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 (hereinafter referred to as the staff) review of Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the NuScale Power, LLC (hereinafter referred to as the applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report." The staff's regulatory findings documented in this report are based on Revision 5 of the DCA, dated July 29, 2020 (Agencywide Document Access and Management System (ADAMS), Accession No. ML20225A071).

The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the DCA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this safety evaluation 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 DCA and not converted.

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

DCA Part 2, Tier 2, 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
1.14 6.4.4 15.0.3.4	PDC 19—A control room shall be provided from which actions can be taken to operate the nuclear power unit 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 (TEDE) 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 reactor, including necessary instrumentation and controls to maintain the unit in a safe-shutdown condition.
3.9.4 4.2.4 4.3.4.3 4.3.6 4.6.4 15.0.6.4.1	PDC 27—The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained. Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted.
5.4.4.3 5.4.4.6 15.0.5.3	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 assure that the system safety function can be accomplished, assuming a single failure.
4.2.4 6.1.1.4.4 6.2.1.1 6.3.4.1.8 15.0.3.5.3	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 assure that the system safety function can be accomplished, assuming a single failure.
6.2.1.1	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 assure that the system safety function can be accomplished, assuming a single failure.

FSER Section	Principal Design Criteria
6.1.1.4.4 6.2.5.4	PDC 41—Systems to control fission products, hydrogen, oxygen, and other substances which may be released 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 assure 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 assure that its safety function can be accomplished, assuming a single failure.
8.1.5 9.2.5	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 assure 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 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 DCA 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 2, “Unit Specific Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria,” addresses seismic classification.

DCA Part 2, Tier 2: To meet the NRC seismic requirements for the design for earthquakes, DCA Part 2, Tier 2, states that the seismic classification of SSCs is consistent with the guidance of RG 1.29. DCA Part 2, Tier 2, 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.” DCA Part 2, Tier 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.”

The DCA 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. DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structures, Systems, and Components,” identifies these seismic categories. Other sections of DCA Part 2, Tier 2, describe the various system safety functions and list simplified piping and instrumentation drawings (P&IDs).

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 requirements in 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 reactor coolant pressure boundary 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) 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

As described below, the staff reviewed DCA Part 2, Tier 2, Section 3.2.1, "Seismic Classification," and finds that the application appropriately classified components for the seismic design.

The column "Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)" lists "N/A" for the seismic classification of the water portion of the ultimate heat sink (UHS) (SSC Class A1) in DCA Part 2, Tier 2, Table 3.2-1. However, the staff noted that other UHS-related components are classified as seismic Category I. DCA Part 2, Tier 2, Table 3.2-1, shows that the internal reinforced concrete walls and floors of the reactor building (RXB) that form the UHS pool are part of "RXB, Reactor Building"; the pool liner for the UHS pool in the RXB is part of "RBCM, Reactor Building Components"; and the RXB, including the concrete that forms the UHS pool, and UHS pool liner are classified as seismic Category I and are required to withstand the SSE without loss of UHS pool water retention capability. The staff finds the seismic classifications of the UHS pool and the UHS-related components, which conform to RG 1.29, Staff Regulatory Position C.1, to be acceptable.

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. However, DCA, Part 2, Tier 2, Table 3.2-1, indicates that the portions of the RVIs are designated as seismic Category II. DCA Part 2, Tier 2, Section 3.2.2, "System Quality Group Classification," indicates that the design and construction code for the RVIs is the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME BPV Code), Section III, Division 1, Subsection NG. In DCA Part 2, Tier 2, Section 3.2.2 and Table 3.2-1 also identify that the steam generator (SG) tube supports are designed to seismic Category I and the requirements of ASME BPV Code, Section III, Division 1, Subsection NG. Although portions of the RVIs are not classified as seismic Category I, the staff determined that application of the ASME BPV Code, Section III, Division 1, Subsection NG, provides an equivalent level of safety for these SSCs. Based on this review, the staff finds that the classification of RVIs either conforms to or provides an equivalent level of safety as RG 1.29, Revision 5, Staff Regulatory Guidance C.1.b, and concludes that the classification of RVIs is 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. The applicant identified the "AQ-S" QA program applicability category for SSCs that are not safety related but are classified as either seismic Category I or seismic Category II. DCA Part 2, Tier 2, Table 3.2-1, Note 2, states that the pertinent requirements of 10 CFR Part 50, Appendix B, are applicable to seismic Category I or seismic Category II SSCs that are identified as "AQ-S," in accordance with the Quality Assurance Program (QAP). The staff reviewed DCA Part 2, Tier 2, Table 3.2-1, and determined that all SSCs that are not safety related but are designated as seismic Category I or II, have a QAP applicability of "AQ-S." Therefore, the staff finds that Note 2 of DCA Part 2, Tier 2, Table 3.2-1, conforms to RG 1.29, Staff Regulatory Guidance C.3, and concludes that the classification of "AQ-S" is 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. DCA Part 2, Tier 2, Table 3.2-1, includes the following:

> Note 5: Where SSC (or portions thereof) as determined in the as-built plant which are identified as Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II consistent with Section 3.2.1.2, ["Seismic Category II,"] and analyzed as described in DCA Part 2, Tier 2, Section 3.7.3.8 ["Interaction of Non-Seismic Category I Subsystems with Seismic Category I SSC"].

The staff finds that Note 5 to Table 3.2-1 is acceptable because it conforms to RG 1.29. SER Section 3.7.3.4.8 describes in detail the additional safety evaluations of the effects of SSCs that are not classified as seismic Category I on seismic Category I structures.

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 (ADAMS Accession No. 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 (ADAMS Accession No. 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, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," issued March 2007. The applicant provided the class break information and the details of the boundary of the designed SSC and piping classification that show application of the design requirements for ASME BPV Code Class 1, 2, and 3 components and piping, interface requirements, safety criteria, and the ASME BPV Code, Section XI, inspection program. The P&IDs in the DCA identify the interconnecting piping and valves and the interface between the safety-related and nonsafety-related portions of each system.

RG 1.29, Revision 5, Staff Regulatory Guidance C.2, states that the seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the nonseismic system or a sufficient distance into the nonseismic Category I system so that the seismic Category I analysis remains valid. In DCA Part 2, Tier 1 and Tier 2, the applicant provided DCA Part 2, Tier 1, Table 2.1-1, "NuScale Power Module Piping Systems," and Table 2.1-2, "NuScale Power Module Mechanical Equipment," and DCA Part 2, Tier 2, Figure 6.6-1, "ASME Class Boundaries for NuScale Power Module Piping Systems." The tables and figures included component/system class break information. The staff finds the tables and figures acceptable because the class break information conforms to RG 1.29.

DCA Part 2, Tier 2, Table 3.2-1, classifies the bioshields as not safety related, not risk significant, seismic Category II components; DCA Part 2, Tier 2, Table 3.2-1, identifies them as B2. The bioshield is designed to relieve the pressure associated with a high-energy fluid system break at the top of the NuScale Power Module (NPM); however, the bioshield vents have no mechanical devices, such as hinges. The ventilation is a path that is always open and relieves high-temperature and -pressure environments in the operating bay. The bioshield is classified as B2 (not safety related and nonrisk significant) seismic Category II. The staff finds the classification acceptable because the classification of bioshield conforms to RG 1.29. SER Sections 1.2, 3.7.2, 3.7.3, and 12.3 describe the additional safety evaluations of the bioshield vent design in detail.

RG 1.29, Revision 5, Staff Regulatory Guidance C.1.d and C.1.g, include guidance for the seismic classification of SSCs used to control the release of radioactive material. DCA Part 2, Tier 2, Section 3.2 states, in part, that the selection of augmented requirements for SSCs that are not safety related is based on a consideration of the important functionality to be performed by the SSC and regulatory guidance applicable to the functionality consistent with GDC 60 for controlling radioactive effluents. In addition, DCA Part 2, Tier 2, Section 3.2.1, states that seismic categorization of SSCs is also consistent with the guidance in RG 1.143, and Section 3.2.1.4 describes the seismic classification of SSCs that contain radioactive waste. Based on the staff's review of this information and the classifications provided in DCA Part 2, Tier 2, Table 3.2-1, the staff determined that the seismic classifications assigned by the applicant are consistent with GDC 60.

3.2.1.5 Combined License Information Items

DCA Part 2, Tier 2, Table 1.8-2, "Combined License Information Items," lists the combined license (COL) information item number and description related to Section 3.2.1.

Table 3.2.1-1: NuScale COL Information Item for Section 3.2.1

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.2-1	COL Item 3.2-1: A COL applicant that references the NuScale Power Plant design certification will update Table 3.2-1 to identify the classification of site-specific SSC.	3.2.1

3.2.1.6 Conclusion

Based on its review of DCA Part 2, Tier 1 and Tier 2, Section 3.2.1; the applicable P&IDs; and other supporting information in DCA Part 2, Tier 2, the staff concludes that the applicant's small modular reactor (SMR) design 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 DCA Part 2, Tier 2, includes an acceptable process to meet RG 1.29, Revision 5, 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.

3.2.2 System Quality Group Classification

3.2.2.1 Introduction

The staff reviewed DCA Part 2, Tier 2, 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 discussed in SER Section 3.2.1, DCA Part 2, Tier 2, Table 3.2-1, identifies the SSC classification, safety classification/quality group (QG) classification, and the QA requirements necessary to satisfy the requirements of GDC 1 for all safety-related SSCs and equipment. Applicable P&IDs identify the classification boundaries of interconnecting piping and valves. The staff reviewed DCA Part 2, Tier 2, Table 3.2-1, and the P&IDs in accordance with SRP Section 3.2.2. 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 DCA Part 2, Tier 2, Table 1.9-2, "Conformance with Regulatory Guides," the applicant stated that they conform to Revision 4 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

DCA Part 2, Tier 1: In DCA Part 2, Tier 1, Chapter 2, Table 2.1-1 and Table 2.1-2 provide SSC design description information.

DCA Part 2, Tier 2: In DCA Part 2, Tier 2, Section 3.2.2 and Table 3.2-1 classify the applicant's safety-related fluid systems and components as QG A, B, or C. Fluid systems and components that are not safety related that do not fall within QG A, B, or C also appear in Table 3.2-1 as QG D, WR-IIc, and WR-IIa. DCA Part 2, Tier 2, Table 3.2-1, identifies SSC classification as A1, A2, B1, or B2. A1 designates SSCs that are both safety related and risk significant, A2 designates SSCs that are safety related and not risk significant, B1 designates SSCs that are not safety related but are risk significant, and B2 indicates SSCs that are not safety related and not risk significant. Certain SSCs that perform risk-significant functions but are not safety related may require regulatory oversight. The regulatory treatment of nonsafety systems (RTNSS) process identifies the required oversight, as discussed in DCA Part 2, Tier 2, Section 19.3, "Regulatory Treatment of Non-Safety Systems." DCA Part 2, Tier 2, Table 3.2-1, also includes the basic commercial codes and standards applicable to major SSCs and the SSCs to which 10 CFR Part 50, Appendix B, applies. Safety-related SSCs and risk-significant SSCs are subject to the QAP requirements described in DCA Part 2, Tier 2, Section 17.5, "Quality Assurance Program Description," and are documented in the applicable QAP column of Table 3.2-1. In addition, all or part of 10 CFR Part 50, Appendix B, has been applied to some SSCs that are not safety related for which specific regulatory guidance applies (e.g., RG 1.29). DCA Part 2, Tier 2, Table 3.2-1, identifies the application of the pertinent requirements of 10 CFR Part 50, Appendix B, to these specific SSCs that are not safety related.

DCA Part 2, Tier 2, Section 3.2, states that the classification methodology includes consideration for "augmented" requirements for those SSCs that are, by definition, not safety related (based on 10 CFR 50.2, "Definitions"). The applicant stated that the selection of augmented requirements is based on a consideration of the important functionality to be performed by the SSCs that are not safety related and on regulatory guidance applicable to the functionality (e.g., consistent with the functionality specified in GDC 60 for controlling radioactive effluents, augmented requirements are specified for radioactive waste systems based on the guidance in RG 1.143). DCA Part 2, Tier 2, Table 3.2-1, identifies the augmented design requirements, if applicable.

DCA Part 2, Tier 2, Section 3.1.1.1, “Criterion 1—Quality Standards and Records,” states that the applicant’s plant design conforms to GDC 1.

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 2, issued March 2007, 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 DCA conforms to the requirements of QG classifications and quality standards used for design, the staff reviewed DCA Part 2, Tier 2, in accordance with SRP Section 3.2.2 and RG 1.26, Revision 4. The review included the evaluation of the criteria used to establish the QG classifications and the application of the criteria to the classification of principal components in DCA Part 2, Tier 2, Table 3.2-1.

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. DCA Part 2, Tier 2, Table 1.9-2, indicates that the applicant’s design certification (DC) conforms to RG 1.26, Revision 4; RG 1.143, Revision 2, issued November 2001; and RG 1.151, Revision 1, issued July 2010, as discussed in the following in the subsequent technical evaluation sections.

DCA Part 2, Tier 2, Table 3.2-1, indicates that no equipment is considered for the RTNSS. Because the applicant is relying only on the function of safety-related equipment, the staff finds that the applicant's assessment is appropriate and that the information in DCA Part 2, Tier 2, is consistent with this assessment.

The column "Augmented Design Requirements" in DCA Part 2, Tier 2, Table 3.2-1, specifies codes and standards for all components.

- Note 3 for the column "Augmented Design Requirements" of DCA Part 2, Tier 2, Table 3.2-1, clarifies that additional augmented design requirements are reflected in the columns "Quality Group/Safety Classification" and "Seismic Classification."

- Note 4 refers the reader to Sections 3.2.2.1 through 3.2.2.4 for the applicable codes and standards for each RG 1.26 QG, and DCA Part 2, Tier 2, Section 3.2.2.4, "Quality Group D," identifies the codes and standards for SSCs designated as QG D in accordance with RG 1.26.

- Note 4 refers to Section 3.2.1.4 for a description of RG 1.143 classifications for RW-IIa, RW-IIb, and RW-IIc.

- Note 2 of Table 3.2-1 identifies the applicability of augmented QA requirements for SSCs that are not safety related.

- In addition to the RG 1.26 QG, the column "Quality Group/Safety Classification" in DCA Part 2, Tier 2, Table 3.2-1, designates the applicable RG 1.143 safety classification. This designation is used in conjunction with more detailed information on applicable codes and standards from RG 1.143 in DCA Part 2, Tier 2, Table 11.2-10, "Codes and Standards from Regulatory Guide 1.143, Table 1"; Table 11.3-10, "Codes and Standards from Regulatory Guide 1.143, Table 1"; and Table 11.4-1, "List of Systems, Structures, and Components Design Parameters."

- DCA Part 2, Tier 2, Table 3.2-1, for the RVIs, lists SG supports separately as components of the reactor coolant system (RCS).

- DCA Part 2, Tier 2, Table 10.4-20, "Auxiliary Boiler System Component Design Parameters," identifies the section of the ASME BPV Code that applies to the fired power boiler.

The staff finds that the augmented design requirements discussed in the above paragraphs are acceptable because they conform to RG 1.26 and RG 1.143.

The column "Augmented Design Requirements" in DCA Part 2, Tier 2, Table 3.2-1, defines the treatment and design requirements of component supports.

- DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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.

- 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. DCA Part 2, Tier 2, Chapter 7, “Instrumentation and Controls,” and Chapter 8, “Electric Power,” provide further details on the codes and standards for instrumentation and electrical systems, respectively.

- In DCA Part 2, Tier 2, 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 staff finds that the design requirements discussed in the above paragraphs conform to RG 1.26 and are acceptable.

DCA Part 2, Tier 2, Section 5.2.4.1, “Inservice Inspection and Testing Program,” states that Section 3.2 and Section 5.2.1, “Compliance with Codes and Code Cases,” summarize the RCPB components subject to inspection. The staff finds that the fastener classification in DCA Part 2, Tier 2, Sections 5.2.1 and 5.2.4.1 and Table 3.2-1, is consistent with ASME BPV Code, Section III, and SRP Section 3.2.2 and is, therefore, acceptable.

DCA Part 2, Tier 2, Table 10.3-5, “Material Specifications and Corrosion Allowances,” provides material specification and grade along with corrosion allowances. DCA Part 2, Tier 2, Section 10.3.6.2, refers to DCA Part 2, Tier 2, Table 3.2-1, for the QG and seismic design classifications. The codes and standards for QG D SSCs are located in DCA Part 2, Tier 2, Section 3.2.2.4. The staff finds that the design provisions described above conform to RG 1.26 and are acceptable.

DCA Part 2, Tier 2, Table 3.2-1, provides the classification information on the piping systems of the steam generator system (SGS), decay heat removal system (DHRS), Auxiliary Boiler System (ABS), Condensate and feedwater system (CFWS), and UHS. The piping classification of SGS, DHRS, ABS, CFWS, and UHS system piping is the same as the classification for the components to which the piping is connected. The DHRS components are classified as for a safety-related system, as reflected in the QG and seismic Category I classifications, and the QG classification for these components is QG B, as identified in DCA Part 2, Tier 2, Section 5.4.3.1, “Design Basis,” and DCA Part 2, Tier 2, Table 3.2-1. The staff finds that the classification approach to piping components and DCA Part 2, Tier 2, Section 3.2.2, conform to RG 1.26 and are therefore acceptable.

3.2.2.5 Combined License Information Items

DCA Part 2, Tier 2, Table 1.8-2, lists the COL information item number and description related to Section 3.2.2.

Table 3.2.2-1: NuScale COL Information Item for Section 3.2.2

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.2-1	A COL applicant that references the NuScale Power Plant design certification will update Table 3.2-1 to identify the classification of site-specific SSC.	3.2.2

3.2.2.6 Conclusion

Based on its review of the applicable information in the DCA and on the above discussion, the staff concludes that the QG classifications of the pressure-retaining and nonpressure-retaining SSCs important to safety, as identified in DCA Part 2, Tier 2, Table 3.2-1, and the related P&IDs in the DCA, are in conformance to RG 1.26 and therefore are acceptable. DCA Part 2, Tier 2, Table 3.2-1, and the P&IDs identify major components in fluid systems (i.e., pressure vessels, heat exchangers, pumps, storage tanks, piping, valves, and applicable supports) and in mechanical systems (i.e., cranes, fuel-handling machines (FHMs), and other miscellaneous handling equipment). In addition, the P&IDs in the DCA identify the classification boundaries of interconnecting piping and valves. 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 satisfying 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 Severe Wind Loading

3.3.1.1 Introduction

The staff reviewed the applicant’s DCA Part 2, Tier 2, Section 3.3.1, “Severe Wind Loadings,” which addresses the design of structures that are required to withstand the effects of severe winds. The staff considered the information provided by the applicant in DCA Part 2 and the responses to the staff’s requests for additional information (RAIs) in establishing the reasonable assurance of safety conclusion.

3.3.1.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 5, Table 5.0-1, “Site Design Parameters,” provides the Tier 1 information associated with this section.

DCA Part 2, Tier 2: The applicant provided the design wind loading criteria for the NuScale plant in DCA Part 2, Tier 2, Section 3.3.1 and Table 2.0-1, “Site Design Parameters.” The applicant used the design windspeed, its recurrence interval, the speed variation with height, and the applicable gust factors as input parameters to establish the wind load to be used in the structural design. The applicant adopted the wind design parameters and the wind design procedure from the American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 7-05, “Minimum Design Loads for Buildings and Other Structures,” which is a reference of practice in wind design.

For the NuScale design, the applicant used a basic windspeed of 64.8 meters per second (m/s) (145 miles per hour (mph)), a 100-year return period, a 3-second gust at 10.1 meters (m) (33 feet (ft)) above ground with an exposure category C and a wind importance factor of 1.15 for the design of the RXB, control building (CRB), and radioactive waste building (RWB).

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 regulation during this review:

- GDC 2 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 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, 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.

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.

3.3.1.4 Technical Evaluation

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

The staff assessed and accepted an importance factor of the structures and an exposure category of the site 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 DCA Part 2, Tier 2, Section 3.3.1.2, "Determination of Severe 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 7-05 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.

3.3.1.5 Combined License Information Items

The COL applicant that refers to the NuScale DC will assess whether the actual site characteristics of severe wind are within the corresponding severe wind characteristics considered in the NuScale design. If the actual site characteristics of severe wind are not within corresponding severe wind characteristics considered in the NuScale design, the COL applicant should reevaluate the design of SSCs to the actual site-specific characteristic.

DCA Part 2, Tier 2, Table 1.8-2, lists the COL information item number and description related to Section 3.3.1.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.3-1	A COL applicant that references the NuScale Power Plant design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Reactor Building or Seismic Category I portion of the Control Building.	3.3.1 and 3.3.2

3.3.1.6 Conclusion

The staff concludes that the severe wind loadings used in the design of the SSCs for the NuScale application meet the guidelines of SRP Section 3.3.1 and therefore meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2. The staff has determined that the information in DCA Part 2 provides a reasonable assurance that SSCs important to safety will be designed to withstand the effects of severe winds.

3.3.2 Extreme Wind Loads (Tornado and Hurricane Loads)

3.3.2.1 Introduction

The staff reviewed DCA Part 2, Tier 2, Section 3.3.2 “Extreme Wind Loads (Tornado and Hurricane Loads),” which addresses the design of structures that are required to withstand the effects of the tornado and hurricane phenomena for the NuScale plant. The staff considered the information provided by the applicant in DCA Part 2 and the responses to the staff’s RAIs in establishing the reasonable assurance of safety conclusion.

3.3.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 5, Table 5.0-1, provides the Tier 1 information associated with this section.

DCA Part 2, Tier 2: The applicant provided the design parameters for the design-basis tornado and hurricane windspeed in DCA Part 2, Tier 2, Section 3.3.2 and Table 2.0-1. 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 design, the applicant used the maximum design-basis tornado windspeed of 103 m/s (230 mph) with the translational speed of 20.6 m/s (46 mph), the maximum rotational speed of 82.3 m/s (184 mph), the radius of maximum rotational speed of 46 m (150 ft), the pressure drop of 8.3 kilopascals (kPa) (1.2 pounds-force per square inch (psi)), and the rate of pressure drop 3.4 kPa (0.5 psi) per second for the design of the RXB, CRB, and RWB. In addition, the applicant used the maximum design-basis hurricane windspeed of 130 m/s (290 mph) for the design of these buildings.

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.

3.3.2.3 Regulatory Basis

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

- Under 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 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, 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.

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.

3.3.2.4 Technical Evaluation

SER Sections 2.3.1 and 2.3.2 document the staff's evaluations of the design-basis tornado parameters and the windspeed for the design-basis hurricane, respectively.

SER Section 3.3.1.4 documents the staff's evaluations of the wind importance factor, the exposure category, and the minimum value for the velocity pressure exposure coefficient.

In DCA Part 2, Tier 2, Section 3.3.2.3, "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.

SER Section 3.5.1.4, “Missiles Generated by Tornadoes and Extreme Winds,” documents the staff’s evaluations of the hurricane and tornado wind-generated missiles, respectively.

3.3.2.5 Combined License Information Items

The COL applicant that refers to the NuScale DC will assess whether the actual site characteristics of meteorological conditions of extreme wind (tornadoes and hurricanes) loads 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 extreme wind (tornadoes and hurricanes) loadings are not within corresponding site parameters of the NuScale design.

Table 3.3.2-1 lists the COL information item number and description related to the interaction of nonseismic Category I structures with seismic Category I structures from DCA Part 2, Tier 2, Section 3.3.2.

Table 3.3.2-1: NuScale COL Information Item for Section 3.3.2

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.3-1	A COL applicant that references the NuScale Power Plant design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the RXB or Seismic Category I portion of the CRB.	3.3.1 and 3.3.2

3.3.2.6 Conclusion

The staff concludes that the extreme wind loadings used in the design of the SSCs for the NuScale application meet the guidelines of SRP Section 3.3.2, because the applicant used the maximum tornado parameters including maximum windspeed, translational speed, rotational speed, and atmospheric pressure change, 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. Therefore, the staff finds that the NuScale application meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, because the information presented in DCA Part 2 provides a 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 DCA Part 2, Tier 1, Sections 3.11.1 and 3.13.1, both titled “Design Description,” and Tier 2, 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 includes 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 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

DCA Part 2, Tier 1: Internal flooding barriers provide confinements so that the impact from internal flooding is contained within the flooding area of origin in the RXB and CRB. In DCA Part 2, Tier 1, Sections 3.11.1 and 3.13.1 provide information related to the design of these buildings.

DCA Part 2, Tier 2: The applicant provided internal flooding analyses 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. DCA Part 2, Tier 2, Section 3.4.1, provides the information related to the flooding analyses.

ITAAC: The applicant provided the ITAAC associated with internal flooding barriers in the RXB and CRB in DCA Part 2, Tier 1, Table 3.11-2, "Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria," Item 2, and Table 3.13-1, "Control Building Inspections, Tests, Analyses, and Acceptance Criteria," Item 2, 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 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 nonseismic 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."

3.4.1.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 1, Sections 3.11.1 and 3.13.1, and DCA Part 2, Tier 2, Section 3.4.1, as supplemented by applicant letters dated March 5, 2018 (ADAMS Accession No. ML18064A889), and October 11, 2017 (ADAMS Accession No. ML17284A914), in accordance with SRP Section 3.4.1 to ensure compliance with the regulations as delineated in SER Section 3.4.1.3. The staff's evaluation included the review of the methodology and assumptions used in performing flood analyses and the mitigating measures for rooms that contain SSCs subject to flood protection. As stated in DCA Part 2, Tier 2, Section 3.4.1, the applicant conducted a level-by-level and room-by-room flooding analysis consisting of the following steps:

- Identify potential flooding sources.
- Identify rooms or areas that contain equipment subject to flood protection.
- Estimate the flood depth in the identified rooms or areas.
- Determine the need for protection and mitigation measures for rooms containing equipment subject to flood protection.

DCA Part 2, Tier 2, Table 3.4-1, discusses the flooding sources and lists the water sources and volumes in the RXB and CRB. For the RXB, the flooding sources include fire suppression riser; fire suppression activities; main steam (MS); feedwater (FW); site cooling water support for heating, ventilation, and air conditioning; site cooling water header; demineralized water; auxiliary boiler; the chemical and volume control system (CVCS); pool surge control system; and spent fuel pool/reactor pool cooling. In DCA Part 2, Tier 2, Section 3.4.1.1, the applicant stated that unless a stress 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 nominal diameter (DN) 50 (2-in. NPS) is assumed to have a full circumferential break in any room or area where it passes. The applicant described the results of the RXB flooding analysis in DCA Part 2, Tier 2, Section 3.4.1.2. Within the RXB, the applicant considered breaks in the fire protection main lines, fire suppression activities (e.g., area sprinklers and fire hoses), main steamline breaks, main feedline breaks, and breaks of other auxiliary fluid systems (e.g., CVCS, spent fuel pool cooling, demineralized water). As shown in DCA Part 2, Tier 2, Table 3.4-1, the spent fuel pool pipe break is the most limiting case in terms of internal flooding water sources. DCA Part 2, Tier 2, Section 3.4.1.3, described the results of the CRB flooding analysis. Within the CRB, the applicant considered breaks in the fire protection main, fire suppression activities (e.g., area sprinklers and fire hoses), and breaks in the chilled water and potable water systems. The staff determined that the applicant appropriately considered potential flooding sources that could impact the functionality of important-to-safety SSCs.

