 standard design will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. An applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.	3.12.5.1

FSAR Section 3.12 also mentions that if a COL applicant referencing the NuScale Power Plant US460 standard design (SDA) finds it necessary to route Class 1, 2, and 3 piping not included in the SDA so that it is exposed to wind, hurricanes, or tornadoes, the piping must be designed to the plant design-basis loads for these events.

The staff finds these COL information items acceptable because they adequately describe actions necessary for the COL applicant.

3.12.6 Conclusion

Based on its review of the information in FSAR Section 3.12, the staff concludes, for the reasons given above, that the applicant has established an acceptable basis for the structural integrity and functional capability of the NuScale ASME Class 1, 2, and 3 piping and its supports. Based on the above, the staff further concludes that the applicant has provided reasonable assurance that safety-related piping and its supports are structurally adequate to perform their intended design function and comply with 10 CFR 50.55a; 10 CFR 52.137(a)(22); 10 CFR Part 50, Appendices B and S; and GDC 1, 2, 4, 14, and 15.

3.13 Threaded Fasteners—ASME BPV Code Class 1, 2, and 3

3.13.1 Introduction

The applicant submitted the information in FSAR Section 3.13, “Threaded Fasteners (ASME BPV Code Class 1, 2, and 3),” to address the application of ASME BPV Code, Section III, Division 1, Class 1, 2, and 3 pressure-retaining bolts, studs, nuts, and washers (collectively referred to as threaded fasteners).

The staff evaluation considered the materials selection, mechanical testing, special processes and controls, fracture toughness requirements for ferritic materials, fabrication inspection, quality records, and preservice and inservice inspection requirements.

3.13.2 Summary of Application

FSAR: The applicant described the use of threaded fasteners associated with Class 1, 2, and 3 pressure-retaining joints in FSAR Section 3.13, which is summarized in the following discussion. Other sections of the FSAR also discuss the use of threaded fasteners.

As described in FSAR Section 1.9, "Conformance with Regulatory Criteria," and Section 3.13, the NuScale design conforms or partially conforms to the guidance in the following RGs:

- RG 1.28, Revision 4, "Quality Assurance Program Criteria (Design and Construction)," issued June 2010 (ML100160003)
- RG 1.65, Revision 1, "Materials and Inspections for Reactor Vessel Closure Studs," issued April 2010 (ML092050716)
- RG 1.84, Revision 36 (ML13339A515)

The applicant stated that no ASME BPV Code Cases were used for the design of the Class 1, 2, and 3 threaded fasteners. This is consistent with FSAR Table 5.2-1, "American Society of Mechanical Engineers Code Cases."

FSAR Table 1.9-3, states that the SDAA conforms to the acceptance criteria in SRP Section 3.13.

3.13.2.1 Design Considerations

FSAR Section 3.13.1, "Design Considerations," states that pressure boundary threaded fasteners comply with ASME BPV Code Class 1, 2, and 3 requirements.

FSAR Section 5.2.3.6, "Threaded Fasteners," discusses the RCPB threaded fasteners and cites Section 3.13. FSAR Section 6.2.6.2, "Containment Penetration Leakage Rate Test," states that all CNV bolted closures have dual O-ring seals and a testing port between the seals. FSAR Section 6.2.4.2.1 also discusses the CNV flange connections.

FSAR Section 5.4.1.5, "Steam Generator Materials" discusses the pressure-retaining components that are part of the SGs, including bolting material, which are listed in FSAR Table 5.4-3, "Steam Generator Piping, Piping Supports, and Flow Restrictor Materials."

FSAR Section 5.3.1.7, "Reactor Vessel Fasteners," states where threaded inserts are used on the RPV, as well as their design requirements. FSAR Table 5.2-3, "Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances," provides allowed materials for threaded inserts. The seal weld for all ASME Class 1 threaded inserts requires a fabrication examination using magnetic particles or liquid penetrant in accordance with the ASME BPV Code, Section III, Division 1, paragraph NB5271.

FSAR Sections 5.3.1.7 and 6.2.1.1.2 describe the function of lock plates. Lock plates are held in place by studs that are stud welded onto the RPV and directly threaded into the CNV upper flange. The stud welds have a fabrication liquid penetrant exam and no PSI or ISI exams.