DCA Part 2, Tier 2, Table 3.4-2, gives the results of the flooding analyses. In areas where equipment subject to flood protection exists, the applicant stated that mitigation of potential flooding in the identified rooms and areas 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. As discussed in

Section 3.1.4.5 of this SER, a COL applicant that references the NuScale DC is directed to confirm the final location of SSCs subject to flood protection; select the mitigation strategy; develop an inspection and maintenance program to ensure that each watertight door, penetration seal, or other “degradable” measure remains capable of performing its intended function; and confirm that site-specific tanks or water sources are located where they cannot cause adverse flooding conditions in the RXB and CRB.

To address specific questions about the assumptions used to analyze fire suppression activities, as discussed in letters dated March 5, 2018 (ADAMS Accession No. ML18064A889), and October 11, 2017 (ADAMS Accession No. ML17284A914), the staff reviewed these assumptions in the internal flooding analyses in detail and compared them to RG 1.189 with respect to the available amount of water in the fire suppression system and the duration of the fire suppression activity. In addition, the staff reviewed the bounding conditions for all areas in the RXB and CRB and the justifications for the flow rate and fire suppression activity time duration. In DCA Part 2, Tier 2, Section 3.4.1.1, the applicant stated that fire barriers divide the RXB and CRB into fire areas and concluded that 139 square meters (m^2) (1,500 square feet (ft^2)) of fire suppression coverage is required for each fire area. Therefore, the sprinkler output for each fire area is designed to be 12 liters per minute per square meter (lpm/m^2) (0.3 gallons per minute per square foot (gpm/ft^2)) for the RXB and 8 lpm/m^2 (0.2 gpm/ft^2) for the CRB. In addition, the applicant stated that the fire suppression flow estimate for each fire area includes one manual hose stream of 950 lpm (250 gpm). Considering the methodology used by the applicant to perform the flooding analysis, which established that fire areas are compartmentalized and limited to 139 m^2 (1,500 ft^2), the staff concluded that the assumptions of fire suppression activity discharges of 2,600 lpm (700 gpm) for the RXB and 1,900 lpm (500 gpm) for the CRB are reasonable for each of the fire areas. The staff also concluded that the applicant’s assumptions about the durations of fire suppression activity, which were based on fires of 120 minutes for the RXB and 60 minutes for the CRB, are acceptable because they were based on the fire hazard classifications from the National Fire Protection Association (NFPA) 13, “Standard for the Installation of Sprinkler Systems,” 2016 edition, and NFPA 101, “Life Safety Code.”

The staff also reviewed assumptions in the internal flooding analysis with respect to the duration of pipe ruptures. In DCA Part 2, Tier 2, Section 3.4.1.1, the applicant stated that the duration of leakage from piping system ruptures was assumed to be 40 and 30 minutes between leak initiation and leak isolation for the RXB and CRB, respectively. The applicant, in a letter dated October 11, 2017, stated the following:

> These assumptions are based on plant personnel operative walk-downs, the use of plant monitoring equipment, and the use of closed-circuit video monitoring systems. The use of the Remote Camera system, Plant-wide Video Monitoring system, and Plant Security system will aid in keeping visuals on many sections of the plant, including high radiation areas. Additionally, the Control Building was assumed to have a shorter leak time because it is a normally occupied structure.

The staff finds the applicant’s justification acceptable based on the walkdowns, use of plant monitoring equipment, and video monitoring systems as described above. These are consistent with the guidance in SRP Section 3.4.1 and therefore are acceptable.

In addition, because the timely isolation of ruptured piping is also based on the prompt detection of leakages, the staff reviewed other design features such as the availability and accessibility of isolation valves, limited water volumes, or credited actions to isolate postulated ruptured piping. COL Item 3.4-2 in DCA Part 2, Tier 2, Section 3.4.1.5 and Table 1.8-2, states the following:

> A COL applicant that references the NuScale Power plant design certification will develop the on-site program addressing the key points of flood mitigation. The key points to this program include the procedures for mitigating internal flooding events; the equipment list of SSCs subject to flood protection in each plant area; and providing assurance that the program reliably mitigates flooding to the identified structures, systems, and components.

The staff finds that the applicant has provided an adequate measure as described in SRP Section 3.4.1 to ensure that potential pipe breaks in the RXB and CRB can be isolated in a timely manner.

Based on the above, the staff determined that the applicant met the design requirements of GDC 2 and GDC 4 related to the effects of natural phenomena and environmental effects of pipe breaks. The information provided in the DCA is consistent with the guidance in SRP Section 3.4.1 as it comprehensively identifies flooding hazards caused by potential line break accidents and fire suppression activities and provides appropriate mitigation measures, where needed, to preclude adverse effects on safety-related equipment and SSCs important to safety.

3.4.1.5 Combined License Information Items

SER Table 3.4.1-1 lists the COL information item numbers and descriptions (obtained from DCA Part 2, Tier 2, Table 1.8-2) that are related to internal flood protection.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.4-1	A COL applicant that references the NuScale Power plant design certification will confirm the final location of SSCs subject to flood protection and final routing of piping.	3.4.1.5
COL Item 3.4-2	A COL applicant that references the NuScale Power plant design certification will develop the on-site program addressing the key points of flood mitigation. The key points to this program include the procedures for mitigating internal flooding events; the equipment list of SSCs subject to flood protection in each plant area; and providing assurance that the program reliably mitigates flooding to the identified SSCs.	3.4.1.5
COL Item 3.4-3	A COL applicant that references the NuScale Power plant design certification 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
COL Item 3.4-4	A COL applicant that references the NuScale Power plant design certification will confirm that site-specific tanks or water sources are placed in locations where they cannot cause flooding in the RXB or CRB.	3.4.1.5

3.4.1.6 Conclusion

Based on the discussion above, the staff concludes that the NuScale design, as it relates to internal flood protection, meets the guidelines of SRP Section 3.4.1 and therefore satisfies the requirements of 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

3.4.2 Analysis Procedures

3.4.2.1 Introduction

The staff reviewed the applicant's DCA Part 2, Tier 2, Section 3.4.2, "Protection of Structures against Flood 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 DC. The staff considered the information provided by the applicant in DCA Part 2 and the responses to the staff's RAI in establishing the reasonable assurance of safety conclusion.

3.4.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 5, Table 5.0-1, provides the Tier 1 information associated with this section.

DCA Part 2, Tier 2: The applicant provided the flood and ground water site parameters in DCA Part 2, Tier 2, 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 0.3 m (1 ft) below the baseline plant elevation of 30.48 m (100 ft) and the maximum ground water elevation for the design is 0.6 m (2 ft) 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: DCA Part 2, Tier 1, Chapter 3, provides the ITAAC associated with DCA Part 2, Tier 2, Section 3.4.2. These ITAAC are evaluated in Section 14.3 of this SER.

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 regulation 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 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, 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.

SRP Section 3.4.2, Revision 3, "Analysis Procedures," issued March 2007, 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 and 2.4.12 document the staff's evaluations of the flood and ground water site parameters, respectively.

In DCA Part 2, Tier 2, Section 3.4.2.1, 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. DCA Part 2, Tier 2, 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 procedure and DCA Part 2, Tier 2, Figure 3.8.4-27, "Total Static Lateral Soil Pressure Distribution (RXB)," 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 confirmed that the applicant increased the total soil pressure using a conservative uniform loading condition to be applied to seismic Category I structures. The staff determined that the design needs to consider only the hydrostatic effects 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 DCA Part 2, Tier 2, Section 3.4.2.1, of the seismic Category I structures from ground water intrusion. The staff assessed the specified design life for waterstops, waterproofing, dampproofing, and watertight seals and considered how the expansion gap between the end of the tunnel and the corresponding connecting walls on the RXB is protected from the ground water intrusion. Additionally, the applicant proposed inclusion of COL Item 3.4-5 and COL Item 3.4-7, which will instruct a COL applicant to determine the extent of waterproofing and dampproofing needed for the underground portion of the RXB and CRB based on site-specific conditions and provide the specified design life for waterstops, waterproofing, dampproofing, and watertight seals. The COL item also instructs a COL applicant to determine the extent of waterproofing and dampproofing needed for preventing ground water and foreign material intrusion into the expansion gap between the end of the CRB tunnel and the corresponding connecting walls on the RXB. The staff reviewed COL Items 3.4-5 and 3.4-7 and the COL items directing the COL applicant to address the water-leaktight function of the below-grade portions of the RXB and CRB based on site-specific conditions. The staff considers the COL items to be appropriate for this case, as it serves to remind the COL applicant that this is necessary at the COL stage to ensure the protection of the below-grade portions of the seismic Category I structures from ground water intrusion.

DCA Part 2, Tier 2, 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 evaluations of the bounding rain and snow loads.

DCA Part 2, Tier 2, 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 RXB and CRB. The staff reviewed DCA Part 2, Tier 2, Section 3.4.2.3 and Figure 1.2-2, “NuScale Functional Boundaries,” and finds that the applicant properly accounted for the nonseismic Category I structures that are adjacent to the seismic Category I 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 nonseismic Category I structures with seismic Category I structures. The applicant stated that the nonseismic portion of the CRB was analyzed along with the seismic Category I portion of the structure to withstand the effects of the probable maximum precipitation and that the RWB has been evaluated and shown to be capable of withstanding the effects of the probable maximum precipitation. In addition, the applicant stated in DCA Part 2, Tier 2, COL Item 3.4-6, 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 DC 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.

Table 3.4.2-1 lists the COL information item numbers and descriptions related to the interaction of nonseismic Category I structures with seismic Category I structures.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.4-5	A COL applicant that references the NuScale Power Plant design certification will determine the extent of waterproofing and dampproofing needed for the underground portion of the RXB and CRB based on site-specific conditions. Additionally, a COL applicant will provide the specified design for waterstops, waterproofing, dampproofing, and watertight seals. If the design life is less than the operating life of the plant, the COL applicant should describe how continued protection will be ensured.	3.4.2.1
COL Item 3.4-6	A COL applicant that references the NuScale Power Plant design certification 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.	3.4.2.3
COL Item 3.4-7	A COL applicant that references the NuScale Power Plant design certification will determine the extent of waterproofing and dampproofing needed to prevent groundwater and foreign material intrusion into the	3.4.2.1

Item No.	Description	DCA Part 2, Tier 2, Section
	expansion gap between the end of the tunnel between the RXB and the CRB, and the corresponding RXB connecting walls.	

3.4.2.6 Conclusion

The staff concludes that the NuScale design, as it relates to protection of structures against flood from external sources, meets the guidelines of SRP Section 3.4.2. The applicant demonstrated conformance with the guidance by considering 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, the NuScale application meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, because the staff has determined that the information presented in DCA Part 2 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 DC.

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 DCA Part 2, Tier 2, Section 3.5.1.1, "Internally-Generated Missiles (Outside Containment)," and Section 3.5.1.2, "Internally-Generated Missiles (Inside Containment)." Turbine-generated 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

DCA Part 2, Tier 1: There is no DCA Part 2, Tier 1, information that directly relates to internally generated missiles or missile protection for SSCs.

DCA Part 2, Tier 2: In DCA Part 2, Tier 2, Section 3.5.1.1 and Section 3.5.1.2 describe credible and noncredible internally generated sources and missile protection for SSCs. These

DCA 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.

RG 1.117, Revision 2, "Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants," issued July 2016, provides guidance for identifying SSCs that should be protected from the effects of the worst case extreme winds and wind-generated missiles. RG 1.117, Appendix A, provides a minimum list of SSCs that should be protected from extreme wind events.

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 DCA Part 2, Tier 2, Section 3.5, "Missile Protection." The staff also reviewed DCA Part 2, Tier 1, and other DCA Part 2, Tier 2, sections noted below.

Compliance with GDC 4 is based on conforming to the guidance in RG 1.117.

DCA Part 2, Tier 2, Section 3.5, in part, addresses protection from internally generated missiles both inside and outside containment. DCA Part 2, Tier 2, Table 3.2-1, identifies safety-related and nonsafety-related SSCs throughout the plant, including the associated seismic category, QG, equipment classifications, and risk significance for each SSC. DCA Part 2, Tier 2, Section 1.2, "General Plant Description," provides the general arrangement drawings that define the building locations. The staff finds that the information in DCA Part 2, Tier 2, Table 3.2-1, is sufficient to identify all the SSCs important to safety that are subject to missile protection.

The applicant stated that safety-related SSCs and risk-significant SSCs that have a safety function following a missile-producing event are potential missile targets. The applicant stated that the following methods will provide missile protection:

- design features that prevent the generation of missiles

- orientation or physical separation of potential missile sources away from equipment subject to missile protection

- use of 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.

DCA Part 2, Tier 2, Section 3.5.1, states that the statistical significance of a potential missile is determined in accordance with the following:

1) If the probability of occurrence of the missile (P_1) is determined to be less than 10^{-7} per year, the missile is dismissed from further consideration because it is not statistically significant.

2) If (P_1) is greater than 10^{-7} per year, its probability of impacting each safety-related or risk-significant target (P_2) is determined. If the combined probability is less than 10^{-7} per year, the missile and target combination is not considered statistically significant and is dismissed from further consideration.

3) If the product of (P_1) and (P_2) is greater than 10^{-7} per year, the probability for damage to the target (P_3) is assessed. If the combined probability is less than 10^{-7} per year, the missile and target combination is not considered statistically significant and is dismissed.

4) If the product of (P_1), (P_2) and (P_3) is greater than 10^{-7} per year, barriers or other measures are taken to protect the SSC.

The staff finds this approach acceptable because it is consistent with SRP Section 3.5.1.1 and Section 3.5.1.2 acceptance criteria.

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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. Therefore, the staff finds the above list of noncredible missile sources acceptable.

DCA Part 2, Tier 2, Section 3.5.1.1.1, states that bolted bonnet valves and pressure-seal bonnet valves constructed in accordance with ASME BPV Code, Section III; ASME B16.34, "Valves Flanged, Threaded, and Welding End"; or an equivalent consensus standard are not considered credible missiles. Additionally, using a retaining ring prevents bolted bonnet valves and pressure-seal bonnet valves from becoming missiles.

The staff finds this acceptable because the specification of design and construction codes demonstrates the structural integrity of the components and minimizes the likelihood of missile generation.

The staff also reviewed information on the potential missiles generated from rotating components. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," presents the design of the main turbine generators related to missile generation, and SER Section 3.5.1.3 evaluates the design. DCA Part 2, Tier 2, Section 3.5.1.1.3, also determined that the catastrophic failure of rotating equipment, such as fans and compressors, is not a considered 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 DCA information on the potential missiles generated from rotating components acceptable, because it is 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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, because it is consistent with the guidance in SRP Section 3.5.1.1.

The staff evaluated the potential for missiles generated from explosions. DCA Part 2, Tier 2, 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 design also incorporates valve-regulated lead acid batteries that reduce the hydrogen production in battery rooms as compared to vented lead acid batteries. 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 nonseismically 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. DCA Part 2, Tier 2, Section 9.1.5, "Overhead Heavy Load Handling Systems," discusses measures used to address the safe operation of the RXB crane and module assembly equipment, and SER Section 9.1.5 evaluates such measures. In addition, procedures developed in accordance with DCA Part 2, Tier 2, 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, because it is consistent with the guidance in SRP Section 3.5.1.1.

The staff also reviewed the potential for internally generated missiles from inside containment. DCA Part 2, Tier 2, Section 3.5.1.2, states that the NPMs use a steel containment that encapsulates the reactor pressure vessel (RPV) and that there is no rotating equipment inside containment. All pressurized components inside containment, including control rod drive mechanism (CRDM) housings, are ASME BPV Code Class 1 or 2 and therefore are not considered credible missile sources. The applicant does not consider these pressurized components a credible missile source because of the material characteristics, inspections, quality control during fabrication and erection, and prudent operation. The staff reviewed the applicant's bases as described above and finds the applicant's conclusion on the elimination of the above components as credible missile sources acceptable, because it is consistent with the guidance in SRP Section 3.5.1.1. DCA Part 2, Tier 2, 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 because the applicant has conformed to the guidance in SRP Section 3.5.1.1 and Section 3.5.1.2 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

DCA Part 2, Tier 1: There is no Tier 1 information for this area of review.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.5.1.3, and the applicant's supplemental letter dated November 1, 2019 (ADAMS Accession No. ML19305E534), describe the NuScale DC, as summarized, in part, below.

In DCA Part 2, Tier 2, Figure 1.2-2, Figure 1.5-3, 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. The turbine generator rotor shafts are physically oriented such that the RXB and CRB are within the turbine low-trajectory hazard zone and therefore are considered to be unfavorably oriented with respect to the NPMs, as defined by RG 1.115, Revision 2, "Protection Against Turbine Missiles," issued January 2012.

The NuScale design employs a barrier approach, in combination with 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 (ADAMS Accession No. 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. As discussed in DCA Part 2, Tier 2, Section 3.5.3.4, 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 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 TM-5-885-1, "Fundamentals of Protective Design for Conventional Weapons," issued November 1986 (ADAMS Accession No. 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 DCA Part 2, Tier 2, Sections 3.5.1.3, 3.5.2, and 3.5.3. The staff reviewed this information using the guidelines in DSRS Sections 3.5.1.3 and 3.5.3.

DCA Part 2, Tier 2, Section 3.5.1.3, states that the turbine generators are unfavorably oriented such that essential SSCs, including the NPMs, 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 and CRB. 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 and

CRB from low-trajectory turbine missiles. NuScale included low-trajectory missiles in the analysis of the barriers 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.

In 2019, the staff conducted an audit (ADAMS Accession No. ML19018A112) to review pertinent technical information including the analyses, calculations, 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 DCA Part 2, Tier 2, 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 NUREG-1275, "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 applicant had also postulated a turbine blade and a turbine blade with a fragment of the rotor, but the staff's assessment was that these are not bounding. 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, the staff finds the applicant's proposed bounding missile weight of 1,626.6 kilograms (kg) (3,586 pounds (lb)) acceptable since it is based on a typically sized turbine of 50 megawatts electric (MWe) with a 122-centimeter (cm) (48-in.) diameter and 30.48-cm (12-in.) wide last stage turbine rotor, and a typical turbine rotor material in accordance with ASTM A470 "Standard Specification for Vacuum-Treated Carbon and Alloy Steel Forgings for Turbine Rotors and Shafts" Class 4.

The staff also finds that the applicant's proposed bounding missile speed 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 DCA Part 2, Tier 2, 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 lists the COL information item number and description related to turbine missiles from DCA Part 2, Tier 2, Table 1.8-2.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.5-1	A COL applicant that references the NuScale Power Plant certification will demonstrate that the site specific turbine missile parameters are bounded by the design certification analysis, or provide a missile analysis using the site-specific turbine generator parameters to demonstrate that the barriers adequately protect essential structures, systems and components from turbine missiles. Parameters to verify include: limiting turbine missile spectrum (rotor and blade material properties); turbine rotor design, including geometry and number of blades; final design of the RXB exterior wall; final design of the CRB exterior wall and grade-level slab; and location of the turbines with respect to the RXB and CRB.	3.5.1.3
COL Item 3.5-2	A COL applicant that references the NuScale Power Plant design certification will address the effect of turbine missiles from nearby or co-located facilities.	3.5.1.3

3.5.1.3.6 Conclusion

Based on the above, 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 to the staff, in accordance with the guidance of DSRS Sections 3.5.1.3 and 3.5.3 and RG 1.115, 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 DC will assess whether the actual site characteristics fall within the site parameters specified for the NuScale design. 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Table 5.0-1, lists the design-specific tornado and hurricane site parameters.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.5.1.4, "Missiles Generated by Tornadoes and 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. DCA Part 2, Tier 1, Table 5.0-1, 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 SRP sections.

- Regulatory Guide (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 DCA Part 2, Tier 2, Section 3.5.1.4. The staff also reviewed DCA Part 2, Tier 1, Section 5.0, “Site Parameters,” and other DCA Part 2, Tier 2, sections noted below.

DCA Part 2, Tier 2, 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 (DCA Part 2, Tier 2, Section 3.3.2.1)
  - A tornado has a maximum 3-second gust of 103 m/s (230 mph).
  - A hurricane has a maximum 3-second gust of 130 m/s (290 mph).
- tornado-generated missile spectra
  - A massive high-kinetic-energy missile that deforms on impact, such as a 1,800-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 4.6-m-long, 130.2-kg, 15-cm-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 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. DCA Part 2, Tier 2, 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. In addition, COL Item 3.5-3 states the COL applicant will confirm that automobile missiles cannot be generated within a 0.8-km (0.5-mi) radius of safety-related and risk-significant SSCs requiring missile protection that would lead to an impact higher than 9 m (30 ft) above plant grade.

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 and RG 1.221. In addition, the staff finds that COL Item 3.5-3 addresses the potential for an automobile missile impact higher than 9 m (30 ft).

In addition, the staff reviewed COL Item 3.5-4 in DCA Part 2, Tier 2, Section 3.5.2 and Table 1.8-2, and finds the proposed COL Item 3.5-4 acceptable because it addresses the potential for site-specific hazards that could produce missiles more energetic than the design-basis missiles.

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 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 and RG 1.221 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 DCA Part 2, Tier 2, Table 1.8-2) that are related to DCA Part 2, Tier 2, Section 3.5.1.4.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.5-3	A COL applicant that references the NuScale Power Plant certified design will confirm that automobile missiles cannot be generated within a [0.8-km] 0.5 mile radius of safety-related SSCs and risk significant SSCs requiring missile protection that would lead to impact higher than [9.1 m] 30 feet above plant grade. Additionally, if automobile missiles impact at higher than [9.1 m] 30 feet above plant grade, the COL applicant will evaluate and show that the missiles will not compromise safety-related and risk-significant SSCs.	3.5.1.4
COL Item 3.5-4	A COL applicant that references the NuScale Power Plant design certification will evaluate site-specific hazards for external events that may produce more energetic missiles than the design basis missiles defined in FSAR Tier 2, Section 3.5.1.4.	3.5.2

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.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

3.5.1.5.1 Introduction

This section explains that the design is based on tornado missiles, which are assumed to be the most severe missiles generally. However, hurricane missiles, if determined to be more limiting than tornado missiles, will be considered. 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 DC stage.

3.5.1.5.2 Summary of Application

DCA Part 2, Tier 1: There is no Tier 1 information for this area of review.

DCA Part 2, Tier 2: In DCA Part 2, Tier 2, 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 certified 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 52.47(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 52.47(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 DC 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 DCA Part 2, Tier 2, Table 1.8-2, COL Item 2.2-1 directs a COL applicant that references the NuScale DCA 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. DCA Part 2, Tier 2, Table 1.8-2, COL Item 3.5-2, directs a COL applicant that references the NuScale DCA to address the effect of turbine missiles from nearby or colocated facilities.

3.5.1.5.5 Combined License Information Items

The COL items to be addressed by a COL applicant referencing the NuScale DCA are presented in Table 3.5.1.5-1.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 2.2-1	A COL applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	3.5.1.5
COL Item 3.5-2	A COL applicant that references the NuScale Power Plant certified design will address the effects of turbine missiles from nearby or co-located facilities.	3.5.1.5

3.5.1.5.6 Conclusion

As described above, DCA Part 2, Tier 2, states that the COL applicant will provide the site-specific information under COL Item 2.2-1 and COL Item 3.5-2. Because this information is site specific, the applicant's statement in the NuScale DCA that the COL applicant will supply this site-specific information, and as called for in COL Item 2.2-1 and COL Item 3.5-2 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

DCA Part 2, Tier 1: There is no Tier 1 information for this area of review.

DCA Part 2, Tier 2: In DCA Part 2, Tier 2, Section 3.5.1.6, the applicant stated that, as described in Section 2.2, the NuScale Power Plant certified 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 52.47(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 52.47(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 1×10^{-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 DCA 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 DCA Part 2, Tier 2, Table 1.8-2.

3.5.1.6.5 Combined License Information Items

The COL item to be addressed by a COL applicant referencing the NuScale DCA is presented in Table 3.5.1.6-1.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 2.2-1	A COL applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	3.5.1.6

3.5.1.6.6 Conclusion

As described above, the applicant stated, in DCA Part 2, Tier 2, 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 DCA that the COL applicant will supply 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. 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 would review it at the time a COL application is submitted. This 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 guidance in SRP Section 3.5.2, Revision 3, "Structures, Systems, and Components To Be Protected from Externally-Generated Missiles," issued March 2007, 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Sections 3.11 and 3.13, provide the information associated with this section and describe the design of the RXB and CRB, respectively.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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. DCA Part 2, Tier 1, Sections 3.11 and Section 3.13, specify that design-basis loads, including those from extreme wind missiles, are to be applied to the design of the RXB and CRB, respectively. These ITAAC are evaluated in Section 14.3 of this SER.

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 1, 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 essential SSCs against externally generated missiles in accordance with the guidance in SRP Section 3.5.2, Revision 3. The staff reviewed DCA Part 2, Tier 2, Section 3.5.2; DCA Part 2, Tier 1, Sections 3.11 and 3.13; and other sections of DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Table 3.2-1, identifies the SSCs that are safety related and risk significant, and DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.2-1, as A1 (Safety-Related, Risk-Significant), A2 (Safety-Related, Non-Risk-Significant), or B1 (Non-Safety-Related, Risk-Significant) and are located inside the RXB or CRB.

The staff reviewed the NuScale application to determine whether all identified essential SSCs necessary for supporting the reactor facilities are appropriately protected from externally generated missiles. DCA Part 2, Tier 2, Section 3.5.2, states that all safety-related and risk-significant SSCs that must be protected from external missiles (i.e., those categorized as A1, A2, and B1) are located in the RXB and 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 DCA Part 2, Tier 2, Section 3.5.1.4. Based on the above information, the staff determined that the SSCs identified in DCA Part 2, Tier 2, Table 3.2-1 and Table 9A-7, as requiring missile protection are located within seismic Category I structures and openings and will be protected. Therefore, the staff concludes that this aspect of the NuScale plant design conforms to the guidance in RG 1.13 and RG 1.117.

The RXB and CRB turbine missile protection 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 DCA Part 2, Tier 1 and Tier 2, which is documented in the staff's evaluation set forth above, the staff concludes that the SSCs to be protected from externally generated missiles are 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 DCA Part 2, Tier 2, 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 wind-borne and turbine missile impact. The staff acknowledges the limitations in the applicability of the National Defense Research Committee's (NDRC's) 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 applicant's responses and supplemental responses to RAIs and confirmatory items.

The COL applicant that refers to the NuScale DC will assess whether the actual data on missile parameters are within the corresponding site parameters and the turbine characteristics of the NuScale design. The 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

DCA Part 2, Tier 1: In DCA Part 2, Tier 1, Sections 3.11 and 3.13 provide the information associated with this section and describe the design of the RXB and CRB, respectively.

DCA Part 2, Tier 2: The applicant provided the barrier design procedures used for the NuScale design in DCA Part 2, Tier 2, Section 3.5.3. For the prediction of local damage from wind-borne missiles, the applicant applied the Modified National Defense Research Committee's formulas for missile protection in concrete barriers. The applicant stated that the NuScale design does not use steel and composite barriers. For the prediction of local damage from turbine missiles, the applicant stated in Section 3.5.1.3.4.1 that a combination of a nonlinear finite element analysis (FEA) and the NDRC's equations for minimum scabbing and perforation thickness is used.

In DCA Part 2, Tier 2, Sections 3.5.1.3.1 and 3.5.1.3.2 provide the information associated with the barriers providing protection against turbine missiles for the RXB and CRB, respectively.

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 BC-TOP-9A, Revision 2, "Design of Structures for Missile Impact," issued September 1974 (ADAMS Accession No. ML14093A217), to determine the structural responses for the design-basis automobile missile. The solid sphere missile was not included for its contribution to overall structural response.

For overall damage prediction from turbine missile impact, the applicant relied on the same FEA model to assess its contribution to the overall structural response. For the turbine missiles, the applicant credited barriers in combination with the separation of the redundant safety equipment as providing the required protection.

ITAAC: There are no ITAAC directly associated with SSCs to be protected from external missiles. In DCA Part 2, Tier 1, Sections 3.11 and 3.13 specify a design commitment that RXB and CRB maintain overall structural integrity under the design-basis loads, which also include those 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 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 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 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

- RG 1.142, Revision 2, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments," issued November 2001

3.5.3.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, 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 wind-borne and turbine missile impact. The staff, in addition, reviewed the appropriateness of the finite element model (FEM) and the nonlinear analysis code used in the local and overall damage prediction for turbine missiles.

3.5.3.4.1 Finite Element Model Used for Turbine Missile Analysis

The staff reviewed the applicant's information in DCA Part 2, Tier 2, Section 3.5.1.3.4.2, on the selection of an analytical model sized smaller than the actual size of the impacted wall, along with calculations of the impact analyses. The staff finds that the penetration depth computed for strikes at locations other than the center did not vary significantly between locations. The staff concluded that the effect of the impact was localized, as shown by the strain contours indicating strain energy dissipation by punching shear. The staff also concluded that the location of the boundaries of the analytical model did not influence the resulting strain in analysis, as the strains diminished rapidly to negligible levels much before the location of the boundary. Thus, the reduced size of the analytical model has minimal effect on the analysis and is acceptable. The staff, in addition, agrees with the applicant's position that the same model can be used to establish both the local and global effects of the missile impact as the strain demands from the impact do not extend beyond the immediate vicinity of the impact zone and do not lend to any global effect. A model with a larger analysis area would not provide additional information about the global effects.

3.5.3.4.2 Nonlinear Analysis Code for Turbine Missiles

The applicant has used the TeraGrande explicit nonlinear finite element computer code to perform the turbine missile impact analysis, along with the ANACAP-U constitutive relationship to model the behavior of the reinforced concrete of the barrier and capture the strain developed in the concrete and steel as the missile penetrates the barrier. The staff has experience with the use of the TeraGrande code in the beyond-design-basis assessment of aircraft impact on nuclear power plant structures. The nonlinear capabilities of TeraGrande, along with the constitutive relationship for concrete, is well recognized and used in the industry. In Section 3.5.1.3.4.4 of the DCA, the applicant stated that the code was audited by the applicant's QAP following the guidance in RG 1.231 for commercial dedication and EPRI Technical Report TR-1025243 "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications," Revision 1, December 2013. In addition, the applicant simulated the EPRI test data from actual turbine missile tests using TeraGrande, and the comparison of the results of this simulation and the actual test data is presented in Section 3.5.1.3.4.4 of the DCA. The staff performed an audit (ADAMS Accession No. ML19018A112) of the applicant's turbine missile analysis approach, including the validation of the TeraGrande code, and determined that the TeraGrande code has a significant history of use for similar analysis problems and the computer code generated solutions that were substantially identical to accepted experimental tests (i.e., the difference between the simulation and test data is small). Therefore, the staff concluded that the TeraGrande computer code can be reliably used to estimate turbine missile penetration depth in a reinforced concrete barrier.

3.5.3.4.3 Local Damage Prediction

For the concrete missile barriers, the applicant tabulated the calculated concrete thickness to preclude perforation or scabbing from the design-basis hurricane and tornado pipe and sphere missiles in DCA Part 2, Tier 2, Table 3.5-1, "Concrete Thickness to Preclude Missile Penetration, Perforation, or Scabbing." The staff reviewed DCA Part 2, Tier 2, Table 3.5-1, and concluded that it addresses the two low-kinetic-energy missiles (i.e., pipe and sphere) but does not include the more massive high-kinetic-energy automobile missile, which is identified in DCA Part 2, Tier 2, Section 3.5.1.4, as a potential missile threat to the barrier. In a letter dated September 25, 2017 (ADAMS Accession No. ML17268A251), the applicant stated that the design-basis hurricane and tornado automobile missile are incapable of producing significant local damage, such as penetration and spalling, perforation, and scabbing, as the "massive missile," such as an automobile, is readily deformable upon the impact. Therefore, such a missile strike will have a greatly reduced penetration power and will not cause any significant local damage. The staff agrees that the automobile frame will buckle, absorbing impact energy in the process, and distribute the residual energy over a relatively large footprint, thus reducing the density of the applied impact load; hence, the automobile frame does not affect the local damage evaluation.

For the concrete missile barriers, DCA Part 2, Tier 2, Section 3.5.3.1.1, "Concrete Barriers," provides the applicant's barrier design procedures for hurricane and tornado wind-generated missiles. The applicant established penetration and spalling, perforation, and scabbing equations using the Modified NDRC's formulas and stated that the concrete barrier thicknesses calculated for perforation and scabbing are increased by 20 percent. The staff reviewed DCA Part 2, Tier 2, Section 3.5.3.1.1, and finds that the applicant applied the Modified NDRC's 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 Section F.2.1 of American Concrete Institute (ACI) 349-06, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary." Based on this review, the staff finds the barrier design procedures used for the prediction of local damage in the impacted area to be acceptable.

The staff reviewed the results of the turbine-generated missile analysis and the barrier design methodology for the RXB provided in DCA Part 2, Tier 2, Section 3.5.1.3.5. The results show that the maximum depth of penetration is around 80 percent of the wall thickness. Perforation is limited, as the last layer of reinforcement does not yield completely. The minimum wall thickness required to prevent back-face scabbing and perforation using the NDRC formula is beyond the available wall thickness of 150 cm (60 in.). Thus, the SRP Section 3.5.3 criterion is not satisfied by this barrier alone. Therefore, the applicant considered the effect of secondary missiles from scabbing on the systems and equipment housed in the service gallery. The majority of essential equipment that needs to be protected in the NuScale design is located on top of the NPM. The NPM is located behind an additional 1.5-m- (5-ft)-thick barrier (i.e., the RXB pool wall) and is unaffected by the postulated design-basis turbine missile or secondary missiles generated by concrete scabbing. In DCA Part 2, Tier 2, Section 3.5.1.3.6, the applicant evaluated all SSCs located in the RXB and CRB that require protection from turbine-generated missiles. DCA Part 2, Tier 2, Table 3.5-3, provides a summary of the results of this assessment. The applicant concluded that, within the RXB, all damage from concrete scabbing caused by a turbine-generated missile is contained within the gallery space opposite the reactor pool, at either the 30.5-m or 38.1-m (100-ft or 125-ft) elevations of the RXB. The applicant stated that in the event equipment in either of these rooms is rendered inoperable, the plant remains in a safe condition due to the redundancy of equipment in the other gallery space room. Therefore, the applicant concluded that, given the physical separation of the redundant safety-related equipment in the NuScale design, no single turbine missile can prevent an essential system from performing its safety function. The staff reviewed the applicant's analysis and determined that sufficient redundancy and spatial separation exist in the RXB gallery spaces to permit continued support for essential functions as defined by RG 1.115, Appendix A, following a turbine missile impact. Based on this review, the staff finds the barrier design methodology used for the prediction of local damage in the impacted area to be acceptable.

The staff reviewed the results of the turbine missile analysis and the barrier design methodology for the CRB in DCA Part 2, Tier 2, Section 3.5.1.3.5. The results of the analysis show that the exterior wall of the CRB is perforated, with the missile impacting the floor slab at grade level with a significantly reduced velocity. The depth of penetration is around 8 percent of the slab thickness. The minimum thickness required for scabbing and perforation from the NDRC formula is well exceeded by the slab thickness. The applicant observed that since the safety-related equipment and the main control room are below the slab at grade level, the CRB structure adequately protects the safety-related equipment. Based on this review, the staff finds the barrier design methodology used for the prediction of local damage in the impacted area to be acceptable.

Based on a letter dated September 25, 2017, the applicant stated, in DCA Part 2, Tier 2, Section 3.5.3.1.2, "Steel Barriers," that the design does not use any steel missile barriers.

In DCA Part 2, Tier 2, Section 3.5.3.1.3, "Composite Barriers," the applicant stated that the design does not use composite barriers.

3.5.3.4.4 Overall Damage Prediction

The staff reviewed DCA Part 2, Tier 2, Section 3.5.3.2, "Overall Damage Prediction," and noted that the analysis procedure did not address the design-basis sphere missile. The applicant stated that the design-basis 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 the applicant's response acceptable because the impact of the design-basis sphere missile is indeed too small to affect the overall structural response of the RXB and CRB.

In DCA Part 2, Tier 2, Section 3.5.3.2, the applicant used EPRI NP440 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 DCA Part 2, Tier 2, 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 requested that the applicant provide an engineering justification that would support the appropriateness of the use of an alternate method of qualification. In a letter dated September 25, 2017, the applicant stated that the proposed procedures use a dynamic analysis method from EPRI NP440 and BC-TOP-9A, 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 DCA Part 2, Tier 2, Revision 0, Section 3.5.3.2, the applicant stated that the design for impulsive and impactive loads is in accordance with ACI 349-06 for concrete structures and American National Standards Institute (ANSI)/American Institute of Steel Construction (AISC) N690-12, "Specification for Safety-Related Steel Structures for Nuclear Facilities," for steel structures. The applicant also stated that stress and strain limits for the missile impact equivalent static load conform to applicable codes and RG 1.142, Revision 2. Based on its review of the information the applicant provided, the staff finds the applicant's proposed alternative procedures acceptable.

For the turbine missile analysis, the staff reviewed DCA Part 2, Tier 2, Section 3.5.1.3.5, in which the applicant established, based on analysis results, that the steel near the point of impact yields, leading to localized crushing of concrete and failure by punching shear. Reinforcement away from the impact location is seen to be elastic at the end of the impact analysis. This localization of the impact and failure of the wall at the location of impact prevent the load transfer to the zones further than one diameter of the missile. The staff agrees that where a structural member is expected to deform beyond its elastic limits, the usefulness of the load combination equations presented in ACI 349-06 is rather limited, as acknowledged in Appendix F to ACI 349-06. These load combination equations do not provide any means of accounting for the additional work done as the structure continues to deform beyond its effective yield point eventually to failure. As the strain beyond the immediate zone of impact is negligible, the load transfer is nominal beyond the impact zone. The staff agrees, based on the strain contours in the calculations, that the impact does induce global demands. The applicant has presented the maximum demand to capacity (D/C) ratio from other load combinations in DCA, Part 2, Tier 2, Table 3B-14 to estimate available margin against impactive load that may cause additional demand in the overall structural component. The staff agrees with the applicant's position that no overall damage is caused by the impact and the remainder of the wall maintains its load-carrying capacity in accordance with ACI 349-06 load combinations. Based on this review of the information provided by the applicant, the staff finds the applicant's alternative approach to overall damage prediction for turbine missile impacts to both the RXB and the CRB acceptable.

3.5.3.4.5 Shock Analysis

Due to the high kinetic energy associated with a turbine missile impact, there is a potential for a shock wave to develop that could impact the functionality of SSCs that are important to safety. The staff reviewed the information provided by the applicant in DCA Part 2, Tier 2, Section 3.5.1.3.5, in which the applicant stated that the shock wave from the impact on the

RXB will not affect the redundant equipment important to safety, as the equipment is physically separated from the location of impact. The staff reviewed this information and finds that the localized failure of the reinforced concrete barrier will allow the missile to continue its path by accommodating plastic deformation of the material. Thus, this local wall segment will provide a momentary resistance followed by diminishing stiffness or resistance as the impact zone disintegrates till perforation. The resulting shock would be a pulse of a very short duration. This type of wave form does not have any influence on structural response but may affect sensitive equipment that operates across very small displacements. Given the seismic qualification of equipment to high-frequency ground motion which is of much longer duration than the impact pulse, the staff concludes that the shock from the impact will not adversely affect the protected equipment.

The CRB equipment is below grade and mounted on the floor of the CRB. The applicant, in Section 3.5.1.3.5, describes that the essential equipment is located below grade and is physically distanced from the potential impact zones that will not be affected by the shock wave from a missile impact. The staff reviewed this information and concludes that any shock wave from a missile impact on the CRB will be rapidly attenuated by the soil embedment. The staff concludes that the shock from a missile impact will not adversely affect the protected equipment.

3.5.3.5 Combined License Information Items

No COL information items from DCA Part 2, Tier 2, 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 the 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, "Nuclear Power Module," discusses information related to protection against pipe rupture effects. DCA Part 2, Tier 1, Table 2.1-1, identifies the piping systems associated with the NPMs.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, Section 3.6.1, addresses the separation and redundancy of essential systems and methods for analyzing piping failures.

ITAAC: DCA Part 2, Tier 1, Table 2.1-4, includes ITAAC 4, 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 DCA Part 2, Tier 2, Section 3.6, "Protection against Dynamic Effects Associated with Postulated Rupture of Piping," the applicant identified NuScale TR-0818-61384-P, Revision 2, "Pipe Rupture Hazards Analysis," dated July 2019 (ADAMS Accession No. ML19212A682), 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 DCA Part 2, Tier 2, 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 postaccident 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 DCA Part 2, Tier 2, Table 3.6-1, "High- and Moderate-Energy Fluid System Piping," the applicant identified the fluid systems that contain high- and moderate-energy piping.

In DCA Part 2, Tier 2, Section 3.6.1, 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 2, "Postulated

Rupture Locations in Fluid System Piping Inside and Outside Containment," issued March 2007, 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 nonseismic SSCs could fail. This section evaluates the impact of full-circumferential ruptures of nonseismic 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 DCA Part 2, Tier 2, Section 3.6.1.2, the applicant stated that the NuScale Power Plant has only a small number of safety-related and risk-significant systems and components. The applicant stated that the following systems and components are credited to ensure safe shutdown of the reactor and require protection against high-energy line breaks (HELBs):

- RCS
- module protection system (MPS)
- neutron monitoring system
- CVCS
- control rod assembly (CRA) and the control rod drive system (CRDS)
- containment system (CNTS)
- DHRS
- SGS
- emergency core cooling system (ECCS)
- UHS

In DCA Part 2, Tier 2, 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 six 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) in the RWB
(6) on site (outside the 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 SER Section 3.6.1.4.2 evaluates the consequences of these failures. Accordingly, the staff finds that the consequences of full-circumferential ruptures of nonseismic moderate-energy piping are bounded by high-energy failures.

The CRB has no high-energy lines. The failure of moderate-energy piping is evaluated for flooding environmental conditions. SER Section 3.4 evaluates flooding, and SER Section 3.11 evaluates the environmental conditions caused by pipe failure.

No essential equipment is located in the 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. SER Sections 3.6.1.4.2 and 3.6.2 document the staff's evaluation of the adequacy of the protection of essential SSCs from the impact of full-circumferential ruptures of nonseismic, 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 that require protection. The protection measures credited in the CNV are discussed in SER Section 3.6.2. The protection measures for the RXB are discussed further below (SER Section 3.6.1.4.2.3). 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.6.1, the applicant stated the following:

- On a limited basis, portions of pipe may be excluded from postulating breaks and cracks provided they meet criteria regarding the design arrangement, stress and fatigue limits, and a high level of inservice inspection (ISI). The criteria for this exclusion are provided in BTP 3-4, Section B.A.(ii), "Fluid System Piping in Containment Penetration Areas,"

- Systems that can demonstrate a low probability of rupture prior to the detection of a leak may be excluded from HELB dynamic effect considerations. This is referred to as leak before break (LBB).

- For high- and moderate-energy systems that cannot be fully excluded using the criteria of BTP 3-4, Section B.A.(ii), or LBB, line breaks and leakage cracks are postulated. The criteria for the specific locations for the postulated breaks are provided in BTP 3-4 (e.g., Section B.A.(iii)). In general, locations meeting certain stress, fatigue, and design requirements may be excluded and are not required to be postulated to rupture.

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 and 3.6.3, the staff evaluates the applicant's implementation of the pipe break location methodology and LBB methodology, respectively.

Additionally, the applicant stated 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). 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 DCA Part 2, Tier 2, Section 3.6, the applicant identified TR-0818-61384-P (DCA Reference 3.6-21) 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-0818-61384-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 six distinct pipe break zones 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 DCA Part 2, Tier 2, Section 3.6.1.1.2, 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 DCA Part 2, Tier 2, Section 3.6.2.1.2.2, "Break Exclusion," 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 nonmechanistic 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 DCA Part 2, Tier 2, 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 main steam system (MSS) piping (DN 300 (NPS 12)) and the feedwater system (FWS) (DN 100 (NPS 4)). The applicant proposed to postulate a nonmechanistic break of 77 cm^2 (12 square inches (in.^{2})) for the MSS and 37.9 cm^2 (5.87 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 nonmechanistic break size to 77 cm^2 (12 in.^{2}) for the MSS and 37.9 cm^2 (5.87 in.^{2}) for the FWS.

Section 3.5.2.2 of TR-0818-61384-P, Revision 2, 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. Section 3.9.9.2 of TR-0818-61384-P, Revision 2, indicates that the essential SSCs in this area are qualified for the pressure and temperature conditions resulting from a nonmechanistic break in the area.

The staff reviewed the information in DCA Part 2, Tier 2, and TR-0818-61384-P, Revision 1, and determined that the applicant adequately applied the methodology described in DCA Part 2, Tier 2, 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 nonmechanistic pipe failure, the applicant has protected the SSC functions important to safety against a nonmechanistic 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 DCA Part 2, Tier 2, 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, COL Item 3.6-2, and COL Item 3.6-3 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 must arrange the MSS and FWS pipes or provide pipe whip restraints to prevent a collateral rupture, or a pipe whip impact analysis must be conducted to show that a collateral rupture does not occur. However, for the purpose of the bounding analysis, TR-0818-61384-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-0818-61384-P indicates that the safety-related SSCs in the RXB (outside the bioshield) include the CIV hydraulic power unit (HPU) 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 HPU 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 CIV HPU skids 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-0818-61384-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-0818-61384-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 DCA Part 2, Tier 2, Section 3.6.1, the applicant indicated that the flooding evaluation related to postulated pipe failures is discussed in DCA Part 2, Tier 2, Section 3.4.1. The staff evaluated this section and confirmed that the applicant included an assessment of the expected flooding impact caused by the postulated pipe failures. 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 DCA Part 2, Tier 2, Section 3.6.2.1.4, the applicant indicated that the CRB has no high-energy lines. In DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Sections 3.6.1 and 3.6.2, for conducting a PRHA. The staff finds that the PRHA described in the DCA complies with the guidance in SRP Sections 3.6.1 and 3.6.2 and 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 DCA.

3.6.1.5 Combined License Information Items

SER Table 3.6.1 lists the COL information item numbers and descriptions (obtained from DCA Part 2, Tier 2, Table 1.8-2) that are related to the PRHA for site-specific high- and moderate-energy piping systems.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.6-1	A COL applicant that references the NuScale Power Plant design certification will complete the routing of piping systems outside of the CNV and the area under the bioshield, identify the locations of high- and moderate-energy lines, and update Table 3.6-1 as necessary.	3.6
COL Item 3.6-2	A COL applicant that references the NuScale Power Plant design certification will verify that the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the CNV (under the) bioshield is applicable. If changes are required, the COL applicant will update the pipe rupture hazards analysis, design additional protection features as necessary, and update Table 3.6-2.	3.6
COL Item 3.6-3	A COL applicant that references the NuScale Power Plant design certification will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay and design appropriate protection features. This includes evaluations and dispositions of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The COL applicant will update Table 3.6-2 as appropriate.	3.6

The staff reviewed the COL information items listed in Table 3.6.1-1 pertaining to pipe rupture hazards analysis discussed in DCA Part 2, Tier 2, Section 3.6.1, and found them 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 important to safety from the effects of piping failures outside containment, meets 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Table 2.1-4, Item 4, and Table 3.11-2, Item 8, describe the pertinent design commitment and associated as-built ITAAC related to protection against postulated pipe rupture effects for the safety-related SSCs of the NPM and the RXB.

DCA Part 2, Tier 2: 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, LBB, and break exclusion. DCA Part 2, Tier 2, Section 3.6, describes the applicant's approach to mitigate the dynamic effects of postulated HELBs. In addition, the applicant submitted TR-0818-61384-P, Revision 2, to supplement the PRHA-related information contained in DCA Part 2, Tier 2, Section 3.6.2. Specifically, TR-0818-61384-P, Revision 2, describes the details of the applicant's PRHA methodologies and the associated results for the NuScale plant. The updated DCA Part 2, Tier 2, Section 3.6, information is discussed below.

DCA Part 2, Tier 2, Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and its associated DCA Part 2, Tier 2, 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 in the Radioactive Waste Building"; and Section 3.6.2.1.6, "Pipe Breaks Onsite (i.e., Outside the Buildings)," 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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." In DCA Part 2, Tier 2, Section 3.6.2.1.7, "Types of Breaks," and Section 3.6.2.1.8, "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.

DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, Section 3.6.2.2.2, "Pipe Whip," describes the methodology for assessing the pipe whip effects. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Table 1.8-2, lists three action items (COL Item 3.6-1, COL Item 3.6-2, and COL Item 3.6-3) for a COL applicant in regard to the PRHA for its associated plant areas. DCA Part 2, Tier 2, Section 3.6.1.1.6, "Onsite (outside the buildings)," Section 3.6.2.1.2, and Section 3.6.2.1.3, describe those three COL information items, respectively.

ITAAC: As discussed above, DCA Part 2, Tier 1, Table 2.1-1, Item 4, and Table 3.11-2, Item 8, 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-0818-61384-P as the relevant TR. DCA Part 2, Tier 2, Section 1.6, "Material Referenced," Table 1.6-2, "NuScale Referenced Technical Reports," has specified that TR-0818-61384-P, Revision 2, is incorporated by reference into the NuScale DCA.

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 2, 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, Revision 2, 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 plant 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 DCA Part 2, Tier 2, Section 3.6, and TR-0818-61384-P, Revision 2.

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

DCA Part 2, Tier 2, Section 3.6.1.1, defines high- and moderate-energy piping systems. DCA Part 2, Tier 2, Table 3.6-1, lists 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 DCA Part 2, Tier 2, Table 3.6-1, is within the scope of SRP Section 3.6.1 and is described in SER Section 3.6.1.4.

DCA Part 2, Tier 2, Section 3.6.1.1, 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.

DCA Part 2, Tier 2, 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 six groups, including inside the CNV, outside the CNV (under the bioshield), in the RXB (outside the bioshield), in the CRB, in the RWB, and on site (outside the RXB, CRB, and RWB). 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 plant in DCA Part 2, Tier 2, Section 3.6. In DCA Part 2, Tier 2, Sections 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.

DCA Part 2, Tier 2, Section 3.6.2.1.2, states that the CIVs for the RCS injection, RCS discharge, PZR spray, and RPV high-point degasification lines are each dual independent valves in a single body that is welded directly to an Alloy 690 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 (injection and spray) or excess flow check (discharge) valve welded directly to 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. The MSS lines each have a single CIV. Between the CNV nozzle and the valve body are two tee fittings to which the DHRS steamlines 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.

DCA Part 2, Tier 2, 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 or check/excess flow check valve. In addition, the applicant stated that, for welds in the containment penetration areas, 100 percent of the volumetric examination provisions of BTP 3-4, Section B, Item A(ii), has been applied to preclude the need to postulate breaks.

In DCA Part 2, Tier 2, 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'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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.6.2.1.2.2, and Appendix A, "Break Exclusion—Compliance with Regulatory Acceptance Criteria," to TR-0818-61384-P, Revision 2, 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 DCA Part 2, Tier 2, 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-0818-61384-P, Revision 2, 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 in BTP 3-4, Section B, Item A(ii)(7), and meet the BTP 3-4 stress and fatigue limits, to ensure a low probability of rupture. DCA Part 2, Tier 2, Table 6.2-3, "Containment Vessel Inspection Elements," and Table 6.6-1, "Examination Categories and Methods," 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 NuScale DCA Part 2, Tier 2, Section 3.6.2.1.2. In DCA Part 2, Tier 2, Section 3.6.2.1.2.2; Section 3.6.2.1.2.3, "Leakage Cracks"; Figure 3.6-33, "Application of BTP 3-4 Break Location Guidance in the NPM Bay and RXB"; and in Appendix A to TR-0818-61384-P, Revision 2, 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 DCA Part 2, Tier 2, Sections 3.6.2.1.2, 3.6.2.1.2.2, 3.6.2.1.2.3, Figure 3.6-33, Table 6.2-3, Table 6.6-1, and Appendix A to TR-0818-61384-P, Revision 2, 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. 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 DCA Part 2, Tier 2, Section 3.6.2.7, the applicant stated that each of three RVVs and each of two RRVs in the NuScale design are bolted directly to reactor vessel nozzles. These five 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 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 highly 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. ISI requirements for bolting associated with the RPV are provided in NuScale DCA Part 2, Tier 2, Table 5.2-6, "Reactor Pressure Vessel Inspection Elements." The applicant stated that the highly sensitive leakage monitoring system (being sensitive to a leak rate as low as 3.79 milliliters (0.001 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.