3.13.2.2 Materials Selection

FSAR Section 3.13.1.1, "Materials Selection," states that the materials selected for the threaded fasteners meet the requirements of the ASME BPV Code, Section II and Section III, and are selected in accordance with FSAR Table 3.13-1, "ASME BPV Code Section III Criteria for Selection and Testing of Bolted Materials." FSAR Table 3.13-1 is based on SRP Table 3.13-1. FSAR Table 3.13-1 also references the ASME BPV Code sections related to material test coupons, fracture toughness requirements, examination criteria, and certified material test reports (CMTRs).

The materials for the threaded fasteners are selected based on the environmental conditions for the lifetime of the plant. Furthermore, the materials are chosen to avoid galvanic corrosion and stress corrosion cracking (SCC). FSAR Table 5.2-3, "Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances," provides allowed materials to be used for reactor pressure boundary threaded fasteners. FSAR Table 6.2-1, "Material Specifications for ESF Components," provides allowed materials to be used for threaded fasteners associated with the containment vessel. FSAR Table 6.1-4, "Pressure Retaining Materials for Reactor Coolant Pressure Boundary and Engineered Safety Feature Valves," provides allowed materials to be used for threaded fasteners associated with engineered safety feature valves.

FSAR Section 3.13.1.1 states the reactor coolant and containment vessel closure studs and nuts use SB-637 UNS N-7718 (Alloy 718), FSAR Section 5.2.3.6 states 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 Alloy 718 was selected because of its resistance to general corrosion and SCC. To improve the Alloy 718 resistance to SCC, all ASME BPV Code Class 1, 2, and 3 Alloy 718 threaded fasteners receive a final solution anneal as described in FSAR Section 3.13.1.1. This is in accordance with Section II of the ASME BPV Code but has a more restrictive solution temperature range of 982 to 1,010 degrees C (1,800 to 1,850 degrees F) before the precipitationhardening heat treatment. FSAR Sections 3.13.1.1 and 5.3.1.7 discuss the applicability of RG 1.65 to the Alloy 718 RPV main flange threaded fasteners. The applicant stated that RG 1.65, Regulatory Position 2(b), does not apply because of Alloy 718's resistance to general corrosion, which is different from traditionally used low-alloy steel RPV main flange threaded fasteners. Additionally, since Alloy 718 is an austenitic, precipitation-hardened, nickel-base alloy, the fracture toughness requirements of the ASME BPV Code and 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," do not apply. Finally, the applicant stated that since the fracture toughness requirements do not apply, the concern in RG 1.65, Regulatory Position 1(a)(i), is not applicable as it is related to the maximum allowable yield strength.

The applicant stated that the design for threaded fasteners meets the cleaning criteria of ASME NQA-1-2015 Part II, Subpart 2.1. The applicant also stated that lubricants will be selected in accordance with the guidance in NUREG1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," issued June 1990 (ML031430208), and lubricants containing molybdenum sulfide are prohibited.

3.13.2.3 Fracture Toughness Requirements for Threaded Fasteners Made from Ferritic Materials

The applicant stated that pressure-retaining Class 1, 2, and 3 components that are made of ferritic material meet the requirements of the ASME BPV Code. Pressure-retaining Class 1, 2, and 3 components that are part of the RCPB also meet the requirements of 10 CFR Part 50, Appendix G.

3.13.2.4 Preservice Inspection Requirements

The applicant stated that the PSI requirements are in accordance with the ASME BPV Code, Section XI.

3.13.2.5 Certified Material Test Reports (QA Records)

The applicant stated that all pressure-retaining ASME BPV Code Class 1, 2, and 3 threaded fasteners are certified in accordance with the ASME BPV Code, Section III, paragraphs NCA-3861 and NCA3862. Additionally, the applicant stated- that the pressure -retaining threaded fasteners are furnished with CMTRs and have material identification in accordance with the ASME BPV Code, Section III. Finally, CMTRs will be retained in accordance with 10 CFR 50.71, “Maintenance of Records, Making of Reports.”