In addition, from August 31, 2019, to September 19, 2019, the staff conducted an audit of the ECCS valve flange bolting stress and fatigue calculation as described in the audit report (ADAMS Accession No. ML19340A019). The staff's audit findings concluded that the applicant's stress and fatigue evaluation meets those pertinent stress and fatigue design limits as described in DCA Part 2, Tier 2, Section 3.6.2.7, and is therefore acceptable.

Based on the review of the information described in the above applicable DCA sections and the staff's audit findings described in the audit report, 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 DCA Part 2, Tier 2, 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 3.6.2.1.8, 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 DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, 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 finds that those DCA 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.

DCA Part 2, Tier 2, Section 3.6.2.1, 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.

DCA Part 2, Tier 2, Section 3.6.2.1.8, states that, for high-energy lines, except for those portions of piping within the break exclusion areas as described in DCA Part 2, Tier 2, 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 finds those criteria, as described in DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.6.2.1.8, 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 finds those criteria, as described in DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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 finds this criterion in DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, Section 3.6.2.1.7, 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. DCA Part 2, Tier 2, Section 3.6.2.1.8, describes the postulated high- and moderate-energy leakage cracks for the NuScale plant. In DCA Part 2, Tier 2, Sections 3.6.2.1.7 and 3.6.2.1.8 describe the respective criteria for determining the postulated rupture configurations and the sizes for circumferential breaks, longitudinal breaks, and leakage cracks.

DCA Part 2, Tier 2, Section 3.6.2.1.7, 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 finds those criteria, as described in DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.6.2.1.8, 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 staff finds that the information described in DCA Part 2, Tier 2, Section 3.6.2.1.8, 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.

DCA Part 2, Tier 2, Section 3.6.2.1.8, also states that fluid flow from a leakage crack should be based on a circular opening of area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width. It also states that 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 finds those criteria, as described in DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, 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-0818-61384-P, Revision 2 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 Nonconservatism 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 DCA Part 2, Tier 2, Section 3.6.2.2.1, and TR-0818-61384-P, Revision 2, Appendix F. DCA Part 2, Tier 2, 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 plant design.

DCA Part 2, Tier 2, 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 18.0 code. The applicant demonstrated the acceptability of using the ANSYS CFX Version 18.0 code 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-0818-61384-P, Revision 2, 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 18.0 code 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 plant.

DCA Part 2, Tier 2, 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 1/25th 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 plant. 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

DCA Part 2, Tier 2, Section 3.6.2.2.3, and TR-0818-61384-P, Revision 2, 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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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 determining the two-phase jet thrust force, a thrust coefficient of 1.26 is applied. 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-0818-61384-P, Revision 2, 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.

DCA Part 2, Tier 2, Section 3.6.2.2.3, and TR-0818-61384-P, Revision 2, 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-0818-61384-P, Revision 2, 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-0818-61384-P, Revision 2, Appendix F, a jet expansion half-angle exceeding 60 degrees was seen for the initial jet formation. 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-0818-61384-P, Revision 2, Table E-2, "CVCS Steam Jet Impingement Pressure Versus Distance," compares the CVCS steam jet impingement pressure to the jet penetration distance for a jet spreading half-angle of 30 degrees and 60 degrees, respectively. The comparison shows that, beyond 2.5 cm (1 in.) from the break exit, the assumed 30-degree half-angle expansion would result in a jet impingement pressure of at least 300 percent higher than the jet pressure resulting from the expected minimum 60-degree half-angle jet expansion. Therefore, the applicant concluded that a 30-degree half-angle jet expansion assumption is sufficiently conservative to bound actual jet impingement pressures caused by local variations (i.e., center or edge peaking) within the jet. The applicant stated that the ZOI for the steam jet in the NuScale CNV is conservatively assumed to be in the forward-facing hemisphere.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.6.2.2.3, and TR-0818-61384-P, Revision 2, 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 flowpath. 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, noncondensable 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 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 axisymmetry, 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 plant, and the potential dynamic amplification and resonance-induced pressure loading is not a concern for jet impingement on the NuScale plant 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 NuScale plant 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

DCA Part 2, Tier 2, Section 3.6.2.2.2, and TR-0818-61384-P, Revision 2, 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 or LBB, 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-0818-61384-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 DCA Part 2, Tier 2, Section 3.6.2.2.2, and TR-0818-61384-P, Revision 2, 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. DCA Part 2, Tier 2, Section 3.6.2.3.1.1, describes the design criteria for the pipe whip restraints for the NuScale plant. 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 N690-12. 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.6-4, "NuScale Power Module Piping Systems Design and Operating Parameters," the applicant stated that the initial conditions assumed for dynamic response to pipe breaks are selected to bound system conditions for hot standby or at 102-percent power. The staff finds that the initial conditions assumed for dynamic response to pipe breaks is 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 DCA Part 2, Tier 2, Section 3.6.2, the applicant submitted TR-0818-61384-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-0818-61384-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.

DCA Part 2, Tier 2, Section 3.6.2.2.3, and TR-0818-61384-P, Revision 2, 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 12 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 LBB 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 DCA Part 2, Tier 2, 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-0818-61384-P, Revision 2, 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 DCA Part 2, Tier 2, Section 3.9, for components and supports and with DCA Part 2, Tier 2, 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 blast effects load is small and its effect on load combinations is inconsequential. Therefore, its exclusion does not affect compliance with the applicable ASME BPV Code allowable limits. In Table 5-1, "Summary of Approach and Result for Line Break Assessment by Plant Area," of TR-0818-61384-P, Revision 2, the applicant summarized the evaluations and results of the NuScale PRHA analysis. The applicant stated that the application of the criteria for break exclusion and LBB 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-0818-61384-P, Revision 2, 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-0818-61384-P, Revision 2, demonstrate that the NuScale 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 LBB, 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 3.6.3, 3.8.4, 6.2.1, 3.4, and 3.11, respectively. In TR-0818-61384-P, Revision 2, Section 3.5.2.5, the applicant addressed the issue related to BTP 3-3, Section B, Item 1.a(1), for a postulated nonmechanistic 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-0818-61384-P, Revision 2, addresses the effects of high-energy and moderate-energy pipe breaks and cracks in the NuScale NPM and RXB. As stated in DCA Part 2, Tier 2, Sections 3.6.1.1.6, 3.6.2.1.2, and 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, COL Item 3.6-2, and COL Item 3.6-3. SER Section 3.6.2.5 describes the staff's evaluation of these three COL information items.

3.6.2.5 Combined License Information Items

SER Table 3.6.2-1 lists the COL information item numbers and descriptions from DCA Part 2, Tier 2, Table 1.8-2, related to the PRHAs for their associated plant areas.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.6-1	A COL applicant that references the NuScale Power Plant design certification will complete the routing of piping systems outside of the CNV and the area under the bioshield, identify the location of high- and moderate-energy lines, and update Table 3.6-1 as necessary. This activity includes the performance of associated final piping stress analyses, design and qualification of associated piping supports, evaluation of subcompartment pressurization effects (if applicable), and completion of the Balance of Plant Pipe Rupture Hazards Analysis, including the design and evaluation of pipe whip/jet impingement mitigation devices as required. This includes an evaluation and disposition of multi-module impacts in common pipe galleries.	3.6
COL Item 3.6-2	A COL applicant that references the NuScale Power Plant design certification will verify that the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the CNV (under the bioshield) is applicable. If changes are required, the COL applicant will update the pipe rupture hazards analysis, design additional protection features as necessary, and update Table 3.6-2.	3.6
COL Item 3.6-3	A COL applicant that references the NuScale Power Plant design certification 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 RXB, and update Table 3.6-2	3.6

Item No.	Description	DCA Part 2, Tier 2, Section
	as appropriate. This includes an evaluation and disposition of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The COL applicant will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB or perform the pipe rupture hazards analysis of the high- and moderate-energy lines outside the buildings.	

In DCA Part 2, Tier 2, Sections 3.6.1.1.6, 3.6.2.1.2, and 3.6.2.1.3 describe the details of those three COL information items, respectively. The staff finds that those three COL information items adequately describe 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 DC 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

DCA Part 2, Tier 2, 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

By application dated December 31, 2016 (ADAMS Accession No. ML17013A229), as supplemented by letters dated August 3, 2017 (ADAMS Accession Nos. ML17215A977 and ML17215A978), the applicant addressed the application of LBB to the MSS and FWS piping inside the CNV.

DCA Part 2, Tier 1: The Tier 1 information associated with this section appears in DCA Part 2, Tier 1, Section 2.1, in which the applicant described the components of the RCS, including LBB piping. The Tier 1 information includes design information and ITAAC related to verification that the ASME BPV Code Class 2 piping systems and interconnected equipment nozzles will be evaluated for LBB.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.6.3, "Leak-Before-Break (LBB) Evaluation Procedures," provides a Tier 2 description of LBB evaluation procedures.

The applicant indicated that the application of LBB is limited to the ASME BPV Code Class 2 MSS and FWS piping inside the CNV. The FWS piping analysis addresses significant FW cyclic transients and produces bounding loads for the ASME BPV Code Class 2 piping with respect to LBB.

The applicant followed methods and criteria to evaluate LBB that are consistent with the guidance in SRP Section 3.6.3 and NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, issued November 1984. DCA Part 2, Tier 2, Section 3.6.3.1, describes the potential degradation mechanisms; Section 3.6.3.2 details the materials used in the MSS and FW piping; Section 3.6.3.3 describes the analysis methodology involving load combination methods, leakage flaw size estimation, and the flaw stability method using limit-load analyses; Section 3.6.3.4 provides the LBB analysis results for the MSS and FWS piping in the form of smooth bounding analysis curves (SBACs); and Section 3.6.3.5 discusses leak detection.

ITAAC: As noted above in this section, DCA Part 2, Tier 1, Section 2.1, lists the ITAAC related to LBB. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: There are no GTS directly related to this area of review. The review of the leakage detection and GTS 3.7.3 related to the LBB application are discussed in Section 5.2.5 of this SER.

Technical Reports: There are no TRs related to LBB materials and design.

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.

The LBB should be applied only to high-energy, ASME BPV Code Class 1 or 2 piping or the equivalent. Applications to other high-energy piping will be considered based on an evaluation of the proposed design and ISI requirements as compared to ASME BPV Code Classes 1 and 2.

Approval of the elimination of dynamic effects from postulated pipe ruptures is obtained individually for particular piping systems at specific nuclear power units. LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points. When LBB technology is applied, all potential pipe rupture locations are examined. The examinations are not limited to those postulated pipe rupture locations determined from SRP Section 3.6.2.

The guidance in SRP Section 3.6.3 lists the acceptance criteria adequate to meet the above requirements and provides review interfaces with other SRP sections.

3.6.3.4 Technical Evaluation

This section describes the technical evaluation of DCA Part 2, Tier 2, Section 3.6.3, in the order in which it is presented. The staff's review of the applicant's LBB evaluation procedures is closely related to its review of the RCPB leakage detection system in SER Section 5.2.5.

3.6.3.4.3 Potential Degradation Mechanisms for Piping

The applicant has explained how it reviewed and addressed various degradation mechanisms for piping, as described below.

3.6.3.4.3.1 Erosion/Corrosion

The applicant has indicated that the MSS and FWS piping is fabricated from SA-312 and SA-182 Type 304/304L austenitic stainless steel material and compatible austenitic stainless steel weld filler metals. Austenitic piping materials are not susceptible to erosion/corrosion. The selection of these materials and implementation of water chemistry control provide assurance that the likelihood of failure from wall thinning by erosion/corrosion is very low.

With regard to degradation resulting from cavitation, the applicant stated that the MSS and FWS piping inside the CNV does not have inline components that significantly decrease the pressure of the fluid in the piping in the direction of flow. Therefore, conditions conducive to fluid cavitation do not exist.

Based on the information provided by the applicant above in the application, the staff concludes that the MSS and FWS piping being evaluated for LBB is not susceptible to failure by erosion/corrosion and will not violate ASME BPV Code requirements.

3.6.3.4.3.2 Stress-Corrosion Cracking

For stress-corrosion cracking (SCC) to occur, material susceptibility, a corrosive environment, and tensile stress conditions must occur simultaneously and within the limited ranges for each parameter.

The applicant indicated that both the MSS and FWS materials are SCC resistant and are not exposed to a corrosive environment and that tensile stresses are not present. Therefore, between materials selection and water chemistry control, the likelihood of MSS and FWS piping failure resulting from SCC is very low.

DCA Part 2, Tier 2, Section 3.6.3.1.2, states that, during reactor shutdown conditions, the outside surfaces of some piping inside the CNV are exposed to borated water. Minimizing the chloride levels in the water, along with the low levels of oxygen in the water, reduces the potential for SCC. The temperature of the water on the outside of the piping is maintained near room temperature, which prevents SCC initiation in conjunction with minimizing chlorides in solution. Water chemistry conditions during shutdown conditions are controlled to preclude SCC initiation on the outer surface of the piping by using the water treatment methods described in DCA Part 2, Tier 2, Section 10.3.5.

The applicant stated that SA-312 TP304/304L dual-certified stainless steel is also resistant to SCC, given adequate control of dissolved oxygen levels. The alloy contains 0.03-maximum-weight-percent carbon, which mitigates sensitization. The use of cold-worked austenitic stainless steels is generally avoided; however, if such stainless steels are used, the yield strength, as determined by the 0.2-percent offset method, does not exceed 620 MPa (90 kilopounds per square inch (ksi)). The applicant further stated that, if cold bending is used, the maximum strain that could be induced in the MSS and FWS pipes is 15 percent.

DCA Part 2, Tier 2, Section 3.6.3.1.2, states that cold-worked LBB pipes will be subjected to a solution annealing process if the fracture toughness reduction caused by cold working would affect the applicability of the limit-load analysis methodology.

Based on the information presented in the application, the staff concludes that, based on the material selected, the water chemistry controls, and the solution annealing process after cold working, the MSS piping and FWS piping being evaluated for LBB are not susceptible to failure by SCC.

3.6.3.4.3.3 Creep and Creep Failure

Creep and creep failure are typically not of concern for austenitic stainless steel piping below 427 degrees C (800 degrees F). Because the design and operating temperatures of the piping systems are below these limits for both the MSS and FWS, creep and creep fatigue failure mechanisms are not a concern for LBB piping. Because the austenitic stainless steel piping is below the temperature where creep and creep failure are a concern, the staff concludes that creep and creep failure are not a concern.

3.6.3.4.3.4 Water Hammer/Steam Hammer

DCA Part 2, Tier 2, Section 3.6.3.1.4, specifies that the MSS piping design considers the dynamic load resulting from water hammer by using drain pots, line sloping, and drain valves to minimize their effect. DCA Part 2, Tier 2, Section 3.6.3.1.4, states that the FWS and SGs also have features to minimize the water hammer dynamic load effects.

The applicant further stated that it calculated the piping forces and moments for the MSS and FWS LBB piping lines and compared them with the SSE loads at each location. The applicant stated that it compared the water hammer loads to the SSE loads for the MSS and FWS lines. The results showed that the water hammer loads for the MSS and FWS lines were below the SSE loads. DCA Part 2, Tier 2, Section 3.6.3.1.4, states that the SSE loading used for the LBB evaluations bounds the water hammer loading for both the FWS and MSS lines.

Based on the information in the application, the staff concludes that the MSS piping and FWS piping being evaluated for LBB are not susceptible to failure by water hammer.

3.6.3.4.3.5 Fatigue

DCA Part 2, Tier 2, Section 3.6.3.1.5, states that low cycle fatigue is addressed by using stress reduction factors from ASME BPV Code, Section III, Subsection NC, for Class 2 piping. The staff finds this approach to be acceptable because ASME BPV Code, Section III, thoroughly defines the stress intensification factors for various piping components under fatigue loading; therefore, the effect of low cycle fatigue is not a concern.

3.6.3.4.3.6 Thermal Aging Embrittlement

DCA Part 2, Tier 2, Section 3.6.3.1.6, addresses thermal aging of stainless steel materials used in piping systems that NuScale proposes to qualify for LBB. DCA Part 2, Tier 2, Section 3.6.3.1.6, indicates that the ferrite content for austenitic stainless steel with low molybdenum (308/308L) and high molybdenum (316/316L) is limited to ferrite numbers of 5-20 and 5-16, respectively, and is consistent with the guidance in RG 1.31, Revision 4, "Control of Ferrite Content in Stainless Steel Weld Metal," issued October 2013. In addition, the applicant stated that the piping for which LBB will be applied is SA-312 TP304/304L stainless steel.

DCA Part 2, Tier 2, Section 3.6.3.2.3, states that only gas tungsten arc welding is used for MSS and FWS piping subject to LBB qualification and that weld filler metals are limited to SFA-5.9 (ER308, ER308L, ER316, and ER316L) and SFA-530 (IN308, IN308L, IN316, and IN316L).

Therefore, because the applicant proposed the use of materials for LBB areas that are not subject to thermal aging embrittlement, the staff concludes that thermal aging is not a concern in the piping systems that the applicant is qualifying for LBB.

3.6.3.4.3.7 Thermal Stratification

Thermal stratification in piping occurs when fluid at a significantly different temperature is introduced into a long horizontal run of piping. DCA Part 2, Tier 2, Section 3.6.3.1.7, indicates that, because the MSS and FWS do not have any long horizontal pipe runs, the likelihood of failure resulting from thermal stratification is very low.

Based on the information provided by the applicant, which states that thermal stratification in piping occurs when fluid at a significantly different temperature is introduced into a long horizontal run of piping, the staff concludes that since the FWS and MSS lines being applied to LBB do not have long horizontal pipe runs, failure resulting from thermal stratification is not likely.

3.6.3.4.3.8 Irradiation Effects

Irradiation-assisted stress-corrosion cracking (IASCC) typically affects components such as core support structures in regions with high neutron fluence near the core and inside the reactor vessel. DCA Part 2, Tier 2, Section 3.6.3.1.8, states that, because the neutron fluence level is low, the likelihood of the MSS and FWS being susceptible to IASCC is very low. Additionally, the piping material toughness should have no radiation degradation. The staff finds that, because the MSS and FWS piping is outside of the reactor vessel and above the core, the fluence is insufficient to be an IASCC concern.

3.6.3.4.3.9 Rupture from Indirect Causes

DCA Part 2, Tier 2, Section 3.6.3.1.9, states that, because the MSS and FWS piping is located inside the CNV, the likelihood of rupture from indirect causes such as fires, missiles, or other natural phenomena is very low because the design precludes them. DCA Part 2, Tier 2, Section 3.6.3.1.9, specifically notes the following:

- The NPM and the components inside the CNV are safety related and seismic Category I, which precludes adverse interactions from a seismic event.
- Because the NPM and the components are inside the CNV, they are protected from fires, external missiles, or damage from moving heavy loads.
- There are no internal missile sources inside containment (see Section 3.5 of DCA Part 2, Tier 2).
- Containment is flooded as part of the normal shutdown process; therefore, the design considers flooding.

Based on the information provided above by the applicant, the staff finds the programs in place to prevent pipe degradation or failure from indirect causes is very low and are acceptable.

3.6.3.4.3.10 Cleavage-Type Rupture

DCA Part 2, Tier 2, Section 3.6.3.1.10, states that the FWS and MSS piping is made of austenitic stainless steels and nickel-based alloys that are highly ductile; therefore, the likelihood of failure from cleavage type rupture is very low. This is true for the stainless steel welds (regardless of welding procedure) and Inconel 690 safe-ends and welds. Based on the information provided by the applicant that the austenitic stainless steel and nickel-based alloys are highly ductile, the staff finds that cleavage-type rupture is not a concern.

3.6.3.4.4 Materials

DCA Part 2, Tier 2, Section 3.6.3.2.1, lists the six segments that are analyzed for the MSS piping, including the following:

- DN 200 (NPS 8) piping connecting perpendicularly to DN 300 (NPS 12) piping
- a transition where DN 300 (NPS 12) piping reduces to be welded to a DN 200 (NPS 8) elbow

DCA Part 2, Tier 2, Section 3.6.3.2.1, identifies the locations where cracks were postulated for the perpendicular connection between the DN 200 (NPS 8) pipes and the DN 300 (NPS 12) pipes and the transition where the DN 300 (NPS 12) piping reduces to be welded to the DN 200 (NPS 8) elbow. The applicant stated that, at these locations, the stress points were below the SBAC and, therefore, these locations meet the LBB criteria.

DCA Part 2, Tier 2, Section 3.6.3.2.4, describes the procedure used to estimate the weld metal minimum yield strength, which is needed for conducting the limit-load analysis described in DCA Part 2, Tier 2, Section 3.6.3.3.2. The applicant stated that it used weld metal strength in the LBB analysis because it did not know whether the SBAC for the weld would be bounded by that of the base metal. The applicant stated that the SBAC for the weld is overall higher than the

SBAC for the base metal. In addition, the applicant stated that gas tungsten arc welding would be used for the FWS and MSS lines subject to LBB qualification. The applicant added the weld metals and welding process to DCA Part 2, Tier 2, Section 3.6.3.2.3.

A review of DCA Part 2, Tier 2, Section 3.6.3.2.5, indicates that some of the crack morphology parameters (i.e., roughness, number of turns, flowpath/thickness ratio) are for air fatigue cracks and others are for intergranular stress-corrosion cracking flaws as identified in Section 3.2.1 through Section 3.2.4 of NUREG/CR-6004, “Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications,” issued April 1995. The applicant stated that only these two cracking mechanisms were considered because the base metal and welds for the MSS and FWS lines are austenitic stainless steel. No cast material is used for this piping. The staff concluded that based on the information provided by the applicant, the materials selected for the MSS and FWS that are analyzed for LBB are acceptable, based on the guidelines followed in NUREG/CR-6004 and the operating experience derived from the use of these materials in the operating fleet.

3.6.3.4.5 Analysis Methodology

DCA Part 2, Tier 2, Section 3.6.3.3, describes the LBB analysis methods followed by the applicant and what adheres to the requirements in SRP Section 3.6.3 of a margin of 10 on the leak rate and a margin of 2.0 on the flaw size. The applicant described the load combination methods (both the algebraic and absolute sum) that are used, along with a limit-load (net section collapse) analysis to predict flaw stability (failure). For leak rate calculations, the elastic-plastic fracture mechanics method is used to predict the leakage flaw size for any given leak rate with crack morphology parameters. The results of the LBB analyses of both the MSS and FWS are presented in graphical form as SBACs.

For clarification, the applicant provided an example of how it developed a moment versus leak rate curve, which is an intermediate step in the LBB analysis procedure and is needed to verify the margin of safety associated with the SBAC approach described in DCA Part 2, Tier 2, Section 3.6.3.3. The staff reviewed the applicant’s methodology and was able to recreate the applicant’s results following the methodology. Based on the staff’s confirmatory analysis of the information and examples provided by the applicant, the staff concludes that the applicant’s methodology is acceptable.

3.6.3.4.5.1 Analysis of Main Steam and Feedwater Piping inside Containment

The equations in DCA Part 2, Tier 2, Section 3.6.3.3, have been used to generate the SBAC figures referenced in DCA Part 2, Tier 2, Section 3.6.3.4; they show normal operating (N) and maximum (N+SSE) stresses for LBB locations.

By letters dated November 15, 2017 (ADAMS Accession No. ML17319B002), and July 23, 2018 (ADAMS Accession No. ML18204A142), the applicant provided proprietary tables and data to enable the staff to understand how the moment versus leak rate curves were generated for the LBB analysis. In addition, the applicant gave the staff proprietary data to enable it to proceed with the confirmatory analysis and generate the SBACs.

DCA Part 2, Tier 2, Section 3.6.3.4.1.6, discusses the load-limit analysis for the base metal of the DN 200 (NPS 8) elbow. In a letter dated November 17, 2017 (ADAMS Accession No. ML17319B002), the applicant clarified that the crack analyzed for the DN 200 (NPS 8) elbow is the through-wall circumferential crack at the outer radius of the elbow. The NRC staff performed a confirmatory analysis on the DN 100 and DN 125 (NPS 4 and NPS 5) FWS piping and on the DN 200 and DN 300 (NPS 8 and NPS 12) MSS piping. The staff performed the confirmatory analysis based on the revised proprietary tables and data received from the applicant (ADAMS Accession No. ML19157A326 and ML19176A580). The NRC staff selected the locations for both the FWS and MSS that had the smallest margins (i.e., closest to the SBAC) to perform the independent confirmatory analysis. The results of the independent confirmatory analysis for the FWS and MSS verified the acceptability of the applicant's limit-load analysis and its SBACs.

The NRC staff concludes that there is reasonable assurance the applicant's LBB analysis bounds the (1) normal operation and (2) normal operation plus SSE design conditions and finds the applicant's LBB analysis acceptable.

3.6.3.4.6 Leak Detection

Although DCA Part 2, Tier 2, Section 5.2.5, describes the leak detection system in detail, DCA Part 2, Tier 2, Section 3.6.3.5, describes how the system complies with Staff Regulatory Positions 2.1 and 2.2 in RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage." Leakage monitoring is provided by two means: (1) change in pressure within the CNV and (2) collected condensate from the containment evacuation system sample vessel. The review of the leak detection and a technical specifications requirement relating to LBB application are discussed in Section 5.2.5 of this SER. The maximum leak rate specified in GTS 3.7.3 is consistent with the deterministic fracture mechanics evaluation supporting the LBB application and supports the confirmatory analysis of the LBB evaluation.

3.6.3.5 Combined License Information Items

There are no COL information items from DCA Part 2, Tier 2, that affect this section.

3.6.3.6 Conclusion

Based on its review, the staff concludes that the LBB evaluation procedures and methods identified by the applicant in the DCA are acceptable and comply with the acceptance criteria in SRP Section 3.6.3.

The provisions for LBB were based on sound engineering principles and on the following:

- Evaluation of potential degradation mechanisms, including water hammer, corrosion, creep, fatigue, erosion, environmental conditions, and indirect sources are remote causes of pipe rupture.

- The deterministic fracture mechanics evaluation method has been completed and approved by the staff.

- The leak detection systems are sufficiently reliable, redundant, diverse, and sensitive, and margin exists to detect the through-wall flaw used in the deterministic fracture mechanics evaluation.

- Based on the proprietary information provided by the applicant, the staff was able to perform the confirmatory analysis of the LBB SBACs. The NRC staff concludes that there is reasonable assurance the applicant's LBB analysis bounds the (1) normal operation and (2) normal operation plus SSE design conditions and finds the applicant's LBB analysis acceptable.

Compliance with the criteria in SRP Section 3.6.3 constitutes an acceptable basis for satisfying the requirements of 10 CFR Part 50, Appendix A, GDC 4, and the applicable requirements and acceptance criteria. Therefore, consideration of the dynamic effects of pipe rupture for the applicable piping may be eliminated from the design.

3.7 Seismic Design

3.7.1 Seismic Design Parameters

3.7.1.1 Introduction

DCA Part 2, Tier 2, 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 of the NuScale standard plant. DCA Part 2, Tier 1, Section 5.0, specifies a set of design parameters that bound the site conditions that are suitable for standard plant operation. 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
- the supporting media for seismic Category I structures

3.7.1.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 5, "Site Parameters," provides the Tier 1 information associated with this section.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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.

ITAAC: There are no ITAAC associated with DCA Part 2, Tier 2, Section 3.7.1.

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

Technical Reports: There are no TRs associated with DCA Part 2, Tier 2, Section 3.7.1.

3.7.1.3 Regulatory Basis

DSRS Section 3.7.1, "Seismic Design Parameters," describes the relevant requirements of the NRC's regulations for seismic design parameters and the associated acceptance criteria. The following NRC regulations contain the relevant requirements for this review:

- In GDC 2, the NRC requires the design basis for SSCs important to safety to 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 historical data have been accumulated.

- In 10 CFR Part 50, Appendix S, the NRC requires, in part, that, for SSE ground motions, 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 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 one-third or less of the SSE, an explicit analysis or design is not required. 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. Appendix S also requires the horizontal component of the SSE ground motion in the free field at the foundation level of the structures to be an appropriate response spectrum with a peak ground acceleration of at least 1/10 the acceleration of gravity (0.1g).

- In 10 CFR 52.47(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.

The guidance in DSRS Section 3.7.1 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:

- DSRS Section 3.7.1, Revision 0, issued June 2016, for reviewing seismic design parameters to ensure that they are appropriate and contain a sufficient margin to allow seismic analyses (reviewed under other DSRS sections) to accurately or conservatively represent the behavior of SSCs during postulated seismic events (ADAMS Accession No. ML15355A384)

- 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 (ADAMS Accession No. ML13210A432)

- RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," issued March 2007, for determining the acceptability of damping values used in the dynamic seismic analyses of seismic Category I SSCs (ADAMS Accession No. ML070260029)

- Interim Staff Guidance (ISG) 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 (ADAMS Accession No. ML081400293)

- DC/COL-ISG-017, "Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analysis," dated March 24, 2010 (ADAMS Accession No. ML100570203)

- NUREG/CR-5347, "Recommendations for Resolution of Public Comments on USI A-40, 'Seismic Design Criteria,'" issued June 1989, for determining the acceptability of the development of target power spectral density functions (ADAMS Accession No. 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, for determining the acceptability of the ground motion characteristics (ADAMS Accession No. ML013100232)

3.7.1.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Section 3.7.1, against the agency’s regulatory guidance to ensure that the DCA represents the complete scope of information related to this review topic. The staff evaluated DCA Part 2, Tier 2, Section 3.7.1, and DCA Part 2, Tier 1, Revision 2, Chapter 5, 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 (Yermo, Capitola, Chi-Chi, Izmit, and El Centro) and the CSDRS-high frequency (HF) seismic input response spectra and the corresponding CSDRS-HF response spectra-compatible ground motion time history (Lucerne). 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 seismic loadings.

This SER section presents the results of the staff’s technical evaluation of DCA Part 2, Tier 2, 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 substructures and subsystems.

3.7.1.4.1 Design Ground Motion

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, further states 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

DCA Part 2, Tier 2, Section 3.7.1.1.1, “Design Ground Motion Response Spectra,” applies the design response spectra, which would become the CSDRS once the NuScale DCA is certified, 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 DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, states 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 illustrates 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 certified, will be used as the CSDRS for the NuScale plant 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 were reasonable and described in sufficient detail for DC. 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

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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," compare 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). DCA Part 2, Tier 2, 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 figures in DCA Part 2, Tier 2, show that 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 DCA Part 2, Tier 2, 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.A.i 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.A.i states that, for a DCA, 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

DCA Part 2, Tier 2, 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.

3.7.1.4.4.1 Seed Records for Development of the Certified Seismic Design Response Spectra and the Certified Seismic Design Response Spectra-High Frequency Matched Time Histories

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 from 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.

3.7.1.4.4.2 Meeting the Criteria in Design-Specific Review Standard Section 3.7.1, Revision 0, Option 1, Approach 2

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.7.1.1.2, provides the following numerical values to show how the design time histories meet the acceptance criteria in DSRS Section 3.7.1, Revision 0, Option 1, Approach 2, in the frequency range from 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, are longer than 6 seconds. They 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 DCA Part 2, Tier 2, Table 3.7.1-4, "Duration of Time Histories."

- 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 DCA Part 2, Tier 2, 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/visual examination (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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.

The staff reviewed DCA Part 2, Tier 2, Section 3.7.1.1.2.4, "Results," which validated the use of the modified time histories in NuScale's analysis of the seismic Category I structures. The applicant stated that "The five CSDRS compatible time histories sets and one CSDRS-HF compatible time histories set are used for the design of the buildings, the bioshield, the fuel storage rack, and the reactor building crane." The staff noted that this may imply that the applicant did not use all the sets of time histories for the design of all the SSCs. In a letter dated September 5, 2017 (ADAMS Accession No. ML17249A965), the applicant clarified that all the SSCs have been designed using a minimum of one seed time history. The applicant also stated that the seismic Category I structures, the bioshield, the fuel storage rack, and the RXB crane have been conservatively designed using a combination of the Yermo, Capitola, Chi-Chi, Izmit, El Centro, and Lucerne seed time histories. DCA Part 2, Tier 2, Table 3.7.2-33, "Definition of

Seismic Analysis Identification Codes," shows the definition of the eight seismic analysis identification codes, and Table 3.7.2-34, "SSC Seismic Analysis Identification Code Assignments," shows which codes were used in the seismic analysis of the SSCs. The applicant further stated that the seismic input used for the design of the SSC varies based on different requirements for each system and level of conservatism.

In DCA Part 2, Tier 2, Section 3.7.1, and DCA Part 2, Tier 1, Section 5.0, 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. In addition, DCA Part 2, Tier 1, Section 3.14.1, "Design Description," states that the seismic Category I equipment withstands design-basis seismic loads without loss of its safety functions during and after an SSE. 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.

DCA Part 2, Tier 2, Table 2.0-1, clarifies that the RXB and CRB are designed for both the CSDRS and CSDRS-HF and that other seismic Category I SSCs are designed only for the CSDRS. In addition, DCA Part 2, Tier 1, Figures 5.0-3 and 5.0-4, and DCA Part 2, Tier 2, Figures 3.7.1-3 and 3.7.1-4, include notes to clarify the basis of the design-basis seismic loads for applicable SSCs. The staff reviewed the applicant's application of the CSDRS to seismic Category I SSCs and determined it was acceptable.

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 prescribed in DSRS Section 3.7.1.II.1.B.

3.7.1.4.4.3 Percentage of Critical Damping Values

DCA Part 2, Tier 2, 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 1, and provides both SSE and OBE damping values in DCA Part 2, Tier 2, Table 3.7.1-6, "Generic Damping Values for Dynamic Analysis." 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 any subsequent dynamic analysis.

3.7.1.4.4.4 Structural Damping

The applicant indicated that the safety-related structures, which are characterized as reinforced concrete structures, may experience some cracking during a seismic event; therefore, the model includes two levels of stiffness (cracked and uncracked) to account for any cracking experienced by the concrete structures. The applicant stated that reducing the stiffness of the walls and diaphragms by 50 percent for flexure and shear represents the cracked conditions. DCA Part 2, Tier 2, Table 3.7.1-7, "Effective Stiffness of Reinforced Concrete Members," provides the effective stiffness for the beams, columns, and wall and diaphragms used in the analysis of the NuScale seismic Category I and II structures.

The applicant further stated that, for the SSI analysis with cracked concrete conditions, all the structural members may not reach their cracked shear and moment values. As a result, the envelope of the member forces for the uncracked and cracked concrete, with 7-percent damping, is used for the design of the safety-related structures. Additionally, the enveloping of the results for both the cracked and uncracked reinforced concrete conditions, with 4-percent damping is used for generating the in-structure response spectra (ISRS). The staff finds the approach of enveloping results of cracked concrete with those of uncracked concrete analysis to be conservative. Hence, the staff finds this approach acceptable for the NuScale design and the choice of critical damping for structures to be acceptable because it complies with RG 1.61.

3.7.1.4.4.5 Soil Damping

In DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Figure 3.7.1-14, "Soil Shear Modulus Degradation Curves," and Figure 3.7.1-15, "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. DCA Part 2, Tier 2, Table 3.7.1-8, "Soil Shear Modulus Degradation and Strain-Dependent Soil Damping (0–120 ft)"; Table 3.7.1-9, "Soil Shear Modulus Degradation and Strain-Dependent Soil Damping (120 ft–1,000 ft)"; and Table 3.7.1-10, "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 concluded 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 design as they are consistent with the guidance in SRP Section 3.7.1.II.2.

3.7.1.4.4.6 Supporting Media for Seismic Category I Structures

In DCA Part 2, Tier 2, 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 footprints of both the RXB and CRB are rectangular and are embedded 26.2 m and 16.8 m (86 ft and 55 ft) below grade, respectively. The NuScale seismic Category I structures are assumed to be founded on competent soil or rock, which should have a shear wave velocity greater than or equal to 304.8 m/s (1,000 ft/s). The standard design considers four subgrade cases, including soft soil (Type 11), firm soil/soft rock (Type 8), rock (Type 7), and hard rock (Type 9). DCA Part 2, Tier 2, Tables 3.7.1-11 through 3.7.1-14, provide the number of layers, thickness, depth, shear wave velocity, weight density, and Poisson's ratio for each layer of the four generic soil profiles, respectively.

DCA Part 2, Tier 2, Figure 3.7.1-16, “Shear Wave Velocities for All Soil Types,” shows the shear wave velocities for the four soil profiles, ranging from 241.80 m/s (793.3 ft/s) to 2,438 m/s (8,000 ft/s) on the ground surface and reaching the bedrock at various depths. The shear wave velocity of the bedrock is assumed to be 2,438 m/s (8,000 ft/s). The four soil profiles considered in the NuScale standard design represent a wide range of soil conditions. The SSI analyses of the NuScale seismic Category I structures used the generic soil profiles in DCA Part 2, Tier 2, Tables 3.7.1-11 through 3.7.1-14.

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 DCA Part 2, Tier 2, Tables 3.7.1-15 through 3.7.1-17. For the CSDRS-HF, the applicant used only one set of compatible time histories; therefore, no averaging was performed. DCA Part 2, Tier 2, Tables 3.7.1-8 and 3.7.1-19, show the strain-compatible properties for the CSDRS-HF time histories for Soil Types 7 and 9, respectively. 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 structural foundation dimension and depth, the depth of the four 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 that the descriptive information and referenced tables and figures in DCA Part 2, Tier 2, Section 3.7.1.3, (1) contain sufficient information on the supporting media and (2) are consistent with the acceptance criteria in DSRS Section 3.7.1.II.3.

3.7.1.5 Combined License Information Items

SER Table 3.7.1-1 lists the COL information item numbers and descriptions related to the design parameters from DCA Part 2, Tier 2, Section 3.7.1.1.3, “Site-Specific Design Ground Motion,” and Section 3.7.1.3.3, “Site-Specific Soil Profile.”

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.7-1	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific SSE.	3.7.1.1.3
COL Item 3.7-2	A COL applicant that references the NuScale Power Plant design certification 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 PGA, ground velocity, and ground displacement, respectively) are consistent with characteristics values for the magnitude and distance of the appropriate	3.7.1.1.3

Item No.	Description	DCA Part 2, Tier 2, Section
	controlling events defining the site-specific uniform hazard response spectra. Additional site-specific seismic analysis is performed by the COL applicant to confirm the adequacy of the seismic input motion and deterministic soil columns used in the soil structure interaction (SSI) analysis. The FIRS is the starting point for conducting an SSI analysis and for making a one-to-one comparison of the seismic design capacity of the standard design and the site-specific seismic demand for a site. The FIRS for the vertical direction is obtained with the vertical to horizontal (V/H) ratios appropriate for the site. For deeply embedded structures, the variation of V/H spectral ratios on ground motion over the depth of the facility will be considered. In addition to the FIRS, the COL applicant will develop one or more performance-based response spectra (PBRS) at intermediate depths between the foundation and ground surface consistent with the Interim Staff Guidance (ISG)-017 (Reference 3.7.1-13). The PBRS for the vertical direction can be obtained with the appropriate V/H ratios used to develop the FIRS. The site-specific FIRS response spectra satisfy the same performance criteria as the GMRS. The GMRS are those derived from the global understanding of the site soil layers above the rock condition as determined from the site exploration activities and, therefore, are unique to a particular site.	
COL Item 3.7-3	A COL applicant that references the NuScale Power Plant certification will perform the following: • develop a site-specific strain compatible soil profile; • confirm that the criterion for the minimum required response spectrum has been satisfied; and • determine whether the seismic site characteristics fall within the seismic design parameters such as soil layering assumptions used in the certified design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity.	3.7.1.3.3
COL Item 3.7-9	A COL applicant that references the NuScale Power Plant design certification will include an analysis of performance-based response spectra (PBRS) 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 V/H spectral ratios used in establishing the site-specific foundation input response spectra (FIRS) and PBRS for the vertical direction.	3.7.1.1.3

In relation to the seismic analysis of deeply embedded nuclear structures, a COL application referencing a DC should include the PBRS established at the surface and intermediate depth(s), and the selection of the number and locations of the intermediate depths should account for the complexities of the subsurface layer profiles of the site. Additionally, the adequacy of the input ground motion and deterministic soil columns used in the site-specific SSI analysis should be demonstrated by using the PBRS at the ground surface and intermediate depth(s) as the benchmarks.

The applicant further stated that in addition to the FIRS, the COL applicant will develop one or more PBRS at intermediate depths between the foundation and ground surface consistent with DC/COL-ISG-017. The PBRS for the vertical direction can be obtained with the appropriate V/H ratios used to develop the FIRS. The site-specific FIRS satisfy the same performance criteria as the GMRS. GMRS are those spectra derived from the global understanding of the site soil layers above the rock condition as determined from the site exploration activities and therefore are unique to a particular site.

The staff finds the information to be acceptable because (1) the applicant included a COL information item (COL Item 3.7-9) that requires a COL applicant that references the NuScale Power Plant DC to include an analysis of the PBRS established at the surface and intermediate depth(s) that takes into account the complexities of the subsurface layer profiles of the site and to provide a technical justification for the adequacy of V/H spectral ratios used in establishing the site-specific FIRS and PBRS for the vertical direction and (2) the applicant's approach for ensuring consistent hazard seismic input for the SSI analysis is consistent with the regulatory guidance in DC/COL-ISG-017.

3.7.1.6 Conclusion

The staff concludes that the seismic design parameters used in the design of the SSCs for the NuScale application are acceptable and meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2; 10 CFR 52.47(a)(1); and 10 CFR Part 50, Appendix S; and the acceptance criteria in DSRS Section 3.7.1, Revision 0. The applicant meets these requirements specifically by its use of (1) acceptable smooth broadband CSDRS, (2) synthetic acceleration time histories that envelop the CSDRS and that have sufficient power in the frequency range of interest to the NuScale standard design, (3) a percentage of critical damping values that conforms to the regulatory guidance in RG 1.61, (4) four generic soil profiles (i.e., soft soil, firm soil/soft rock, rock, and hard rock) that cover a wide range of site conditions, and (5) CSDRS-HF and the associated synthetic acceleration time histories for the evaluation of the NuScale seismic Category I SSCs against high-frequency seismic motions. This ensures that the seismic design parameters are adequate for use in the seismic analysis and design of the NuScale seismic Category I SSCs to withstand CSDRS and CSDRS-HF seismic loadings.

3.7.2 Seismic System Analysis

3.7.2.1 Introduction

For the seismic design of nuclear power plants, 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 OBE and the SSE. The provisions of 10 CFR Part 50, Appendix S, for SSE ground motion require that SSCs be designed to remain functional and within applicable stress, strain, and deformation limits and that the seismic analysis must account for SSI effects and the expected duration of the vibratory motion. For the NuScale 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 provides the staff's evaluation of the methods used to perform seismic analyses and their results for seismic Category I structures and other structures of the NuScale standard design.

3.7.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 3, "Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria"; Chapter 4, "Interface Requirements"; and Chapter 5, provide the information associated with seismic system analysis. DCA Part 2, Tier 1, Sections 3.11, 3.12, and 3.13, include the design descriptions and ITAAC for the RXB, RWB, and CRB, respectively. DCA Part 2, Tier 1, Section 4.1, "Interface Requirements—Site Specific Structures," addresses site-specific structures not within the scope of the NuScale standard design. DCA Part 2, Tier 1, Table 5.0-1, specifies site parameters used in the NuScale standard design. In DCA Part 2, Tier 1, Figures 5.0-1 and 5.0-2 specify the CSDRS for all seismic Category I SSCs, and Figures 5.0-3 and 5.0-4 specify the CSDRS-HF for the RXB and CRB.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.7.2, provides information associated with the seismic system analysis, as summarized below.

The NuScale standard design includes two site-independent seismic Category I structures—the RXB and CRB. The RXB is designed for up to 12 installed NPMs. The design-basis seismic analysis is performed with all 12 NPMs in place. The applicant also discussed the effect on the structure if a seismic event were to occur during operation with less than the full complement of 12 NPMs. Because of its proximity to the RXB, the RWB is categorized as seismic Category II. NuScale designed the RWB using the same methodology as for the seismic Category I structures. The applicant discussed the potential interaction of the seismic Category II RWB with the seismic Category I RXB. The RXB includes the UHS pool, which contains a large body of water. The UHS pool consists of the reactor pool, spent fuel pool, refueling pool, and dry dock and is assumed to be full of water for 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 affect the dynamic characteristics of the building.

The applicant used the complex frequency response analysis method to analyze seismic Category I structures, including the effects of SSI. Seismic Category I structures are modeled as 3-D finite element models (FEMs). In addition to SSI analyses, the applicant analyzed structure-soil-structure interactions (SSSIs) to evaluate the potential seismic interactions between adjacent structures (i.e., the RXB and CRB, and the RXB and RWB). The results from seismic response analyses include member forces and moments, displacements, soil pressures, and nodal acceleration time histories from which the ISRS are developed. The analyses are performed in each of the three orthogonal directions of the earthquake ground motion—two horizontal and one vertical.

The design of the seismic Category I SSCs of the NuScale standard plant is based on the CSDRS, as shown in DCA Part 2, Tier 1, Figures 5.0-1 and 5.0-2, and in DCA Part 2, Tier 2, Figures 3.7.1-1 and 3.7.1-2, for the horizontal and vertical directions. Further, the seismic Category I buildings (RXB and CRB) of the NuScale standard plant are also designed for the CSDRS-HF shown in DCA Part 2, Tier 1, Figures 5.0-3 and 5.0-4, and in DCA Part 2, Tier 2, Figures 3.7.1-3 and 3.7.1-4. The seismic design of the NuScale standard plant considers a set of generic subgrade profiles ranging from soft soil to hard rock, as described in DCA Part 2, Tier 2, Section 3.7.1.3.

ITAAC: DCA Part 2, Tier 1, Chapter 3, provides the ITAAC associated with DCA Part 2, Tier 2, Section 3.7.2. These ITAAC are evaluated in Section 14.3 of this SER.

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

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

3.7.2.3 Regulatory Basis

Relevant requirements of the NRC regulations for seismic system analysis include the following:

- 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 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 SSI effects and the expected duration of the vibratory motion. If the OBE is set at one-third or less of the SSE, explicit response or design analyses are not required. 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. In addition, 10 CFR Part 50, Appendix S, requires that 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 52.47(a)(1), as it requires a DCA to include the site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters.

- 10 CFR 52.47(a)(20), as it requires a DCA to include the information necessary to demonstrate that the standard plant complies with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.

In addition, acceptance criteria and regulatory guidance associated with the review of DCA Part 2, Tier 2, Section 3.7.2, include the following:

- DSRS Section 3.7.2, Revision 0, “Seismic System Analysis,” issued June 2016

- RG 1.61, Revision 1

- 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

- DC/COL-ISG-017

3.7.2.4 Technical Evaluation

In this section, the staff describes its evaluation of the applicant's seismic analysis for the site-independent 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 II and I structures. The staff reviewed the information in DCA Part 2, Tier 2, Section 3.7.2, against the acceptance criteria of DSRS Section 3.7.2 and the RGs and ISGs 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 function during and following the earthquake.

The applicant performed seismic SSI analysis using the computer program SASSI2010. The applicant used the computer program ANSYS to capture the hydrodynamic loads of the pool water during the earthquake while accounting for the effect of fluid-structure interaction (FSI) between the water and pool structure. The analysis for 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), four generic soil profiles (soft soil, firm 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 three different building models (the RXB, CRB, and triple building), and the triple building model consisting of the RXB, CRB and RWB captures the SSSI effect.

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

3.7.2.4.1 Seismic Analysis Methods

The staff reviewed the seismic analysis methods used for the NuScale standard design in accordance with the guidance in DSRS Section 3.7.2.II.1. DCA Part 2, Tier 2, Section 3.7.2.1, discusses analytical methods, FEMs, and computer programs used for the seismic analysis. It also describes the analysis method used to capture the FSI effects between the UHS pool water and pool structure.

DSRS Section 3.7.2.II.1 provides guidance that recommends that the seismic analysis of all seismic Category I SSCs should use a suitable dynamic analysis method or an equivalent static load method. DCA Part 2, Tier 2, Section 3.7.2.1, contains information with respect to the seismic analysis methods applied to the NuScale seismic Category I SSCs. The applicant indicated that the seismic analysis of all seismic Category I SSCs used either linear equivalent static analysis, linear dynamic analysis based on complex frequency response methods, or nonlinear analysis. The applicant also indicated that it primarily analyzed the two site-independent seismic Category I structures (RXB and CRB) using the time history method and that, for systems and components, it developed and used ISRS to calculate forces and moments using the response spectrum analysis method. Further, the NPM and RXB cranes are analyzed using the time history analysis method and the response spectrum analysis method, respectively.

The staff reviewed the descriptions contained in the DCA and found them acceptable because the seismic analysis methods used for NuScale seismic Category I SSCs are generally recognized methods and meet the acceptance criteria in DSRS Section 3.7.2.II.1.

3.7.2.4.1.1 Dry Dock Modeling

In DCA Part 2, Tier 2, Section 3.7.2.1.3, the staff notes that the design-basis seismic demand analysis assumes that the dry dock is full of water and part of the UHS, and the nominal water level is set at elevation 28.7 m (94 ft). In DCA Part 2, Tier 2, Section 9.1.3, the staff notes that the dry dock can be drained partially or completely to support plant operations and that a failure of the dry dock gate while the dry dock is empty could result in a decrease in water level at the UHS pool by less than 3.66 m (12 ft). Because the dry dock contains a large body of water, draining a large mass of water could affect the dynamic characteristics of the SASSI and ANSYS models, thereby potentially affecting the seismic demand, which is based on the assumption of a full dry dock.

To address this issue, the applicant revised DCA Part 2, Tier 2, Section 3.7.2.1.3, summarizing the results of the sensitivity studies performed for evaluating the effect of an empty dry dock on the design-basis seismic demand. Three separate SASSI models were created for this purpose involving three different NPM stiffnesses—the nominal NPM stiffness, 1.3 times the nominal, and the nominal divided by 1.3. Each of these three SASSI models uses Soil Type-7, CSDRS-compatible Capitola input motion, cracked concrete condition, 4-percent structural damping for ISRS generation, and 7-percent structural damping for forces and moments calculation. The applicant calculated the maximum forces and moments in the four RXB exterior walls and in the four walls around the dry dock, the lug support reactions at the 12 NPMs, and forces and moments in one pilaster in the north wall at column line RX-4, for the empty dry dock condition and compared them with the corresponding design capacities based on the full dry dock condition. In addition, the applicant provided comparisons of ISRS at different floor elevations and other equipment locations including the RXB crane wheels. The applicant also provided information about the structural design criteria for the dry dock gate.

The staff reviewed the comparisons of the structural seismic demands and design capacities and found that the empty dry dock condition is bounded by the RXB design, which is based on the full dry dock condition. In addition, all ISRS from the empty dry dock condition are either bounded by or are within approximately 10 percent of the full dry dock condition, and therefore, the design-basis ISRS for equipment qualifications based on full dry dock condition remain valid. The staff noted that the dry dock gate is designed to withstand the effect of the SSE, and therefore, there would be no adverse seismic interactions between the dry dock gate and adjacent seismic Category I SSCs. Further, during the regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML19098A162), the staff reviewed the design evaluation results for the dry dock gate connection to the RXB pool walls and confirmed that the connection possesses greater capacity than the design basis demands.

The staff found that the applicant has adequately demonstrated that the design-basis seismic demands based on the full dry dock condition bound the demands computed based on the empty dry dock condition.

3.7.2.4.1.2 Fluid-Structure Interaction Correction Factor

In DCA Part 2, Tier 2, Section 3.7.2.1.3.4, the applicant discussed analysis methods for the UHS pool subjected to the design-basis earthquake ground motion. The UHS pool contributes a large amount of weight to the global mass of the RXB and affects the dynamic characteristics of the building. The RXB SASSI2010 model addresses the hydrodynamic loads caused by the pool water mass during the earthquake by assigning lumped masses of water on the pool walls and foundation nodes that are in contact with the pool water. These lumped nodal masses are then multiplied by the nodal accelerations from dynamic analysis of the SASSI2010 model to develop equivalent static loads on the pool walls and foundation. However, the SASSI2010 computer program does not have the capability for explicit fluid element formulation to accurately compute the hydrodynamic effects of UHS pool water during the design-basis earthquake.

To assess the hydrodynamic effects more accurately, the applicant developed an RXB model in ANSYS that uses fluid elements to capture the FSI effects analytically. The results from the ANSYS FSI model are then compared to the results from the SASSI2010 model, and a correction factor, reflecting the difference in results from the SASSI2010 and ANSYS models, is applied to the SAP2000 RXB model as equivalent static loads to account for the missing FSI effects. The applicant determined that an average pressure of 29 kPa (4.2 psi) is needed to be added to the pool walls and foundation to account for the missing FSI hydrodynamic effects. However, instead of directly applying 29 kPa (4.2 psi) to the pool walls and foundation, the applicant took the approach of amplifying the gravity load by a factor of 0.28g. The applicant provided an evaluation that demonstrates that an additional gravity loading increased by a factor of 0.28g creates load demands for the pool walls and foundation that are higher than the demands from the direct 29 kPa (4.2 psi) average hydrostatic pressure. In addition, the applicant included COL Item 3.7-12, which requires a COL applicant that references the NuScale DC to perform an analysis that uses site-specific soil and time histories to confirm the adequacy of the FSI correction factor.