3.13.2.6 Inservice Inspection Requirements

FSAR Section 5.2.4.1 describes the process for assessing inspection and testing of the ASME BPV Code Class 1 components except for SG tubes. FSAR Sections 5.3.1.7 and 6.2.1.1.2 state that the lock plate stud weld to the cladding undergoes a fabrication liquid penetrant inspection, and there is no PSI or ISI for the lock plate.

ITAAC: The ITAAC associated with FSAR Section 3.13, appear in FSAR Section 2.1. SER Section 14.3 discusses the NuScale ITAAC.

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

Technical Reports: The staff reviewed the following TRs, which are incorporated by reference in accordance with FSAR Section 1.6 and Table 1.6-2:

- TR-1116-51962-NP, Revision 1, “NuScale Containment Leakage Integrity Assurance Technical Report,” issued May 28, 2019 (ML19149A298)
- TR-0917-56119-NP, Revision 1, “CNV Ultimate Pressure Integrity,” issued June 2019 (ML19158A382)

3.13.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, GDC 1 and 10 CFR 50.55a require, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- 10 CFR Part 50, Appendix A, GDC 4, requires, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.
- 10 CFR Part 50, Appendix A, GDC 14, requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- 10 CFR Part 50, Appendix A, GDC 30, requires, in part, that components that are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical.

- 10 CFR Part 50, Appendix A, GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” requires, in part, that the RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

- 10 CFR Part 50, Appendix B, Criterion XIII, “Handling, Storage and Shipping,” requires that measures be established to control the handling, storage, shipping, cleaning, and preservation of materials and equipment to prevent damage or deterioration.

- 10 CFR Part 50, Appendix G, specifies fracture toughness requirements for ferritic materials of pressure retaining components of the RCPB to provide adequate margins of safety during any condition of normal operation, including AOOs and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The guidance in SRP Section 3.13, Revision 0, “Threaded Fasteners—ASME BPV Code Class 1, 2, and 3,” issued March 2007, lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

3.13.4 Technical Evaluation

SRP Section 3.13, Table 3.13-1, “ASME BPV Code Section III Criteria for Selection and Testing of Bolted Materials,” and Table 3.13-2, “ASME BPV Code Section XI Examination Categories for Inservice Inspections of Mechanical Joints in ASME BPV Code Class 1, 2, and 3 Systems that Are Secured by Threaded Fasteners,” reference the 2001 Edition of Section III and Section XI of the ASME BPV Code. The staff reviewed the table in FSAR Section 3.13 and verified that they cite the appropriate portions of the 2017 Edition of Section III and Section XI of the ASME BPV Code. The staff recognizes that the citations to the ASME BPV Code may differ if NuScale were to use a future edition of the ASME BPV Code. The staff’s review also focused on the applicability of the ASME BPV Code requirements related to threaded fasteners to the NuScale design.

The staff created Table 3.13-1, shown below, to list the specific threaded fasteners used in the NuScale US460 design. The staff reviewed these fasteners as part of its evaluation of FSAR Section 3.13.

Table 3.13-1: List of Threaded Fasteners Used in the NuScale Design Reviewed in Section 3.13 of the SER

Location	Component	Material	Penetration	FSAR Section(s)
CNV Flange	Main Flange Closure Studs	Alloy 718	N/A	6.1.1.1 Table 6.1-2
RPV Flange	Main Flange Closure Studs	Alloy 718	N/A	5.2.3.6 Table 5.2-3 5.3.1.7