The staff reviewed the applicant's approach to accounting for the FSI effects on seismic demands on pool walls and foundation and found it to be acceptable because the seismic load demands for the pool walls and foundation generated by additional gravity loading increased by a factor of 0.28g are more conservative than the corresponding load demands generated by 29 kPa (4.2 psi) average hydrostatic pressures on the pool walls and foundation and because the applicant provided a COL information item to ensure the adequacy of the site-specific FSI correction factor. Further, during the regulatory audit from December 3 to 7, 2018, the staff reviewed and verified the calculations used in the development of the FSI correction factor and gravity load factor method that the applicant used to implement the FSI correction factor in the RXB design.

3.7.2.4.1.3 Verification and Validation of Computer Programs

DCA Part 2, Tier 2, Section 3.7.5, describes the computer programs used in the analysis of the site-independent NuScale seismic Category I and Category II structures. During the regulatory audit from December 3 to 7, 2018, the staff reviewed the V&V of the SASSI2010 computer program used for determining the seismic demand of the seismic category 1 SSCs.

The V&V problems considered by the applicant included three different categories: (1) NuScale specific examples, (2) vendor-provided examples, and (3) examples created for comparison. The applicant summarized the results from an extensive set of studies conducted for SASSI2010 V&V to demonstrate that the parameters used in the NuScale design-basis seismic demand calculations are within the range of applicability for SASSI2010. The specific parameters that the applicant tested include the following:

- mesh sensitivity—evaluation of solutions for different mesh sizes of finite elements

- aspect ratio—evaluation of solutions for the maximum finite element aspect ratio used
- Poisson's ratio—evaluation of solutions for the maximum Poisson's ratio used
- frequencies of analysis—demonstration that the frequencies of the analysis used are adequate
- impedance functions—validation of the impedance functions or transfer functions (TFs) against the benchmark solutions for frequencies up to 50 Hz for embedded structures
- extended subtraction method (ESM)—adequacy of the ESM as compared to the direct method (DM)
- nonvertically propagating shear waves—evaluation of solutions for nonvertically propagating shear waves and determination of whether this is an important effect that should be included in the NuScale seismic analysis
- number of soil layers—confirmation that the number of soil layers used in the NuScale analysis is within the maximum soil layers validated for SASSI2010
- number of interaction nodes—confirmation that the number of interaction nodes used in the NuScale analysis is within the maximum interaction nodes validated for SASSI2010
- interpolated TFs—validation of the interpolation methodology used in SASSI2010
- other important parameters used in NuScale seismic analysis, including the following:
  - validation of kinematic (wave scattering) SSI solution;
  - validation of element dynamic properties and stress calculations (a 3-D eight-node solid element, 3-D beam element, 3-D spring element, and 3-D thick shell element)
  - validation of symmetric and antisymmetric boundary conditions to analyze a half-model
  - validation of postprocessing for the generation of TFs, maximum accelerations, acceleration time histories, and acceleration response spectra

The applicant also provided three categories of acceptance criteria used for SASSI2010 V&V: (1) the numerical accuracy criterion based on the requirement that the difference in pertinent response values is less than 5 percent, (2) the good agreement criterion based on numerical matching against closed-form solutions, analytical results, or experimental test data, and (3) the expected behavior criterion based on basic knowledge and sound engineering judgment.

The staff found the applicant's V&V for the computer code SASSI2010 to be acceptable because, through an extensive set of V&V sample analyses, the applicant demonstrated that the values of parameters used in computing the NuScale design-basis seismic demands are within the range of applicability for SASSI2010. The scope of the V&V example problems and tested parameters is sufficiently comprehensive, and the acceptance criteria used by the applicant are consistent with the generally accepted industry practice.

3.7.2.4.1.4 Reactor Building and NuScale Power Module Support Interface Loads

DCA Part 2, Tier 2, Appendix 3A, references TR-0916-51502-P, "NuScale Power Module Seismic Analysis," issued April 2019, (ADAMS Accession Nos. ML19094A021 (proprietary) and ML19093B850 (nonproprietary)), for NPM seismic analysis and indicates that analysis of the NPM subsystem was performed using a detailed 3-D NPM model with acceleration time histories from the SASSI2010 RXB model as the input to capture the coupling effects between the RXB and NPMs. The RXB-NPM interface and NPM specific analyses use a detailed NPM beam model generated by adding mass and spring elements to create an FSI response system that is equivalent to a 3-D NPM model and pool bay. The development and validation of the detailed NPM beam model are described in TR-0916-51502. The RXB model that uses the detailed NPM beam model is structurally similar to the SASSI2010 model used for RXB seismic analysis; however, since fluid mass has been added to the detailed NPM beam model, an enhanced methodology for modeling hydrodynamic mass in the pool area was used.

The RXB-NPM interface analysis uses Soil Type 7 and the CSDRS-compatible Capitola input motion and takes into account two different stiffness variations of the RXB concrete (uncracked and cracked) and three different stiffness variations of the NPM (the nominal stiffness, 1.3 times the nominal, and the nominal divided by 1.3). At the interface between the NPM and the RXB, the design loads for the skirt supports are defined as the envelope of the results from the SASSI2010 RXB model and ANSYS 3-D NPM model. The lug supports are designed for a generic capacity in a detailed submodel and checked against the reaction forces from the SASSI2010 RXB model and ANSYS 3-D NPM model as discussed in Appendix 3B.2.7 of the DCA Part 2, Tier 2. The maximum seismic forces on the NPM lug restraints and skirts, shown to be bounded by the corresponding design capacities, are provided in Table 3B-28 of the DCA Part 2, Tier 2.

The staff reviewed the information pertaining to the RXB-NPM support interface loads and found them to be acceptable because the controlling seismic demands from the SASSI2010 RXB model and ANSYS 3-D NPM model at the NPM supports are bounded by the corresponding design capacities for these supports and because COL Item 3.7-10 requires a COL applicant referencing the NuScale DC to ensure that site-specific seismic demands at the NPM supports are bounded by the corresponding DC demands.

3.7.2.4.2 Natural Frequencies and Responses

The staff reviewed the natural frequencies and responses of the NuScale standard plant structures provided by the applicant. DCA Part 2, Tier 2, Section 3.7.2.2, provides information on the dynamic modal properties of the models used in the analysis of the seismic Category I structures (RXB and CRB), including the natural frequencies and modal mass ratios. Because SASSI2000 uses a complex frequency response analysis method, the applicant used the corresponding SAP2000 models with a fixed-base boundary condition to generate the dynamic modal properties. The staff notes that the SAP2000 models for the RXB and CRB serve as the baseline for the corresponding SASSI2010 and ANSYS building models. The applicant also provided seismically induced accelerations, displacements, forces, moments, soil pressures, and ISRS, at key locations of the RXB and CRB necessary for structural design evaluation in DCA Part 2, Tier 2, Sections 3.8.4 and 3.8.5, and equipment qualification in Sections 3.9 and 3.10. The staff found the information provided in DCA Part 2, Tier 2, Section 3.7.2, concerning the dynamic modal properties and responses of the NuScale seismic Category I structures to be acceptable because the scope and nature of the information provided are consistent with the acceptance criteria of DSRS Section 3.7.2.II.2.

3.7.2.4.3 Procedures Used for Analytic Modeling

The staff reviewed the criteria and procedures used in the analytical modeling for seismic systems analysis in accordance with the guidance in DSRS Section 3.7.2.II.3. In DCA Part 2, Tier 2, Sections 3.7.2.1 and 3.7.2.3 describe methods and procedures used for analytical modeling of seismic Category I structures. The staff reviewed various aspects of the analytical modeling involved in the NuScale seismic demand calculations, such as the mesh discretization, finite element aspect ratios, passing and cutoff frequencies, NPM beam model validation, NPM support conditions, rigid spring elements, and SAP2000 and SASSI2010 model comparisons.

3.7.2.4.3.1 Mesh Discretization

In modeling structures using finite elements for dynamic analysis, the discretization should be adequately refined to sufficiently capture the frequency contents of the ground motion in the structural response. DSRS Section 3.7.2.II.3 specifies that the element mesh size should be selected on the basis that a further refinement has only a negligible effect on the solution results. In DCA Part 2, Tier 2, Section 3.7.2.1.3, the applicant described the mesh refinement study performed for defining the element size used in the RXB and CRB standalone models and the triple building model. The applicant provided the following information:

- Meshing of the area elements was done with the SAP2000 model by defining a maximum element size in each direction; the aspect ratios of these elements were kept as low as possible, and internal sharp angles were avoided. The heights of the soil elements were determined based on one-fifth of the wave length. For a mesh sensitivity study, meshes for both the RXB and CRB models were refined further by dividing each side of the area elements into two, thus breaking each area element into four elements. Static analysis cases of 1g loading in the X, Y, or Z directions were used to make comparisons. The study indicates that the effects of further mesh refinement on the structural responses are negligible for both local and global responses.

- Modal analysis was performed and showed minor changes in the natural frequencies and their mass participation ratios, which indicates that the other dynamic characteristics of the building models would not change with mesh refinement. To show that mesh refinement does not have a major impact on ISRS, the ISRS from the CSDRS compatible Capitola ground motion and the ISRS from the CSDRS-HF compatible Lucerne ground motion were compared at several key locations. These comparisons were between the RXB and CRB standalone SAP2000 model used in design-basis seismic demand calculations and the refined-mesh models. The results indicate that further mesh refinement has an insignificant impact on the ISRS. The triple building model has the same mesh as the standalone model, and the SSSI effects are not expected to change with mesh refinement; therefore, no mesh sensitivity analysis was performed for the triple building model.

The staff reviewed the results of the sensitivity study as discussed above and concluded that it has adequately refined the discretization of finite elements used in the seismic analysis of seismic Category I structures (RXB and CRB) to capture the frequency contents of the applied ground motion and therefore is acceptable. The study has shown that further refinement does not affect the structural response according to the guidance in DSRS Section 3.7.2.II.3. Further, during the regulatory audit from December 3 to 7, 2018, the staff reviewed the detailed calculations and conclusions from the mesh sensitivity study and confirmed that they are consistent with the information described in DCA Part 2, Tier 2, Section 3.7.2.

3.7.2.4.3.2 Presence of Coarse Finite Elements

In DCA Part 2, Tier 2, Section 3.7.2.1.1.3, the applicant described cut-off frequencies used for the SASSI2010 building models. For the analysis of Soil Types 7, 8, and 11 with the CSDRS, the cutoff frequency was established at 52 Hz and, for the analysis with the rock profiles (Soil Type 7 and 9) with the CSDRS-HF, the cutoff frequency was established at 72 Hz. The applicant stated that the building models have element sizes that are similar to the 1.905-m (6.25-ft) layers that were used to determine the wave passage frequency of the soil and that there are instances where development of the model required individual elements to have a dimension as large as 3.7 m (12 ft) in the RXB and as large as 6 m (20 ft) in the CRB. Further, the applicant stated that, since the typical element size is approximately 1.8 m (6 ft), the wave passage frequencies of both buildings are above the cutoff frequencies used for the analysis.

The staff reviewed the applicant's information pertaining to cutoff frequencies associated with the models with presence of coarse finite elements and found it to be acceptable because (1) the membrane elements with a maximum dimension of 6 m (20 ft) used in the CRB model are nonstructural and are used only to capture the wind loads for the steel-framed structure, and (2) the basemat solid elements with a maximum dimension of 3.66 m (12 ft) used in the RXB model are isolated and limited in number (24 elements) and therefore would not affect the wave passage frequencies determined based on an average element size of 1.905 m (6.25 ft).

3.7.2.4.3.3 Rigid Spring Constants Used in Reactor Building and Control Building Models

In DCA Part 2, Tier 2, Section 3.7.2.1.3.1, the applicant stated that the rigid springs have a zero length and have a stiffness value large enough to simulate rigid connection and that the large stiffness used is arbitrarily chosen to be 1.75×10^{12} newtons per meter (N/m) (1×10^{10} pounds per inch (lb/in.)) in the three global directions. To confirm the adequacy of the number (1.75×10^{12} N/m (1×10^{10} lb/in.)) chosen for the "rigid" spring constant, the applicant performed a sensitivity analysis by increasing the stiffness of the rigid springs by an order of magnitude (i.e., to 1.75×10^{13} N/m (1×10^{11} lb/in.)) and by comparing the results with those obtained from the base case (i.e., with the rigid spring stiffness equaling 1.75×10^{12} N/m (1×10^{10} lb/in.)). For this study, the applicant used the RXB model with cracked concrete properties, 7-percent concrete damping, Soil Type 7, and the Capitola input motion.

Comparisons of TFs and ISRS showed that increasing the rigid spring stiffness has no discernible effect on the TFs or ISRS. A comparison of the sums of the maximum spring forces shows that the total changed by 0.17 percent, and comparisons of the maximum stresses, forces, and moments in typical solid, beam, and shell elements indicated that the average change over all the elements is less than 0.3 percent. Similar findings were observed from a sensitivity analysis with the CRB model. Based on the evaluation of the results from the sensitivity study, the staff found that the value of 1.75×10^{12} N/m (1×10^{10} lb/in.) for the spring constant is sufficiently large to model the rigid soil springs connecting the basemat and backfill to the free field soils for the RXB and CRB SASSI models.

3.7.2.4.3.4 Comparison of SAP2000 and SASSI2010 Models in Calculating Structural Frequencies

The SASSI2010 model used in NuScale design seismic analysis was obtained by converting the SAP2000 model. In DCA Part 2, Tier 2, Section 3.7.2.1.3.6, the applicant discussed comparison of the fixed-base modal frequencies from the two models to verify that the SAP2000 model had been converted accurately into the SASSI2010 model. The natural frequencies of the fixed-base SAP2000 model are calculated by a modal frequency analysis. The SASSI2010 analysis does not perform modal analysis; however, the major vibration frequencies can be estimated by inspecting the peaks of acceleration TFs. The staff reviewed the information provided by the applicant and found that SAP2000 modal frequencies are close to the corresponding SASSI2010 frequencies estimated from the TF peaks with a maximum difference of about 6 percent, which implies that the mass and stiffness of the structures in the SAP2000 have been closely duplicated in the SASSI2010 model.

The staff's review also focused on how the backfill soil is accounted for in the analysis. In DCA Part 2, Tier 2, Section 3.7.2.1.3.6, the applicant stated that, for both the SAP2000 and SASSI2010 fixed-base analyses, the backfill soil is included as solid elements surrounding the buildings and is measured 7.6 m (25 ft) outward from the exterior walls and extends from the bottom of the RWB, RXB, and CRB basemats to the ground surface. The applicant further stated that the properties of Soil Type 11 are used to model the backfill soil because it has an average shear wave velocity of 234.1 m/s (768 ft/s) for the upper 25.9 m (85 ft) of soil, which is close to a typical backfill soil shear wave velocity of 244 m/s (800 ft/s). The staff found that the applicant's method of modeling backfill soil is acceptable because it closely captures the actual configuration of the backfill surrounding the buildings and Soil Type 11 represents typical material properties of the backfill soil.

3.7.2.4.3.5 Validation of the Simplified NuScale Power Module Beam Model

DCA Part 2, Tier 2, Section 3.7.2.1.3.2, describes the NPM beam model included in the RXB SAP2000 and SASSI2010 models. The applicant indicated that the simplified NPM beam model used in the SAP2000 and SASSI2010 building models is derived from the corresponding NPM detailed 3-D model developed in ANSYS. It is to be noted that the simplified NPM beam model is distinct from the detailed NPM beam model discussed in Section 3.7.2.4.1 of this report under the topic of "Reactor Building and NuScale Power Module Support Interface Loads." The detailed NPM beam model is used specifically for developing the interface time histories for use in the seismic design of the NPM and the NPM supported components, while the simplified NPM beam model is used for seismic design of the rest of the SSCs including the RXB and CRB. To validate the NPM beam model, the applicant performed a modal analysis in three directions to tune the beam model to match the detailed 3-D model response and provided a table in TR-0916-51502-P that compares the dynamic properties of the NPM beam model and the 3-D model. The staff reviewed the applicant's validation of the NPM beam model used in SAP2000 and SASSI2010 building models discussed in DCA Part 2, Tier 2, Section 3.7.2 and TR-0916-51502-P. The staff found the validation to be acceptable because the NPM beam model was developed such that it has dynamic compatibility with the original NPM 3-D model developed in ANSYS. Further, during the regulatory audit from December 3–7, 2018, the staff reviewed the detailed calculations and conclusions from the applicant's model validation analysis and confirmed that they are consistent with the information provided in the DCA and TR-0916-51502-P.

3.7.2.4.4 Soil-Structure Interaction Analysis

The staff reviewed the modeling method used in the seismic system analysis to account for the SSI effects in accordance with guidance in DSRS Section 3.7.2.II.4. DCA Part 2, Tier 2, Section 3.7.2.4, states that the SASSI2010 computer program is used for the SSI and SSSI analysis of seismic Category I and II structures. SASSI is a linear analysis code that performs time history analysis in the frequency domain using a substructuring technique. DCA Part 2, Tier 2, Section 3.7.2.1, also provides information on SSI analysis.

3.7.2.4.4.1 Benchmarking Extended Subtraction Method with Direct Method

The direct method (DM) corresponds to a theoretically correct SSI model for the excavated soil volume. However, the DM analysis is computationally intensive and, to reduce computational time in the design-basis seismic demand analysis, the applicant used a simplified method, called the 7P Extended Subtraction Method (7P ESM), which assumes only the nodes on the seven planes (the four sides of the excavated volume, and the top, bottom, and middle horizontal planes) act as the interaction nodes. To evaluate the adequacy of 7P ESM, the applicant conducted a sensitivity study and compared acceleration TFs and other seismic responses from DM and 7P ESM including structural forces and moments and ISRS at key locations of the RXB and CRB.

Application of DM for SASSI analysis of the full RXB model required the use of interaction nodes that exceeded the SASSI2010 program limit of 20,000 nodes. Therefore, the applicant used a half model to obtain the results by DM. The applicant reported that the ISRS calculated by the RXB and CRB 7P ESM models are within 15 percent of those calculated by DM, and that exceedances were observed at narrow frequency bands around ISRS peak locations. The applicant also reported that the TF shapes show a good agreement between 7P ESM and DM except at a few frequencies where some minor differences were observed including spurious peaks, but they did not significantly affect the analysis results. The applicant stated that adding a frequency point or shifting the frequency close to a spike location usually eliminates the spurious spike. The applicant also compared structural forces and moments from the two methods and the comparison showed negligible differences. Further, the applicant included COL Item 3.7-15 stating that a COL applicant that references the NuScale Power Plant DC will determine the appropriate site-specific number of interaction planes for SSI analysis.

The staff reviewed the information from the applicant's sensitivity study on 7P ESM and DM models and found the applicant's use of 7P ESM in SSI analysis of the NuScale seismic Category I structures to be acceptable because the results from 7P ESM are close to those from DM and any identified differences from the two methods are negligibly small and would not affect the standard design. The COL item will further ensure the adequacy of 7P ESM for the site-specific conditions or an appropriate number of interaction planes will be used for site-specific SSI analysis.

3.7.2.4.4.2 Review of Transfer Functions

DCA Part 2, Tier 2, Section 3.7.2.1.1.3, describes SASSI2010 models used to compute seismic demands for the RXB and CRB and discusses TFs generated from these models. The staff reviewed the TFs because they provide an indication whether the numerical models and their implementation in SSI analyses are adequate. The applicant provided the plots of acceleration TFs and ISRS at key locations in the RXB and CRB from the SSI analysis. The applicant reported spurious spikes at few frequencies in the TF plots and indicated that the corresponding seismic input at those frequencies is insignificant and thus the corresponding ISRS do not exhibit spurious peaks. Based on the ISRS examination and nonexistence of any spurious peaks in the ISRS, the staff concluded that the spurious spikes in TFs have no effect on the design of the RXB and CRB or on the ISRS for equipment qualifications.

3.7.2.4.4.3 Effect of Potential Soil Separation

DSRS Section 3.7.2.II.4 provides guidance that an SSI analysis should consider the effects of potential separation or loss of contact between the structure and the soil during an earthquake. DCA Part 2, Tier 2, Section 3.7.2.1.1.3, discusses potential soil separation and its consideration in the SASSI2010 RXB and CRB models. The applicant provided information summarizing the soil-separation sensitivity study performed and its effects on the seismic demands for the RXB and CRB. The applicant stated that, to model soil separation, the Young's modulus of the backfill elements down to a depth of 7.6 m (25 ft) was decreased to approximately zero. Section 5.1.9 of ASCE 4-16 (2016), "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," discusses a method to address soil separation by assuming no connectivity between the structure and lateral soil over the upper half of the embedment, or 6 m (20 ft), whichever is less. The staff considers a soil separation depth of 7.6 m (25 ft) to be acceptable because it exceeds the minimum depth of 6 m (20 ft) specified by a recognized industry standard. The applicant evaluated acceleration TFs and ISRS at key locations and forces and moments for important structural members. The applicant reported that in some instances the ISRS and design loads increased because of soil separation effects. The applicant also provided COL Item 3.7-11 requiring a COL applicant that references the NuScale DC to perform site-specific analysis that assesses the effects of soil separation and to confirm that site-specific ISRS under soil separation are bounded by the corresponding design-basis ISRS of the NuScale DC.

The staff reviewed the results from the applicant's sensitivity study and noted that soil separation resulted in exceedance of some design-basis demands computed using the intact condition (i.e., no soil separation), both in structural forces and moments and in ISRS for equipment qualifications. However, the applicant demonstrated that the exceedances in the structural forces and moments are covered by available design margins in terms of the D/C ratios. Further, when the ISRS from the soil-separated case exceeded the design-basis ISRS, the applicant enhanced the design-basis ISRS to account for the identified exceedances. The staff found the applicant's approach to account for potential effects of soil separation on the design-basis ISRS and structural forces and moments for the NuScale seismic Category I structures to be acceptable because the approach is consistent with the acceptance criteria in DSRS Section 3.7.2.II.4 and the results from the soil-separation case are generally close to those from the intact case; any identified local exceedances due to soil separation are accounted for either by enhancing the design-basis ISRS such that the exceedances are bounded or through available design margins in terms of D/C ratios for affected structural members. The COL item will further ensure that the site-specific ISRS under soil separation are bounded by the corresponding design-basis ISRS of the NuScale DC.

3.7.2.4.4.4 Soil-Structure Interaction Analysis of Deeply Embedded Structure

DSRS Section 3.7.2.II.4 provides guidance to consider uncertainties associated with SSI analysis of deeply embedded structures. The applicant provided information on the sensitivity study performed for the effects of nonvertically propagating seismic waves on seismic demand calculations. The objective of the SSI analysis study with nonvertically propagating (or inclined) waves was to compare the SSI results with those of the design-basis case, which uses conventional, vertically propagating shear (SV and SH) and P-waves for the seismic input. It is known that a body wave (either SV- or P-wave) propagating at an inclined angle will include both horizontal and vertical motions in the free field, whereas an inclined SH-wave generates only horizontal motion in the free field.

The applicant compared the ISRS from the sensitivity study with the design-basis ISRS and reported exceedances at a few locations at narrow frequency bandwidths. The applicant explained that these exceedances result because the free-field within motions for inclined waves at the foundation level exceed the corresponding motions from the CSDRS with vertically propagating waves, resulting in an effective SSI input motion that is higher than the CSDRS input motion. Therefore, the applicant concluded that combining the coupling responses from nonvertically propagating waves (SV or P) can lead to overly conservative and incorrect structural responses. The applicant also reported that nonvertically propagating SH-waves have an insignificant effect on the RXB torsional responses and that increasing RXB design forces by 5 percent to account for accidental torsion will conservatively cover any additional torsional responses from inclined SH-waves. The applicant also proposed COL Item 3.7-13, which requires a COL applicant referencing the NuScale Power Plant DC to perform site-specific analysis that assesses the effects of nonvertically propagating seismic waves on the free-field ground motions and seismic responses of seismic Category I SSCs for the site conditions.

The staff reviewed the applicant’s evaluation of the effects of nonvertically propagating seismic waves on seismic demand calculations and found them acceptable because (1) combining the coupling responses from nonvertically propagating SV- and P-waves may lead to overly conservative structural responses because the corresponding effective SSI input motion at the foundation level exceeds the design-basis CSDRS input motion, (2) additional torsional responses from nonvertically propagating SH-waves are insignificant and are covered by the provisions of accidental torsion, and (3) the applicant provided a COL information item to cover any site-specific seismic issues associated with nonvertically propagating seismic waves. The staff recognizes that seismic SSI analysis of deeply embedded nuclear structures subject to nonvertically propagating seismic waves is a relatively new area of concern to the nuclear industry, and the level of its understanding and sophistication of implementation are still evolving. Therefore, any site-specific analysis that assesses the effects of nonvertically propagating seismic waves on the seismic demand of nuclear structures should employ the analysis techniques available at the time of the COL application.

3.7.2.4.4.5 Design-Basis Seismic Soil-Structure Interaction Analysis Cases

In DCA Part 2, Tier 2, Section 3.7.1.1.1, and in DCA Part 2, Tier 1, Section 5.0, the applicant stated that the NuScale seismic design basis for all seismic Category I SSCs is the CSDRS. The applicant further stated that it expanded the seismic design basis for the seismic Category I structures, the RXB and CRB, to include the CSDRS-HF to broaden the site applicability for these structures. In DCA Part 2, Tier 2, Section 3.7.2.4, the applicant indicated that 540 SSI analysis cases with five CSDRS-compatible time history inputs and 72 SSI analysis cases with one CSDRS-HF-compatible time history input are considered for seismic demand calculations for the RXB and CRB. The applicant also indicated that all NuScale seismic Category I SSCs are not analyzed for the same set of SSI analysis cases. The seismic analysis cases used to establish seismic demands for the SSCs are discussed in DCA Part 2, Tier 2, Section 3.7.2.4.6. The detailed modeling and analysis parameters used to establish the seismic demand for the SSCs are shown in DCA Part 2, Tier 2, Tables 3.7.2-33 through 3.7.2-35. The response parameters considered in developing seismic analysis cases for the RXB and CRB include (1) RXB Standalone Structural Response, (2) RXB Triple Building Structural Response, (3) RXB Standalone ISRS, (4) RXB Triple Building ISRS, (5) NPM ISRS, (6) CRB Standalone ISRS, (7) CRB Standalone Structural Response, (8) CRB Triple Building Structural Response, and (9) CRB Triple Building ISRS. The staff reviewed the information and found that the applicant used an approach consistent with the guidance in DSRS Section 3.7.2 in developing seismic

SSI analysis cases to establish the design-basis seismic demands for the RXB, CRB, and other seismic Category I SSCs. Therefore, the staff determined the applicant's seismic SSI cases are acceptable.