Location	Component	Material	Penetration	FSAR Section(s)
RPV Head	Reactor Safety Valve	SA-193 Grade B8 Class 1 or Grade 8M Class 1, Alloy 718	RPV 18–19	5.1.3.5 5.2.2.4.1 5.2.5.3 Table 6.1-3
RPV Head	I+C Channels A–D	SA-193 Grade B8 Class 1 or Grade 8M Class 1, Alloy 718	RPV 39–42	Table 5.2-3
RPV Head	RVV Flange	SA-193 Grade B8 Class 1 or Grade 8M Class 1, Alloy 718	N/A	5.2.2 5.2.2.4.2 Table 6.1-3
RPV Head	CRD Pressure Housing	Alloy 718	RPV 23-28	4.5.1
Upper RPV Section	RRV Flange	SA-193 Grade B8 Class 1 or Grade 8M Class 1, Alloy 718	N/A	5.1.3.6 5.2.2.5 Table 5.103
Upper RPV Section	FW Access Port 1–4	SA-193 Grade B8 Class 1, SA-184 Grade 8, Alloy 718	RPV 43–46	5.4.1.2 5.4.1.5 Table 5.4-3
Upper RPV Section	Main Steam Access Port 1–4	SA-193 Grade B8 Class 1, SA-184 Grade 8, Alloy 718	RPV 47–50	5.4.1.2 5.4.1.5 Table 5.2-3 Table 5.4-3
Upper RPV Section	PZR Access Port 1–2	SA-193 Grade B8 Class 1, SA-184 Grade 8, Alloy 718	RPV 21–22	5.2.3.6 Table 5.2-3
Upper CNV	PZR Heater Access Port 1–2	Table 6.1-2	CNV 31–32	3.8.2.1.4 6.1.1.1 Table 6.1-2
Upper CNV	CNV Manway	SA-193 Grade B8 Class 1, SA-194 Grade 8, Grade 630 Condition H1100, Alloy 718	CNV 24-26	3.8.2.1.4 6.1.1.1 Table 6.1-2
Upper CNV	SG Access Port 1–4	SA-193 Grade B8 Class 1, SA-194 Grade 8, Grade 630 Condition H1100, Alloy 718	CNV 27–30	3.8.2.1.4 6.1.1.1 Table 6.1-2

Location	Component	Material	Penetration	FSAR Section(s)
CNV Top Head	CRDM Access Opening	Grade 630 Condition H1100	CNV 25	3.8.2.1.4 6.1.1.1 Table 6.1-2
CNV Top Head	Heater Power 1-2	Table 6.1-2	CNV 15–16	3.8.2.1.4 6.1.1.1 Table 6.1-2
CNV Top Head	CRDM Power	Table 6.1-2	CNV 37	3.8.2.1.4 6.1.1.1 Table 6.1-2
CNV Top Head	I+C Divisions 1–2	Table 6.1-2	CNV 8–9	3.8.2.1.4 6.1.1.1 Table 6.1-2
CNV Top Head	I+C Channels A–D	Table 6.1-2	CNV 17–20	3.8.2.1.4 6.1.1.1 Table 6.1-2
CNV Top Head	RPI Group 1–2	Table 6.1-2	CNV 38–39	3.8.2.1.4 6.1.1.1 Table 6.1-2

FSAR Table 6.1-4, "Pressure Retaining Materials for RCPB and ESF Valves," lists the acceptable materials for the RCPB and ESF valve threaded fasteners. These are not included in the above table.

The staff performed an audit (ML24211A089) related to the use of threaded inserts in the NuScale design. During the audit, the staff gathered more information about the expanded use of threaded inserts throughout the NuScale design. Staff closed out technical issues related to analysis of pull out and safety margin related to the use of threaded inserts as opposed to traditional threading.

3.13.4.1 Materials Selection

To meet the requirements of GDC 1 and 10 CFR 50.55a to ensure that plant SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, the applicant must identify codes and standards that are used. The materials specified for use in these systems must be selected in accordance with the applicable provisions of the ASME BPV Code, Section III, Division 1, or RG 1.84. The ASME BPV Code, Section III, references applicable portions of the ASME BPV Code, Section II.

To meet the GDC 14 and GDC 30 requirements that the RCPB be designed, fabricated, erected, and tested to assure an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, and be designed to the highest quality standards practical, designers must also follow the requirements in the ASME BPV Code.

FSAR Section 3.13 states that Class 1, 2, and 3 threaded fasteners are designed to the ASME BPV Code, Section III, Subsections NB, NC, and ND, respectively. FSAR Table 3.13-1 lists the applicable criteria used for the material selection, and the materials selected meet the requirements in the ASME BPV Code, Section II and Section III.

The materials selected for use for the threaded fasteners are (1) Alloy 718 and (2) SA-564, Grade 630, Condition H1100. The staff reviewed whether the two materials selected for use are permitted by the ASME BPV Code, Section II and Section III. The staff found that the material specifications selected by the applicant are permitted for bolting materials by the ASME BPV Code, Section II and Section III.