3.7.2.4.5 Development of In-Structure Response Spectra

The staff reviewed the procedures and methods used in developing ISRS, in accordance with DSRS Section 3.7.2.II.5 and RG 1.122. These documents provide guidance and criteria for methods acceptable to the staff for developing two horizontal and vertical ISRS from the response time histories.

The staff reviewed DCA Part 2, Tier 2, Section 3.7.2.5, for procedures used in developing the ISRS for seismic Category I structures. In this section, 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 applicant then obtained the total ISRS at each location by applying the square root of the sum of the squares (SRSS) method to the three codirectional ISRS. The ISRS from different analysis cases are enveloped as appropriate, and then the peaks in the total ISRS are widened by ±15 percent on the frequency axis. The ISRS are computed at damping values of 2, 3, 4, 5, 7, and 10 percent. The staff found that the applicant's process for the development of the ISRS from time histories, computation of the ISRS at a minimum number of frequencies, combining the ISRS at each location using the SRSS method, and the 15-percent widening of the peaks in the total ISRS, conform to the guidance in RG 1.122 and 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 of earthquake ground motion in accordance with the guidance in DSRS Section 3.7.2.II.6. The DSRS references RG 1.92, for methods acceptable to the staff for combining three spatial components of seismic responses.

In DCA Part 2, Tier 2, 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 time histories, which are applied to the building models as input to the SASSI2010 analysis, and that the three codirectional responses for the structure are combined using the SRSS method in conformance with RG 1.92, Revision 3. The staff found the applicant's method of combining the three spatial components of seismic responses using the SRSS method to be in conformance with the guidance in RG 1.92 and therefore 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, including 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 damped linear systems.

In DCA Part 2, Tier 2, Section 3.7.2.7, the applicant stated that the analysis of the seismic Category I structures, the RXB and CRB, does not use modal combination. Rather, the analysis applies the SASSI2010 code that uses time history analysis in the frequency domain in which the equations of motion are solved for the soil and structural elements. The staff found that, since the applicant does not use a method based on modal combination, no further review of the combination of modal responses is needed.

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

The staff reviewed the methods the applicant used to assess nonseismic Category I structures to determine whether their failure under 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 DCA Part 2, Tier 1, Section 4.1, the applicant stated that failure of any of the site-specific structures not within the scope of the NuScale Power Plant certified design will not cause any of the seismic Category I structures within the scope of the NuScale Power Plant DC to fail. In DCA Part 2, Tier 2, Section 3.7.2.8, the applicant described the criteria used to provide reasonable assurance that the failure of nonseismic Category I structures under the effect of a design-basis seismic event does not impair the integrity of an adjacent seismic Category I SSC. In DCA Part 2, Tier 2, Section 3.2, the applicant identified seismic Category II SSCs as those SSCs that perform no safety-related function but whose structural failure or adverse interaction could degrade the function or integrity of a seismic Category I SSC to an unacceptable level or could result in incapacitating injury to occupants of the control room during or following an SSE. Because such SSCs are not required to remain functional, the seismic Category II classification is applied only to the portions of systems with a potential for adverse interaction with a seismic Category I SSC.

In DCA Part 2, Tier 2, Section 3.7.2.8, the applicant indicated that the upper portion of the CRB located above elevation 37 m (120 ft) and the RWB adjacent to the RXB are classified as seismic Category II. These seismic Category II structures (the upper portion of the CRB and the RWB) are designed for the CSDRS and CSDRS-HF, the standard for seismic Category I, to ensure that there are no unacceptable interactions. Specifically, the results from the seismic analysis performed using the triple building model indicate no unacceptable seismic interaction between the RWB and RXB. In the same DCA section, the applicant described the turbine generator buildings, central utilities building, and annex buildings, which DCA Part 2, Tier 2, Section 3.2.1, lists as seismic Category III, as structures adjacent to seismic Category I structures. The applicant stated that the turbine generator buildings, central utility buildings, and annex buildings are not included in the scope of the NuScale certified design and are provided for conceptual design information only. The staff notes that DCA Part 2, Tier 2, also includes three COL information items (COL Items 3.3-1, 3.4-6, and 3.7-4) to ensure that nearby structures will not adversely affect the RXB or the seismic Category I portion of the CRB and that analysis and justification will be handled on a site-specific basis to ensure that nonseismic Category I structures will not impair the integrity of an adjacent seismic Category I structure. Based on its review, the staff finds that the applicant’s evaluation of potential interaction of nonseismic Category I structures with seismic Category I SSCs is acceptable because the method is consistent with the acceptance criteria in DSRS Section 3.7.2.II.8.

3.7.2.4.9 Effects of Parameter Variations on Floor Response Spectra

3.7.2.4.9.1 Effect of Structural Stiffness Variations

The staff reviewed the applicant’s consideration of the effects of parameter variations on floor response spectra in accordance with the guidance in DSRS Section 3.7.2.II.9. DSRS Section 3.7.2.II.9 refers to the acceptance criteria in DSRS Section 3.7.2.II.5 on ISRS and to

DSRS Section 3.7.2 II.3 for addressing the effect of potential concrete cracking on the stiffness of the concrete structures. DSRS Section 3.7.2.II.5 references RG 1.122, and DSRS Section 3.7.2.II.3 references ASCE 43-05 (2005), "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," for acceptable stiffness reduction factors for cracked concrete members (e.g., 0.5 for cracked walls for flexure and shear).

The staff reviewed information in DCA Part 2, Tier 2, Section 3.7.2.9, on the effects of parameter variation on floor response spectra; DCA Part 2, Tier 2, Sections 3.7.1.2.2 and 3.7.2.1.1.3, on reduced stiffness for cracked concrete; and DCA Part 2, Tier 2, Section 3.7.2.5, on development of the ISRS. The applicant reduced the bending and shear stiffness by 50 percent for cracked concrete walls and diaphragms in accordance with the criteria in ASCE 43-05. For each seismic Category I structure, the design-basis ISRS are developed by appropriately enveloping the results from different combinations of analysis parameters, including ground motion spectra, soil profiles, damping values, and stiffness variation (cracked and uncracked), and by broadening the enveloped ISRS by ±15 percent on a linear frequency scale in accordance with RG 1.122. The staff's review found the applicant's method of taking into account the effects of parameter variations on floor response spectra to be acceptable because the method is consistent with the relevant guidance in DSRS Section 3.7.2, RG 1.122, and a recognized industry standard (ASCE 43-05).

3.7.2.4.9.2 Effect of Operating with Less than the Full Complement of 12 NuScale Power Modules

DCA Part 2, Tier 2, Section 3.7.2.9.1, provides information about the effects of operation with less than 12 NPMs. To investigate the effect on the design of operations with less than the full complement of 12 NPMs, the applicant performed a sensitivity study that involved 7 NPMs and reported the results obtained from the study. The applicant concluded that the difference in results between operation with 12 NPMs and operation with fewer NPMs in place is small and within the capacity of the building design. The applicant stated that, to design for the multiple configurations of the NPMs, the NPM bays are uniformly designed based on the maximum forces and moments experienced in the highest loaded west wall in a fully loaded (12) NPM configuration. In addition, a seven-module configuration is used as a sensitivity case to replicate how NPMs will be brought into the RXB and to model potential torsional behavior in an asymmetrical configuration.

To address the site-specific operational configurations, outside the scope of the presented 12 NPM and 7 NPM cases, the DCA also established COL Item 3.7-10, which states that a COL applicant that references the NuScale Power Plant DC will perform a site-specific configuration analysis that includes the RXB with the applicable configuration layout of the desired NPMs. The COL applicant will confirm that the following quantities are bounded by the corresponding certified design seismic demands or the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands:

- the ISRS of the standard design at the foundation and roof; see DCA Part 2, Tier 2, Figures 3.7.2-107 and 3.7.2-108 for foundation ISRS and Figure 3.7.2-113 for roof ISRS

- the maximum forces in the NPM lug restraints and skirts

- the site-specific ISRS for the NPM at the skirt support shown to be bounded by the ISRS in DCA Part 2, Tier 2, Figures 3.7.2-156 and 3.7.2-157; the site-specific ISRS for the

NPM at the lug restraints shown to be bounded by the ISRS in Figures 3.7.2-158 through 3.7.2-163

- the maximum forces and moments in the east and west wing walls and pool walls (see DCA Part 2, Tier 2, Tables 3B-22b and 3B-23b)

- the following site-specific ISRS will be shown to be bounded by their corresponding certified ISRS:

  - RXB north exterior wall at elevation 22.86 m (75′-0″): bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-110

  - RXB west exterior wall at elevation 48.40 m (126′-0″): bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-112

  - RXB crane wheels at elevation 44.35 m (145′-6″): bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-114

  - CRB east wall at elevation 22.32 m (76′-6″): bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-119a and Figure 3.7.2-119b

  - CRB south wall at elevation 36.58 m (120′-0″): bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-121a and Figure 3.7.2-121b

Based on its review of the results of the sensitivity studies performed by the applicant, the staff found the applicant’s approach to allowing operations with less than the full complement of 12 NPMs acceptable because (1) the difference in results between the operation with 12 NPMs and operation with fewer NPMs in place is small and remains within the capacity of the building and (2) the applicant provided a COL information item ensuring that the site-specific operating configurations are acceptable based on a site-specific analysis confirming that the site-specific demands are bounded by the corresponding certified design demands, or the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.

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, DCA Part 2, Tier 2, Section 3.7.2.10, indicates that the design of the NuScale seismic Category I and II 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. The DSRS also states that to account for accidental torsion, an additional eccentricity of ±5 percent of the maximum building dimension should be assumed for both horizontal directions.

In DCA Part 2, Tier 2, Section 3.7.2.11, the applicant stated that the element demand forces and moments obtained from SASSI2010 due to horizontal components of the CSDRS and CSDRS-HF inputs have been increased by 5 percent to account for accidental torsion. The staff's review found that the applicant's alternate methodology conservatively meets the intent of the acceptance criteria of DSRS 3.7.2.II.11 and thus is acceptable.

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, DCA Part 2, Tier 2, Section 3.7.2.12, indicates that the response spectrum method is not used in the evaluation of the site-independent NuScale seismic Category I and II 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 DSRS Section 3.7.2.II.13. The guidance in DSRS Section 3.7.2.II.13 states that either the composite modal damping approach or the modal synthesis technique can be used to account for element-associated damping.

In DCA Part 2, Tier 2, Section 3.7.2.15, the applicant stated that the damping values in RG 1.61 are used for 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 indicated that the implementation of these damping values in the dynamic analyses of the NuScale RXB and CRB does not directly follow the guidance in DSRS Section 3.7.2.II.13. Instead, damping procedures that are more suitable for the type of analysis performed are followed. For transient analysis with ANSYS, Rayleigh material damping is used. For SSI analysis with SASSI2010, hysteretic material damping is used. The applicant also indicated that both Rayleigh and hysteretic damping provide responses equivalent to the composite modal damping approach.

The staff reviewed the applicant's approach to implementing damping in dynamic analysis of the seismic Category I structures, including the use of Rayleigh damping in ANSYS and hysteretic damping in SASSI analysis, and found it acceptable because of the following:

- The Rayleigh (or proportional) damping approach uses a linear combination of the mass and stiffness matrices to form the damping matrix. The staff notes that the composite modal damping approach described in DSRS Section 3.7.2.II.13 uses either the mass or stiffness matrix as a weighting function in generating the composite modal damping and, thus, is considered a special case of the more general Rayleigh damping approach used by the applicant for transient dynamic analysis with ANSYS. The staff further notes that the guidance in Section 3.5 of ASCE 4-16 affirms this conclusion.

- The equation of motion in SASSI is solved in the frequency domain with the complex stiffness, incorporating damping in the form of hysteretic damping. The staff identified in the literature that the use of hysteretic damping in complex stiffness results in the same response as the viscous damping under harmonic loading, which can be generalized to any loading type that may be decomposed into a series of harmonic loadings. In SASSI, the transient input motion is decomposed into a series of harmonic motions through the Fourier Transform.

3.7.2.4.14 Determination of Dynamic Stability of Seismic Category I Structures

DSRS Section 3.7.2.II.14 provides guidance on the determination of design seismic overturning moments and sliding forces, structure-to-soil pressures beneath the foundation and alongside walls, and soil frictional forces for seismic Category I structures. In DCA Part 2, Tier 2, Section 3.7.2.14, the applicant indicated that DCA Part 2, Tier 2, Section 3.8.5, provides relevant information on these items. SER Section 3.8.5 evaluates the dynamic stability of seismic Category I structures.

3.7.2.4.15 DCA Part 2, Tier 1, Information

The staff reviewed the DCA Part 2, Tier 1 information related to DCA Part 2, Tier 2, Section 3.7.2, and found it to be acceptable because the design descriptions of the NuScale standard plant SSCs and applicable site parameters in DCA Part 2, Tier 1, are consistent with the information presented in DCA Part 2, Tier 2, Section 3.7.2, which is reviewed and accepted in this section of the SER.

3.7.2.5 Combined License Information Items

Table 3.7.2-1 lists COL information item numbers and descriptions related to seismic system analysis from DCA Part 2, Tier 2, Table 1.8-2.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.7-4	A COL applicant that references the NuScale Power Plant design certification will confirm that nearby structures exposed to a site-specific safe shutdown earthquake will not collapse and adversely affect the RXB or Seismic Category I portion of the CRB.	3.7.2
COL Item 3.7-5	A COL applicant that references the NuScale Power Plant design certification will perform a soil-structure interaction analysis of the RXB and the CRB using the NuScale SASSI2010 models for those structures. The COL applicant will confirm that the site-specific seismic demands of the standard design for critical SSCs in Appendix 3B are bounded by the corresponding design certified seismic demands and, if not, the standard design for critical SSCs 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
COL Item 3.7-6	A COL applicant that references the NuScale Power Plant design certification will perform a structure-soil-structure interaction analysis that includes the RXB, CRB, Radioactive Waste Building and both Turbine Generator Buildings. The COL applicant will confirm that the site-specific seismic demands of the standard design SSCs are bounded by the corresponding design certified seismic demands and, if not, the standard	3.7.2

Item No.	Description	DCA Part 2, Tier 2, Section
	design SSCs will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	
COL Item 3.7-10	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific configuration analysis that includes the RXB with applicable configuration layout of the desired NuScale Power Modules. The COL applicant will confirm the following are bounded by the corresponding design certified seismic demands: 1) The in-structure response spectra of the standard design at the foundation and roof. See DCA Part 2, Figure 3.7.2-107 and Figure 3.7.2-108 for foundation in-structure response spectra and Figure 3.7.2-113 for roof in-structure response spectra. 2) The maximum forces in the NuScale Power Module lug restraints and skirts. 3) The site-specific in-structure response spectra for the NuScale Power Module at the skirt support will be shown to be bounded by the in-structure response spectra in Figure 3.7.2-156 and Figure 3.7.2-157. The site-specific in-structure response spectra for the NuScale Power Module at the lug restraints will be shown to be bounded by the in-structure response spectra in Figure 3.7.2-158 through Figure 3.7.2-163. 4) The maximum forces and moments in the east and west wing walls and pool walls. See DCA Part 2, Table 3.7.2-32. 5) The site-specific in-structure response spectra shown immediately below will be shown to be bounded by their corresponding certified in-structure response spectra: • RXB north exterior wall at elevation 75′ 0″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-110. • RXB west exterior wall at elevation 126′ 0″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-112. • RXB crane wheels at elevation 145′ 6″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-114. • CRB east wall at elevation 76′ 6″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-119a and Figure 3.7.2-119b. • CRB south wall at elevation 120′ 0″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-121a and Figure 3.7.2-121b. If not, the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	3.7.2

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.7-11	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific analysis that assesses the effects of soil separation. The COL applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra shown in DCA Part 2, Figure 3.7.2-107 through Figure 3.7.2-122.	3.7.2
COL Item 3.7-12	A COL applicant that references the NuScale Power Plant design certification will perform an analysis that uses site-specific soil and time histories to confirm the adequacy of the fluid-structure interaction correction factor.	3.7.2
COL Item 3.7-13	A COL applicant that references the NuScale Power Plant design certification 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 3.7-14	A COL applicant that references the NuScale Power Plant design certification will demonstrate that the site-specific seismic demand is bounded by the FSAR capacity for an empty dry dock condition.	3.7.2
COL Item 3.7-15	A COL applicant that references the NuScale Power Plant design certification will determine the appropriate site-specific number of interaction planes for soil structure interaction.	3.7.2

The staff reviewed the COL information items listed in Table 3.7.2-1 pertaining to seismic system analysis discussed in DCA Part 2, Tier 2, Section 3.7.2, and found them to be acceptable based on the staff's technical evaluation presented in SER Section 3.7.2.4.

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

DCA Part 2, Tier 2, Section 3.7.3, "Seismic Subsystem Analysis," covers seismic analysis of seismic Category I subsystems that are not included in the main structural systems, such as miscellaneous concrete and steel structures; buried piping, tunnels, and conduits; and atmospheric tanks. For distribution systems (e.g., cable trays, conduit, heating, ventilation, air conditioning, piping) and equipment, including their supports, the staff reviews supplementary seismic analysis criteria in accordance with DSRS Section 3.7.3 but reviews the actual distribution systems and their supports in accordance with SRP Section 3.9.2, Revision 3 "Dynamic Testing and Analysis of Systems, Structures, and Components," issued March 2007, and SRP Section 3.9.3, Revision 3, "ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures," issued April 2014. The staff also reviews intervening structural elements between these distribution systems and equipment supports and the building structural steel/concrete under this DSRS section.

The main areas of review include the following:

- 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 nonseismic 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
- seismic Category I buried piping, conduits, and tunnels

3.7.3.2 Summary of Application

DCA Part 2, Tier 1: The certified design includes four seismic subsystems specifically evaluated in Tier 2: (1) reactor building crane (RBC), (2) NPM, (3) fuel storage racks, and (4) bioshield.

DCA Part 2, Tier 2: DCA Revision 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.7.2. This section describes the miscellaneous concrete and steel structures, buried piping, conduits, tunnels, dams, and aboveground tanks subsystems.

As applicable, DCA Part 2, Tier 2, Section 3.7.3, references DCA Part 2, Tier 2, Section 3.7.2, and, to a limited extent, Section 3.7.1, for the seismic analysis methods for the subsystems, such as response spectrum analysis, time history analysis, procedure used for analytical modeling, analysis procedures for damping, and modal and spatial response combination methods.

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

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

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

3.7.3.3 Regulatory Basis

DSRS Section 3.7.3 describes the relevant requirements of the Commission regulations for seismic subsystem analysis and the associated acceptance criteria. The specific requirements include the following:

- GDC 2 requires that the design basis shall 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 historical data have been accumulated.

- 10 CFR Part 50, Appendix S, requires that, for 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 take into account SSI effects and the expected duration of the vibratory motion. If the OBE is set at one-third or less of the SSE, an explicit analysis or design is not required. 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. In 10 CFR Part 50, Appendix S, the NRC also requires that 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.

The guidance in DSRS Section 3.7.3 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 1.61, Revision 1, to determine the acceptability of damping values used in the dynamic seismic analyses of seismic Category I subsystems

- RG 1.92, Revision 3, to determine the acceptability of modal and spatial combination methods used in the dynamic seismic analyses of seismic Category I subsystems

- RG 1.122, Revision 1, to determine development of floor design response spectra for seismic design of floor-supported equipment or components

3.7.3.4 Technical Evaluation

Following the guidance in DSRS Section 3.7.3, Revision 0, the staff reviewed DCA Part 2, Tier 2, Section 3.7.3. The staff also reviewed other DCA sections partially or in whole when they were referenced in DCA Part 2, Tier 2, Section 3.7.3. If the staff identified no significant issues in those DCA sections to affect the staff’s safety findings for DCA Part 2, Tier 2, Section 3.7.3, the evaluation of DCA Part 2, Tier 2, Section 3.7.3, below simply refers to the SER sections that evaluate those sections.

3.7.3.4.1 Seismic Analysis Methods

DCA Part 2, Tier 2, Section 3.7.3.1, “Seismic Analysis Methods,” indicates that the NuScale seismic subsystems may be analyzed using the response spectrum analysis method or equivalent static method. DCA Part 2, Tier 2, Section 3.7.2, describes the methods evaluated in SER Section 3.7.2. SER Sections 3.9 and 3.12 address seismic analysis of piping and equipment.

DCA Part 2, Tier 2, Section 3.7.3.1.2, “Equivalent Static Load Method,” indicates that the equivalent static method is available to use for the analysis of simple SSCs if dynamic analysis is not performed. The seismic static load is the product of the equipment or component (SSC) mass times the constant static factor of 1.5 times the peak spectral acceleration of the applicable required response spectra (a smaller factor can be used if adequately justified).

Because the peak spectral acceleration is used regardless of the natural frequency of the SSC, along with the conservative factor of 1.5, the method is consistent with the acceptance criteria of DSRS Section 3.7.2.II.1.B.iii. As such, the staff finds the method to be acceptable because the method is conservative and consistent with the DSRS acceptance criteria.

3.7.3.4.2 Determination of the Number of Earthquake Cycles

DCA Part 2, Tier 2, 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 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Plants," dated June 8, 2005. The staff finds that the DCA specification of these two methods is consistent with DSRS Section 3.7.3, "Seismic Subsystem Analysis," Acceptance Criterion 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 gives the staff's evaluation of the piping and components related to the number of earthquake cycles.

3.7.3.4.3 Procedure Used for Analytical Modeling

DCA Part 2, Tier 2, Section 3.7.3.3, indicates that the criteria and bases described in DCA Part 2, Tier 2, Section 3.7.2.3, "Procedures Used for Analytical Modeling," are 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. In addition, the damping coefficients are consistent with DCA Part 2, Tier 2, Section 3.7.1, Table 3.7.1-6. SER Sections 3.7.1 and 3.7.2 separately address these areas of review. Also, DCA Part 2, Tier 2, Section 3.7.3.3, indicates that the RXB structural weight is greater than 2,200 meganewtons (MN) (500,000 kilo-pounds-force (kips)). A subsystem can be decoupled if the weight is less than 22,000 kilonewtons (kN) (5,000 kips). The larger subsystems, the NPM and RBC, weigh approximately 9,000 kN (2,000 kips) and thus could be decoupled. However, the RXB model included a simplified model of the RBC and NPM. Moreover, distribution systems, such as cable trays, piping, heating, ventilation, air conditioning, and individual components will not have significant weight. Hence, it satisfies the DSRS acceptance criteria for the decoupling. The DCA specifically lists and evaluates four subsystems: (1) RBC, (2) NPM, (3) fuel racks, and (4) bioshield. The staff evaluates these subsystems as follows:

(1) RBC

DCA Part 2, Tier 2, Section 9.1.5, discusses the RBC, and it is evaluated in SER Chapter 9.1.5. Based on the evaluation provided in SER Section 9.1.5, the staff concluded the RBC design is consistent with the guidance in DSRS Section 3.7.3.II.8 and is therefore acceptable

(2) NPM

Each NPM is a subsystem. DCA Part 2, Tier 2, Appendix 3A, describes the seismic analysis of the NPMs. The RXB system model incorporates a simplified representation of the NPMs. The RXB model is then analyzed for SSI to establish the seismic demand. Results from the RXB SSI analysis include in-structure 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. SER Section 3.7.2.4 includes the staff evaluation of the RXB SSI analysis. The staff evaluates the detailed dynamic analysis of the NPM subsystem in SER Section 3.9.2.2.

(3) fuel storage racks

The applicant has deferred the evaluation of the fuel storage racks to the COL applicant. This is now identified as COL Action Item 9.1-8 in DCA Part 2, Tier 2, Section 9.1.2.

(4) bioshields

In DCA Part 2, Tier 2, Section 3.7.3.3.1, the applicant addressed the design of the bioshields. The bioshields are classified as nonsafety-related, not risk-significant, seismic Category II components that are placed on top of each module bay at elevation 38.1 m (125 ft). The bioshields provide radiation protection in the RXB and support personnel access. The bioshields are attached to the bay walls. A vertical assembly is attached to the horizontal slab of the bioshield. The bioshields are detached from the bay walls during refueling activities. During that time, the removed bioshield is placed on top of an in-place bioshield. The staff's evaluation of the bioshield is discussed below.

3.7.3.4.3.1 Bioshield

The applicant stated that the horizontal assembly of the bioshield consists of a 60-cm (23.5-in.)-thick, 34 MPa (5,000 psi) reinforced concrete horizontal slab sandwiched between two 1-cm (0.25-in.) stainless steel plates. The vertical assembly is constructed of a stainless steel tube framing system. The space within the framing holds radiation shielding panels. The vertical assembly is vented for heat removal during normal operations, as well as heat and pressure removal during an HELB. Furthermore, the applicant used two of the ISRS from those developed at multiple locations in the RXB for the seismic design of the bioshield.

The staff reviewed the applicant's RAI responses and the pertinent documents related to the structural design of the bioshield during an audit at the applicant's facility.

In DCA Part 2, Tier 2, Section 3.7.3.3.1, the applicant stated that the bolts used to anchor the bioshield to the operating bay and pool walls are designed in accordance with American Concrete Institute (ACI) Code (ACI 349-06) "Code Requirements for Nuclear Safety-Related Concrete Structures." The vertical assembly and horizontal component of the bioshield are designed in accordance with Chapter J of AISC 360 "Specification for Structural Steel Buildings" for welded connections. All loads and their load combinations are checked per AISC and ACI codes for both members and connections.

Since the rigging for the bioshield is not designed or selected, the applicant proposed COL Item 3.7-16 for the COL applicant to confirm that the bioshield lifting components and connections can withstand the bioshield loads with appropriate load factors. Therefore, the staff accepted the COL item.

Based on the staff review, the staff finds that the applicant’s basis for the bioshield design is acceptable because it is consistent with the guidance in DSRS Section 3.7.3.II.8.

3.7.3.4.4 Basis for Selection of Frequencies

DCA Part 2, Tier 2, Section 3.7.3.4, “Basis for Selection of Frequencies,” describes the basis for the selection of frequencies. The applicant indicated that when practical, components are designed so that the fundamental frequencies of the component are either less than one-half or more than twice the dominant frequencies of the support structure. The applicant also indicated that the equipment will be tested or analyzed to demonstrate that it is adequate, considering 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 guidance in DSRS Section 3.7.3.II.4.

SER Section 3.12 evaluates components and equipment.