The staff finds that the two materials selected for the NuScale threaded fasteners satisfy the applicable requirements of the ASME BPV Code, Section II and Section III, and therefore satisfy GDC 1, GDC 14, GDC 30, and 10 CFR 50.55a.

3.13.4.2 Mechanical Testing, Special Process, and Controls

To meet the GDC 1 and 10 CFR 50.55a requirements that plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, the applicant must identify codes and standards that are used. Following the requirements of the ASME BPV Code, Section III, Division 1, or RG 1.84, will meet GDC 1 and 10 CFR 50.55a. The ASME BPV Code, Section III, references applicable portions of the ASME BPV Code, Section II.

To meet the GDC 14 and GDC 30 requirements that the RCPB be designed, fabricated, erected, and tested to assure an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, and be designed to the highest quality standards practical, designers must also follow the requirements in the ASME BPV Code and may follow the guidance in RG 1.65.

To meet the requirements of GDC 4, SSCs important to safety shall be designed to be compatible with the environmental conditions, including lubricants.

To meet the requirements of 10 CFR Part 50, Appendix B, Criterion XIII, the applicant shall establish measures to control cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or degradation.

SA-564, Grade 630, Condition H1100, threaded fasteners are used only on the CNV top head, which is not subject to borated water from the UHS during operation or refueling. Alloy 718 is used for threaded fasteners in other locations of the design.

The staff's review included the specific heat treatments proposed, as well as the threaded fastener sizes. Alloy 718 and SA-564, Grade 630, Condition H1100 are designed to resist SCC. Alloy 718 was selected for use for the RPV main flange threaded fasteners instead of low-alloy steel. The RPV main flange and other RPV threaded fasteners fabricated of Alloy 718 are in a vacuum during normal operation and exposed to borated water during refueling and accident conditions. The CNV main flange threaded fasteners are exposed to the UHS water during normal operation. The RPV and CNV main flange threaded fasteners are identical in design and application. All Alloy 718 threaded fasteners receive a heat treatment to provide better resistance to SCC. The staff reviewed the selected heat treatment and found that it is designed to provide greater SCC resistance. The staff found that the ASME BPV Code, Section II, permits Alloy 718 threaded fasteners only for use up to a maximum allowable diameter of 15 cm (6 in.), which is greater than the diameter of the RPV and CNV main flange closure studs. SA-564, Grade 630, threaded fasteners are heat treated to the H1100 condition in all applications. The staff reviewed the selection of heat treating to Condition H1100 and found that it provides greater resistance compared to other allowable heat treatments (e.g., Condition H900). Overall, the staff found that these materials, based on their heat treatments and operating environments, are generally resistant to SCC and acceptable for use. The staff also found that the proposed heat treatments and sizing are allowed in accordance with the ASME BPV Code, Section II and Section III.

The applicant discussed the applicability of RG 1.65 to the RPV main flange threaded fasteners. The purpose of RG 1.65 is to ensure the fracture toughness for high-strength, large-diameter bolting and prevent degradation due to corrosion. However, RG 1.65 focuses on low alloy steel. The applicant stated that since the RPV main flange threaded fasteners are resistant to general corrosion, the concerns in RG 1.65, Regulatory Position 2(b), related to protecting the threaded fasteners from general corrosion, do not apply. The staff agrees with this assessment, as the chosen material is resistant to the conditions in the operating environment. Furthermore, NuScale proposed water chemistry controls for the primary, secondary, and UHS. FSAR Sections 5.2.3.2.1, 9.1.3, and 10.3.5, respectively, discuss these water chemistry controls.

SRP Section 3.13 and RG 1.65 state that ferritic steel RPV threaded fasteners should be subject to the fracture toughness requirements in 10 CFR Part 50, Appendix G, and the ASME BPV Code. Since Alloy 718 is nonferrous, the fracture toughness requirements in the ASME BPV Code and 10 CFR Part 50, Appendix G, do not apply.

RG 1.65, Regulatory Position 1(a)(i), states that the maximum permitted yield strength for bolts is 1,034 MPa (150 ksi). This requirement is based on low alloy steel's susceptibility to SCC. The applicant stated that this position does not apply since Alloy 718 is nonferrous and not required to be impact tested. Since this requirement is for ferritic steels, the staff finds this exception acceptable.

While the CNV main flange threaded fasteners have the same design as the RPV main flange threaded fasteners, NuScale did not apply RG 1.65 to the CNV main flange threaded fasteners since the requirement for impact testing and fracture toughness requirements are for low-alloy steel and the fasteners are fabricated from Alloy 718. Since this requirement is for ferritic steels, and Alloy 718 is nonferrous, the staff finds this exception acceptable.

Threaded fasteners should be protected against the detrimental effects of lubricants and boric acid corrosion. The materials selected are generally resistant to boric acid corrosion. Therefore, the staff finds that the threaded fastener materials are acceptable.

The applicant stated that the design for threaded fasteners meets the cleaning criteria of ASME NQA-1-2015 Part II. FSAR Section 3.13.1.1 states that lubricants will be selected in accordance with guidance in NUREG-1339, which is consistent with RG 1.65. FSAR Section 3.13.1.2, "Special Materials Fabrication Processes and Controls," states that lubricants will also be selected to avoid galvanic corrosion and SCC. The staff finds these requirements acceptable because the lubricants are selected in accordance with the guidance in SRP Section 3.13. Based on conformance to ASME NQA-1-2015 Part II and NUREG-1339, the staff finds that controls imposed on threaded fasteners satisfy the requirements of 10 CFR Part 50, Appendix B, Criterion XIII, with respect to controls for cleaning of materials and components, and of GDC 4 concerning the compatibility of components with environmental conditions.

In FSAR Section 3.6.2.7, the applicant stated that each of two RVVs and each of two RRVs in the NuScale design are bolted directly to reactor vessel nozzles. These four bolted-flange connections are also classified as break exclusion areas. The applicant provided its justification to ensure that the bolted connection provides confidence that the probability of gross rupture is extremely low and therefore may be classified as a break exclusion area. Specifically, the applicant stated that the components that comprise these bolted connections (valves, bolts, and nozzles) are classified as ASME BPV Code Class 1 components and are designed, fabricated, constructed, tested, and inspected in accordance with the ASME BPV Code, Section III, Subsection NB. The applicant also stated that the stress design criteria specified in ASME BPV Code, Section III, NB-3230, for the RVV and RRV bolt material provide more margin against yielding than do the rules of ASME BPV Code, Section III, NB-3653, for typical piping system materials and that this meets the intent of the guidance in BTP 3-4 for typical piping systems.

In addition, to support its use of a CUF of 1.0 for those bolted connections, the applicant stated that the fatigue evaluation for these bolts utilizes the fatigue curve from ASME Section III, Division 1, Mandatory Appendix I. Also, as required by NB-3230.3(c) for high-strength bolting, a fatigue strength reduction factor of no less than 4.0 is included in the fatigue evaluation for the NuScale RVV and RRV bolted connection. The applicant described phenomena (e.g., faulty design, improperly controlled fabrication and installation errors, unexpected modes of operation vibration, and other degradation mechanisms) that might adversely affect the fatigue evaluation for piping systems. The applicant explained why the NuScale RVV and RRV bolted connections are not susceptible to these types of phenomena. The applicant also stated that the RVVs and RRVs are within the scope of the NuScale CVAP. The CVAP ensures that the structural components of the NPM exposed to fluid flow are precluded from the detrimental effects of flow-induced vibration (FIV). The applicant discussed NuScale's comprehensive bolting integrity program, the sensitive leakage monitoring system, the augmented fabrication inspections, and the augmented 100-percent volumetric inservice examination requirements specified for the bolts of these flanged connections (ML24313A066). ISI requirements for bolting associated with the RPV are provided in FSAR ~~2~~ Section 5.3.3, "Reactor-Vessel Integrity." The applicant stated that the sensitive leakage monitoring system (being sensitive to a leak rate as low as 189 milliliters (0.05 gallon) per minute), along with the augmented inservice examinations, provides assurance that potential failure mechanisms are detected before the onset of a catastrophic failure of the bolted connections. The staff finds that the applicant's justification, including the conservatism included in the stress and fatigue design criteria for the bolted connection, the highly sensitive leakage monitoring system, as well as the augmented fabrication and inservice examination requirements specified for the bolts of these flanged connections, provide confidence to ensure that the probability of gross rupture at the bolted connection is extremely low and the bolted connection may be considered as a break exclusion area.

The staff requested stress and fatigue limits for representative bolted connection that includes threaded inserts. NuScale provided the information as an Engineering study ((ML24313A066)). The staff reviewed the summary of the information of the Engineering Study that provided an evaluation from DCA calculation for bolted connections using stress classification lines (SCLs). The Engineering Study from a previous Finite Element model in DCA calculation EC-A011-7207 extracted the stress and CUF information for the threaded inserts. The calculation for Final SDAA design will be a COL item to be provided by the COL applicant.

Based on the review of the preliminary information described in the above applicable SDAA sections and RAI response, the staff determines that the applicant's justification for break exclusion at the ECCS valve bolted connections is acceptable because it meets the intent of the BTP 3-4 staff's guideline for break exclusion areas. In particular, it provides reasonable assurance that the probability of gross rupture for the ECCS valves bolted connections is extremely low, and therefore, the bolted connections are considered as break exclusion areas.

3.13.4.3 Fracture Toughness Requirements for Ferritic Materials

To meet the GDC 31 requirement 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 and the probability of rapidly propagating fracture is minimized, the applicant should meet the requirements in the ASME BPV Code and 10 CFR Part 50, Appendix G.

The applicant stated that pressure-retaining Class 1, 2, and 3 components that are made of ferritic material meet the requirements of the ASME BPV Code, and components that are part of the RCPB must also meet the requirements of 10 CFR Part 50, Appendix G. The staff finds these requirements acceptable because the testing of the ferritic threaded fasteners is in accordance with the ASME BPV Code, Section III, and 10 CFR Part 50, Appendix G.

3.13.4.4 Fabrication Inspection

To meet the GDC 1 and 10 CFR 50.55a requirements that plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, the applicant must identify codes and standards that are used. The applicant must, at a minimum, follow the ASME BPV Code to meet these requirements.

To meet the GDC 14 and GDC 30 requirements that the RCPB be designed, fabricated, erected, and tested to assure an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, and be designed to the highest quality standards practical, designers must also follow the requirements in the ASME BPV Code.

For two unique portions of the NuScale design—the threaded inserts and lock plates—NuScale provided augmented fabrication inspections. The seal weld for all threaded inserts requires a fabrication examination using magnetic particle or liquid penetrant in accordance with the ASME BPV Code, Section III, Division 1, paragraph NB-5271. The lock plate stud welds are subject to a liquid penetrant fabrication inspection, which would detect unacceptable indications and ensure cladding integrity. The staff finds the augmented fabrication inspections for these components acceptable.

NuScale proposed augmented fabrication inspections for the RVV and RRV flange connection threaded fasteners as described in FSAR Table 3.13-1, Note 2. These augmented fabrication inspections are related to NuScale’s determination of where to postulate piping ruptures for the RVV and RRV flange connections in FSAR Section 3.6.2.5, “Analytical Methods to Define Forcing Functions and Response Models.” The staff finds these augmented inspections, which go beyond the ASME BPV Code requirements, acceptable as discussed in Section 3.6.2 of this SER.

The applicant stated that fabrication and examination of threaded fasteners are done in accordance with the criteria in FSAR Table 3.13-1 for ASME BPV Code Class 1, 2 and 3 systems. Since FSAR Table 3.13-1 cites the applicable sections of the ASME BPV Code, and the applicant committed to following those portions of the code, the staff finds that the applicant meets the requirements of GDC 1 and 10 CFR 50.55a related to fabrication, design, and inspection.

3.13.4.5 Quality Records

To meet the GDC 1 and 10 CFR 50.55a requirements that plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, the applicant must identify codes and standards that are used and maintain records. The applicant can meet these requirements by following the ASME BPV Code and retaining records in accordance with 10 CFR 50.71.

For ASME BPV Code, Section III, Class 1, 2, and 3 threaded fasteners, CMTRs are part of the ASME BPV Code records that are provided when the parts are shipped and are one of the required records that are maintained at the site. The applicant stated that the fasteners will be furnished with CMTRs and material identification. Further, the applicant stated that CMTRs will be retained in accordance with 10 CFR 50.71. The staff finds the above to be acceptable because the applicant complies with GDC 1 and 10 CFR 50.55a as they relate to quality records because the applicant committed to retaining the CMTRs in accordance with 10 CFR 50.71 and following applicable portions of the ASME BPV code.

3.13.4.6 Preservice and Inservice Inspection Requirements

FSAR Sections 5.2.4, 6.2.1, and 6.6 present additional information on PSI and ISI.

The lock plate stud welds are not subject to PSI or ISI exams. Since the lock plates are not part of the RCPB, and a failure of a lock plate stud would not lead to a significant safety issue, the staff finds that not augmenting PSI or ISI is acceptable for the lock plate stud welds.

The RVV and RRV flange connection threaded fasteners are less than 5 cm (2 in.) in diameter. NuScale proposed augmented ISI for the RVV and RRV flange connection threaded fasteners as described in FSAR Table 5.2-6. These augmented fabrication inspections are related to NuScale's determination of where to postulate piping ruptures for the RVV and RRV flange connections in FSAR Section 3.6.2.5. The staff finds these augmented inspections acceptable as documented in Section 3.6.2 of this report.

Compliance with the requirements of the ASME BPV Code, Section XI, also satisfies the regulatory requirements of 10 CFR 50.55a. FSAR Section 3.13.1.4, Section 3.13.2, Table 3.13-1state that PSI and ISI of threaded fasteners are done in accordance with the ASME BPV Code, Section III and Section XI, respectively. Therefore, the staff finds these requirements acceptable because threaded fasteners must meet the requirements of the ASME BPV Code, Sections III and XI.

SRP Section 3.13 states that the applicant should comply with ASME BPV Code, Section XI, IWA-5000, and pressure testing removal of insulation. FSAR Section 6.2.2.2, "System Design," states that insulation is not used inside containment. Therefore, the staff finds that NuScale meets the requirements.

3.13.4.7 ITAAC

ITAAC related to ASME BPV Code Class 1, 2, and 3 threaded fasteners include the ITAAC that ensure that the ESF systems will conform to the ASME BPV Code, Section III, requirements, which include materials. The staff's review of the proposed ITAAC is documented in SER Section 14.3.

3.13.4.8 Technical Specifications

There are no GTS requirements associated with the ASME BPV Code Class 1, 2, and 3 threaded fasteners. Other sections of the SER discuss required TS for other Class 1, 2, and 3 components. The staff finds this acceptable for the ASME BPV Code Class 1, 2, and 3 threaded fasteners in accordance with 10 CFR 50.36, “Technical Specifications,” as their structural integrity is ensured by meeting the relevant requirements, such as the ASME BPV Code.

3.13.5 Combined License Information Items

Table 3.13-2 lists the COL information item number and description related to the threaded fasteners from FSAR Table 1.8-1.

Table 3.13-2: NuScale COL Information Item for Section 3.13

Item No.	Description	FSAR Section
COL Item 3.13-1	An applicant that references the NuScale Power Plant US460 standard design will provide an inservice inspection program for American Society of Mechanical Engineers Class 1, 2, and 3 threaded fasteners. The program will identify the applicable edition and addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.	3.13

The staff reviewed FSAR Section 3.13.2, which lists COL Item 3.13-1. The staff confirmed the consistency of the wording with FSAR Table 1.8-1. The staff finds the wording of the COL information item acceptable as it will ensure that a COL applicant will develop an ISI program for its threaded fasteners in accordance with the ASME BPV Code.

3.13.6 Conclusion

Based on its review of the information provided by NuScale, the staff concludes that the NuScale SDAA for the ASME BPV Code Class 1, 2, and 3 threaded fasteners is acceptable and meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 1, 4, 14, 30, and 31; 10 CFR Part 50, Appendix B, Criterion XIII; and 10 CFR Part 50, Appendix G.