3.7.3.4.5 Analysis Procedure for Damping

DCA Part 2, Tier 2, Section 3.7.3.5, “Analysis Procedures for Damping,” indicates that the analysis procedure used to account for the damping in subsystems conforms to DCA Part 2, Tier 2, Section 3.7.1.2, “ Percentage oof Critical Dampling Values” and Section 3.7.2.15, “Analysis Procedure for Damping.” The staff finds this approach acceptable because it is consistent with the acceptance criteria in DSRS Section 3.7.3.II.5. The staff evaluates DCA Part 2, Tier 2, Sections 3.7.1.2 and 3.7.2.15, in SER Sections 3.7.1.4 and 3.7.2.4, respectively. The staff evaluates component modal damping of the piping system in SER Section 3.12.

3.7.3.4.6 Three Components of Design Ground Motion

In DCA Part 2, Tier 2, 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 earthquake motions are combined in the same manner as the seismic response resulting from the analysis of building structures, as specified in DCA Part 2, Tier 2, 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 evaluates DCA Part 2, Tier 2, Section 3.7.2.6, in SER Section 3.7.2.

3.7.3.4.7 Combination of Modal Response

In DCA Part 2, Tier 2, Section 3.7.3.7, “Combination of Modal Responses,” the applicant indicated that in response to the spectrum analysis of subsystems, SRSS is used to combine the modal responses when the modal frequencies are well separated; otherwise, the modal responses are combined in accordance with RG 1.92, Revision 3. The applicant also stated that the modes are combined for the structural frequencies when they are not well separated, also in accordance with RG 1.92, Revision 3. The staff finds that the approach is acceptable because it is consistent with DSRS Acceptance Criterion 3.7.3.II.7 and follows the guidance in RG 1.92, Revision 3.

3.7.3.4.8 Interaction of Nonseismic Category I Subsystems with Seismic Category I SSCs

In DCA Part 2, Tier 2, Section 3.7.3.8, the applicant stated that when nonseismic Category I subsystems (or portions thereof) could adversely affect seismic Category I SSCs, the subsystems are categorized as seismic Category II and analyzed following DCA Part 2, Tier 2, Section 3.7.3.1. The staff finds this approach acceptable because it is consistent with the guidance in DSRS Acceptance Criterion 3.7.3.II.8.

The applicant also stated that for nonseismic Category I subsystems attached to seismic Category I SSCs, the modeling of the seismic Category I SSCs includes the dynamic effects of the nonseismic Category I subsystems. The attached nonseismic Category I subsystems, up to the first anchor beyond the interface, are designed in such a manner that the CSDRS does not cause any failure in the seismic Category I SSCs. As defined in DCA Part 2, Tier 2, Section 3.7.1, for the standard NuScale 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 guidance in DSRS Acceptance Criterion 3.7.3.II.8.

SER Section 3.12 evaluates piping and equipment.

3.7.3.4.9 Multiply Supported Equipment and Components with Distinct Inputs

The applicant stated that both the uniform support motion (USM) method and the independent support motion (ISM) method may be used to address multiply supported equipment and components. For the ISM method, the applicant used the guidance in NUREG-1061, Volume 4. The staff finds the applicant's method for treating the multiply supported equipment and components acceptable because it is consistent with the guidance in DSRS Acceptance Criterion 3.7.3.II.9.

SER Sections 3.9.2 and 3.12 evaluate piping and equipment.

3.7.3.4.10 Use of Equivalent Vertical Static Factors

In DCA Part 2, Tier 2, Section 3.7.3.10, the applicant stated that the equivalent vertical static factors are not used in the design of the seismic Category I and seismic Category II structures. The applicant further stated that the vertical seismic loads are generated from the SSI analysis (SASSI2010). The staff finds the applicant's method for vertical seismic loads acceptable because it satisfies DSRS Acceptance Criteria 3.7.2.II.10 and 3.7.3.II.10.

3.7.3.4.11 Torsional Effect of Eccentric Masses

In DCA Part 2, Tier 2, Section 3.7.3.11, 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 above description to be acceptable because it is consistent with SRP Acceptance Criterion 3.7.3.II.11.

3.7.3.4.12 Seismic Category I Buried Piping, Conduits, and Tunnels

In DCA Part 2, Tier 2, Section 3.7.3.12, the applicant stated that the design does not include buried seismic Category I piping or conduits. The tunnel between the CRB and the RXB is analyzed as part of the CRB. SER Sections 3.7.2 and 3.8.4 evaluate the CRB design.

3.7.3.4.13 Methods for Seismic Analysis of Seismic Category I Concrete Dams

In DCA Part 2, Tier 2, Section 3.7.3.13, the applicant stated that the design does not include or require the presence of a dam. Therefore, no evaluation is required.

3.7.3.4.14 Methods for Seismic Analysis of Aboveground Tanks

The NuScale design does not include seismic Category I aboveground tanks. Therefore, no evaluation is required.

3.7.3.5 Combined License Information Items

Table 3.7.3-1 lists COL information item numbers and descriptions related to seismic system analysis from DCA Part 2, Tier 2, Table 1.8-2.

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

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.7-16	A COL applicant that references the NuScale Power Plant design certification will determine the means and method of lifting the bioshield. A COL applicant will demonstrate that bioshield components and connection can withstand the bioshield loads and appropriate load factors.	3.7.3.3.1

3.7.3.6 Conclusion

The staff finds that the applicant has adequately addressed seismic subsystem analysis in accordance with the acceptance criteria delineated in DSRS Section 3.7.3. On this basis, the staff concludes that the regulatory criteria requirements 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. Appendix S to 10 CFR Part 50 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.

3.7.4.2 Summary of Application

DCA Part 2, Tier 1: There is no Tier 1 information associated with this area of review.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.7.4, "Seismic Instrumentation," provides a description of the NuScale design seismic instrumentation. DCA Part 2, Tier 2, Section 3.7.4.2, states that seismic sensors will be located in the free field, RXB, and CRB 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. In DCA Part 2, Tier 2, Section 3.7.4.1, the applicant stated that the NuScale design requires a deviation from the guidance in RG 1.12, Revision 3, "Nuclear Power Plant Instrumentation for Earthquakes," issued November 2017, in that seismic instrumentation cannot be installed inside containment because the containments are flooded as part of the refueling process.

DCA Part 2, Tier 2, Section 3.7.4.2, discusses the location of seismic instrumentation and states that the exact sensor location and sensor type are site specific, and the COL applicant will discuss it as part of the response to COL Item 3.7-1. The applicant also provided criteria to ensure that the site, RXB, and CRB are adequately instrumented for a seismic event.

DCA Part 2, Tier 2, Section 3.7.4.3, states that the seismic monitoring system (SMS) provides seismic Category I annunciation in the main control room. Separately, the SMS provides information to the main control room via the plant control system. The COL applicant will base the alarm levels upon the site-specific OBE as part of the response to COL Item 3.7-1.

In DCA Part 2, Tier 2, Section 3.7.4.4, the applicant provided a comparison with RG 1.166 and stated that the COL applicant will discuss site-specific conformance with RG 1.166, Revision 0, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," issued March 1997, as part of the response to COL Item 3.7-1.

In DCA Part 2, Tier 2, Section 3.7.4.5, the applicant stated that the SMS is expected to be operable during all modes of plant operation, including periods of plant shutdown.

In DCA Part 2, Tier 2, Section 3.7.4.6, the applicant specified that the COL applicant will discuss SMS program implementation as part of the response to COL Item 3.7-8.

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, requires seismic instrumentation and the provision of suitable instrumentation 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 2, issued March 1997, to determine description and location of seismic instrumentation at the plant

- RG 1.166, Revision 0, issued March 1997, to provide requirements for pre-earthquake planning and post-earthquake plant operator actions

3.7.4.4 Technical Evaluation

The staff reviewed the updated DCA Part 2 and evaluated the completeness and adequacy of technical requirements to the placement and operability of the SMS. The applicant stated that exact sensor locations are site specific, and the COL applicant will discuss the sensor locations as part of the response to COL Item 3.7-1.

In DCA Part 2, Tier 2, Table 1.9-2, the applicant noted that they partially conform to RG 1.12, Revision 2, issued March 1997. The basis for the nonconformances was that the selection of specific equipment is the responsibility of the COL applicant and that seismic instrumentation cannot be installed inside the containment. In addition, the applicant noted that the exact sensor locations are site specific and would be discussed by the COL applicant. The staff agreed that the locations of the free-field seismic sensors are site specific and the COL applicant should discuss them. However, the staff requested that NuScale prescribe the locations of the structural seismic instrumentation in DCA Part 2.

In a letter dated September 29, 2017 (ADAMS Accession No. ML17272A607), and a letter dated February 12, 2018 (ADAMS Accession No. ML18043B166), the applicant specified locations of the instructural sensors within the RXB and CRB; the applicant also prescribed installation of an additional seismic sensor in the free-field downhole at the elevation corresponding to the GMRS or FIRS and an additional sensor on the basemat. DCA Part 2, Tier 2, Section 3.7.4.2, includes criteria for placement of eight seismic instrument sensor units to ensure that the site, the RXB, and the CRB are adequately instrumented. The staff reviewed this information and determined that the criteria provide acceptable guidance for a COL applicant to determine the exact sensor locations.

In the letter dated February 12, 2018 (ADAMS Accession No. ML18043B166), the applicant committed to using RG 1.12, Revision 3, issued October 2017, for the seismic monitoring instrument locations.

In the selection of the exact sensor locations, when addressing COL Item 3.7-7, the COL applicant should adhere to the following criteria to ensure that the site, RXB, and CRB are adequately instrumented for a seismic event:

- Two sensor units are located in the free field. One sensor is located at the free ground surface consistent with the site conditions and properties used to determine the site-specific GMRS. The second is a downhole instrument located at the foundation level as close as directly over the first sensor as practical.

- Two sensor units are located in the RXB on the basemat at elevation 7.32 m (24 ft). One sensor is located near the intersection of gridlines RX-1 and RX-A. The other sensor is located near the intersection of gridlines RX-7 and RX-A.

- A fifth sensor unit is located in the RXB at elevation 22.86 m (75 ft) on the east face of gridline RX-6, between RX-B and RX-C.

- A sixth sensor unit is located on the RXB roof near the intersection of gridlines RX-4 and RX-C.

- A seventh sensor unit is located in the CRB on the basemat at elevation 15.24 m (50 ft), near the intersection of gridlines CB-4 and CB-A.

- An eighth sensor unit is located in the CRB at elevation 30.48 m (100 ft) on the east face of gridline CB-1 between CB-B and CB-C.

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 DCA Part 2, Tier 2, Table 1.8-2.

Table 3.7.4-1: NuScale COL Information Items for Section 3.7.4

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.7-1	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific SSE.	3.7.1.1.3
COL Item 3.7-7	A COL applicant that references the NuScale Power Plant design certification will provide a seismic monitoring system and a seismic monitoring program that satisfies RG 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” Revision 3 (or later) and RG 1.166 “Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions,” Revision 0 (or later).	3.7.4
COL Item 3.7-8	A COL applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the seismic monitoring program. In addition, a COL applicant that references the NuScale Power Plant design certification will prepare site-specific procedures for activities following an earthquake. These procedures and the data from the seismic instrumentation system will provide sufficient information to determine if the level of earthquake ground motion requiring shutdown has been exceeded. An activity of the procedures will be to address measurement of the post-seismic event gaps between the fuel racks and the pool walls and between the individual fuel racks and to take appropriate corrective action if needed (such as repositioning the racks or assuring that the as-found condition of the racks is acceptable based on the assumptions of the racks' design basis analysis). Acceptable guidance for procedure development is contained in Regulatory Guide 1.166 "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions," Rev. 0 (or later) and 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event," Rev. 0 (or later).I	3.7.4.6

3.7.4.6 Conclusion

Based on the review of DCA Part 2, Tier 2, Section 3.7.4, the staff concludes that the applicant provided complete and adequate technical requirements for the placement and operability of an SMS suitable to record seismic response of nuclear power plant features important to safety after an earthquake, consistent with 10 CFR Part 50, Appendix S, and the guidance in RG 1.12. The staff, therefore, finds the seismic instrumentation proposed by the applicant acceptable.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This section does not apply to the NuScale Power Plant because the NuScale design uses a steel containment.

3.8.2 Steel Containment

3.8.2.1 Introduction

DCA Part 2, Tier 2, Section 3.8.2, "Steel Containment," states that the containment vessel (CNV) is an integral portion of the NPM, with the following primary functions:

- Provide an essentially leaktight barrier to contain fission product releases for the RCPB during design-basis events (DBEs).
- Contain the mass and energy release from a postulated LOCA and secondary-system pipe ruptures.
- Support operation of the ECCS by containment of the reactor coolant and heat transfer through the CNV wall.
- Contain and support the RPV, RCS, and associated SSCs.

In DCA Part 2, Tier 2, Section 3.8.2, the applicant provided the following information on the steel containment:

- physical description
- applicable design codes, standards, and specifications
- loading criteria, including loads and load combinations
- design and analysis procedures
- structural acceptance criteria
- materials, quality control programs, and special construction techniques
- testing and ISI programs

3.8.2.2 Summary of Application

DCA Part 2, Tier 1: In DCA Part 2, Tier 1, Section 2.1.1, "Design Description," and Table 2.1-2, "NuScale Power Module Mechanical Equipment," and Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," provide the Tier 1 information for the CNV.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.8.2, provides a system description of the steel containment.

ITAAC: DCA Part 2, Tier 1, Table 2.1-4, includes the ITAAC for DCA Part 2, Tier 2, Section 3.8.2. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: The applicant gave the TS associated with DCA Part 2, Tier 2, Section 3.8.2, in DCA Part 2, Tier 2, Section 6.2, "Containment Systems."

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

- TR-0716-50424-P, Revision 1, “Combustible Gas Control,” issued March 2019
- TR-0916-51502-P, Revision 2, “NuScale Power Module Seismic Analysis,” issued April 2019
- TR-0917-56119, Revision 1, “CNV Ultimate Pressure Integrity,” issued June 2019 (ADAMS Accession No. ML1915A382)

3.8.2.3 Regulatory Basis

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

- 10 CFR 50.55a(b)(2)(ix) and GDC 1, as they relate to steel containment being designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed
- GDC 2, as it relates to the ability of the SSCs important to safety to withstand the most severe natural phenomena, such as winds, tornadoes, floods, and earthquakes, and the appropriate combination of all loads
- GDC 4, as it relates to the SSCs important to safety being 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
- GDC 16, “Containment Design,” as it relates to the ability of the reactor containment to act as an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment
- GDC 50, “Containment Design Basis,” as it relates to the reactor containment structure being designed with sufficient margin of safety to accommodate the calculated pressure and temperature conditions resulting from any LOCA
- GDC 53, “Provisions for Containment Testing and Inspection,” as it relates to the reactor containment being designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations, which have resilient seals and expansion bellows
- 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

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 DCA Part 2, Tier 2, Section 3.8.2, against the agency's 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. This section evaluates DCA Part 2, Tier 2, Section 3.8.2, with regard to each of these seven acceptance criteria.

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; 10 CFR 50.44; and 10 CFR 52.47(b)(1). The staff used the guidance in DSRS Section 3.8.2 to review DCA Part 2, Tier 2, Section 3.8.2. 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 staff reviewed DCA Part 2, Tier 2, Section 3.8.2, to establish that the applicant provided sufficient information to define the primary structural aspects and elements relied upon to perform the containment function, particularly the structural and functional characteristics.

The staff also reviewed the DCA in comparison to the guidelines in DSRS Section 3.8.2.I.1.A for geometry of the CNV, including sketches showing plan views at various elevations, sections in at least two orthogonal directions, and dimensions.

In DCA Part 2, Tier 2, Figures 3.8.2-1, 3.8.2-4, 3.8.2-5, 3.8.2-7, 6.2-1, 6.2-2a, and 6.2-3a include appropriate dimensions. Figures 3.8.2-8 and 3.8.2-9 show views of the CNV–RPV boundary at the supports.

Figure 3.8.2-10 in DCA Part 2, Tier 2, Section 3.8.2, identifies the CNV pressure boundary and RCPB for the ECCS trip/reset actuator valve. The figure shows that the CNV penetration and safe-end contain two small hydraulic tubing lines (1.27-cm (1/2-in.) diameter or less) inside of the penetration. These hydraulic lines connect to the valve body at the end of the safe-end. The tubing wall forms the RCPB, and the penetration and safe-end form the CNV pressure boundary. The trip/reset valve body forms both the RCS and CNV pressure boundary.

The staff reviewed DCA Part 2, Tier 2, Figures 3.8.2-1, 3.8.2-4, 3.8.2-5, 3.8.2-7, 6.2-1, 6.2-2a, and 6.2-3a and Figures 3.8.2-8, 3.8.2-9, and 3.8.2-10. In DCA Part 2, Tier 2, Sections 5.4.2.4 and 6.1.1.2 state that no socket welds are used on lines greater than or equal to DN 20 (NPS 3/4) and that socket welds less than DN 20 (NPS 3/4) conform to 10 CFR 50.55a(b)(1)(ii).

The staff finds that DCA Part 2, Tier 2, Figures 3.8.2-1, 3.8.2-4, 3.8.2-5, 3.8.2-7, 6.2-1, 6.2-2a, and 6.2-3a, and Figures 3.8.2-8, 3.8.2-9, and 3.8.2-10 provide the level of detail specified in DSRS Section 3.8.2.I.1.A, and Sections 5.4.2.4 and 6.1.1.2 are in accordance with 10 CFR 50.55a(b)(1)(ii).

The electrical penetration assemblies are bolted to the CNV on the top head and are subject to periodic leak testing, as described in SER Section 6.2.6. The staff reviewed the electrical penetration assemblies and finds them acceptable because they are designed, constructed, tested, qualified, and installed in accordance with IEEE Standard 317-1983, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," as endorsed by RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants."

The electrical penetration assemblies are Class 900 and, in accordance with ASME B16.5, "Pipe Flanges and Flanged Fittings: NPS 1/2 through NPS 24 Metric/Inch Standard," are rated to be acceptable for the CNV design pressure of 7.24 MPa (1,050 psi).

SER Section 8.3.1 evaluates the electrical design of the electrical penetration assemblies. The electrical penetration assemblies are designed to ASME BPV Code Class 1, as stated in DCA Part 2, Tier 1, Table 2.1-2. The staff finds this acceptable for meeting the mechanical requirements of IEEE Standard 317-1983.

3.8.2.4.2 Applicable Design Codes, Standards, and Specifications

The staff reviewed the codes, standards, and specifications in DCA Part 2, Tier 2, Sections 3.8.2.2 and 3.8.2.1.1, against the list in DSRS Section 3.8.2.II.2.

The applicant stated that the CNV is an ASME BPV Code Class MC component that is designed, constructed, and stamped as an ASME BPV Code Class 1 vessel in accordance with ASME BPV Code, Section III, Subsection NB, except that overpressure protection is in accordance with ASME BPV Code, Section III, Article NE-7000, in lieu of ASME BPV Code, Section III, Article NB-7000. The staff finds this consistent with DSRS Section 3.8.2.II.2. The staff finds that this satisfies the applicable requirements of 10 CFR 50.55a and the criteria of GDC 1.

The staff also reviewed DCA Part 2, Tier 2, Section 3.8.2.2.1, "Codes, Standards, and Specifications," and found that it was consistent with DSRS Section 3.8.2.II.2.

3.8.2.4.3 Loading Criteria, Including Loads and Load Combinations

DCA Part 2, Tier 2, Section 3.8.2.3, "Load Combinations," states that stresses and fatigue for the CNV pressure-retaining components were evaluated in accordance with ASME BPV Code, Section III, Subsection NB. DCA Part 2, Tier 2, Appendix 3A, describes seismic loading of the CNV, which references TR-0916-51502-P.

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 earthquake. TR-0916-51502-P, Revision 2, Section 8.4.3, describes the calculated displacement and acceleration time histories, maximum relative displacements, in-structure response spectrum, and maximum forces and moments at representative component interfaces.

The staff reviewed the structure 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 found that the analysis was performed in accordance with the guidance in SRP Section 3.9.2.

During an audit of the stress and fatigue analyses as documented in the audit summary (ADAMS Accession No. ML19340A971), the staff reviewed the updated primary stress analysis of the CNV, which incorporated the seismic loads and the design pressure of 7.24 MPa (absolute) (1,050 psia). The staff also reviewed documents to ensure that the CNV was analyzed with the proper design pressure and seismic loads and that the results conformed to the requirements of the ASME BPV Code. These included the ASME BPV Code calculations for the CNV FW nozzle and CNV MS nozzle and the ASME design specification for the CNV.

The staff noted in the primary stress analysis of the CNV that the controlling stress results in the vessel for design conditions was the general membrane stress, Pm, in the upper CNV shell. As the acceptance ratio was close to the ASME limit, the staff requested the applicant to detail any conservatisms in the stress calculation. The applicant explained that the reactor module seismic load specification, which provided the updated seismic loads to qualify the membrane stress of the CNV shell, was defined at eight elevations of the CNV and the maximum force and moment components from all elevations were used for the calculation of membrane stress. The staff reviewed the conservatisms detailed by the applicant and finds this method acceptable for meeting ASME BPV Code, Section III, requirements for primary stress.

The staff noted in the primary stress analysis of the CNV nozzles that the Level C allowable stress ratios for CNV nozzles 5–7, 11–14, 22, and 23 were close to the ASME limit, and the staff requested the applicant to clarify what conservatisms were used in the stress calculation. During the audit, the applicant stated that the development of the nozzle loads contained the conservatism. The applicant explained that the nozzles are grouped by nominal dimensions of the connected piping, and the maximum force and moment components from all grouped nozzles are used in the stress calculation. The applicant also provided the staff with an example showing the actual loads and the comparison to the bounding loads used. The staff reviewed the conservatisms detailed by the applicant and finds them acceptable for meeting ASME BPV Code, Section III, requirements for primary stress.

DCA Part 2, Tier 2, Section 3.8.2.3, states that the load combinations meet the requirements of ASME BPV Code, Section III, NCA-2141(b), and consider the guidance in RG 1.57. The staff reviewed the load combinations given in DCA Part 2, Tier 2, Tables 3.8.2-2 and 3.8.2-3, and find that the load combinations are consistent with DSRS Section 3.8.2.II.3. The staff finds that this satisfies the criteria in GDC 2, 4, and 16.

The staff reviewed TR-0716-50424-P, Section 3.3.4.4. The staff found that the CNV shell is within ASME Service Level C limits for loads from a reflected detonation event and within ASME Service Level D limits for loads from deflagration to detonation transition. For the severe accident deflagration to detonation transition event, the CNV membrane hoop strain is maintained below the 1.5-percent strain limit. The staff also reviewed TR-0716-50424-P, Section 3.3.4.5, and found that the major CNV flange bolting, main CNV closure, CRDM access, shell manway, SG inspection, and PZR access were within allowable ASME Level C and Level D limits for the combustion event and are, therefore, acceptable. SER Section 6.2 evaluates the deflagration to detonation transition and combustion events. The staff finds that this satisfies the applicable requirements of 10 CFR 50.44.

3.8.2.4.4 Design and Analysis Procedures

DCA Part 2, Tier 2, Section 3.8.2.4.5, discusses the nonlinear (plastic) 3-D FEA performed to determine the ultimate pressure capacity of the CNV. With respect to the FEMs, the applicant stated that analyses conform to guidance in NUREG/CR-6906, “Containment Integrity Research at Sandia National Laboratories—An Overview,” Appendix A, “Containment Capacity Analysis Guidelines,” issued July 2006, and the failure criteria that determine the ultimate pressure capacity of the CNV are based on guidance in RG 1.216.

DSRS Acceptance Criterion 3.8.2.4.F states that design and analysis procedures are acceptable if performed in accordance with the guidance in RG 1.57, applicable guidance in RG 1.216, or both.

RG 1.216, Staff Regulatory Position C.1.k, states that the details of the analysis and results should be submitted in report form with the following:

- calculated static pressure capacity
- dynamic pressure capacity, if applicable (static pressure capacity reduced to account for dynamic amplification effects)
- associated failure modes
- criteria governing the original design and criteria used to establish failure
- analysis details and general results, which include (1) modeling details, (2) description of computer code(s), (3) material properties and material modeling, (4) loading and loading sequences, (5) failure modes, and (6) interpretation of results, with all assumptions made in the analysis and test data (if relied upon) clearly stated and technically justified
- appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure

TR-0917-56119, “CNV Ultimate Pressure Integrity,” issued June 2019, addresses the details of the predicted containment internal pressure capacity above design pressure.

In the stress and fatigue audit summary (ADAMS Accession No. ML19340A971), the staff documented the review of the primary stress calculation for the CNV RPV support. The staff reviewed the FEA performed on the CNV RPV support and found the results to be within Level D limits and to meet the requirements of the ASME BPV Code, Section III, Subsection NF.

From June 1, 2017–August 29, 2017, the staff performed the Phase 1 design specification audit (ADAMS Accession No. ML18018A234), during which it reviewed the ASME CNV design specification; CNV ultimate pressure integrity analysis; CNV primary stress analysis; seismic design criteria; and the drawings of the CNV assembly, top head assembly, upper and lower section, and the CNV-RPV closure bolts. During the review of the CNV primary stress analysis, the staff found that the fatigue evaluation of the CNV was not available, nor was a timetable available for completion. The fatigue evaluation for the CNV was also not available for review during the Phase 2 design specification audit (ADAMS Accession No. ML19018A140).

The applicant performed a fatigue evaluation to determine the most limiting locations. For the CNV, the refueling flange, the MS nozzle, the FW nozzle, and the CVCS nozzles were identified as the most severe.

The staff reviewed the MS and FW nozzle ASME BPV Code calculation documents for the CNV. The staff noted that these were Revision A documents, and the applicant explained that QA procedure QP-0303-10267, Revision 13, "Design Control Process," defines letter and number revisions. This procedure was reviewed during an inspection by the NRC and found to be acceptable (ADAMS Accession No. ML19093A669). According to the definitions, a letter revision is a preliminary document, and a number revision is an approved/verified document. A letter revision document will ultimately be revised.

However, following the update to the primary stress analysis for the CNV, which incorporated the updated design pressure and updated seismic loads, the applicant informed the staff that the MS and FW nozzles ASME BPV Code calculation documents for the CNV would not be updated at this time. The applicant stated that these Revision A documents are preliminary and that these calculation documents would be revised to include the latest design information as prescribed in the design specification for the CNV to complete the ASME design report for these components. The staff acknowledges that these calculation documents are preliminary and that the ITAAC closure process will provide the opportunity for the staff to verify the as-built design report. The staff notes that DCA Part 2, Tier 1, Table 2.1-4, Item 1, specifically includes ITAAC for ASME Code Class 1 components including the CNV.

The staff reviewed the CNV refueling flange stress analysis document, which also contained a fatigue evaluation. The most limiting primary stresses for the refueling flange and for the bolts were within ASME BPV Code requirements. The fatigue usage was found to be less than 1.0 and meets ASME BPV Code requirements.

The staff reviewed the CNV CVC nozzle/weld fatigue analysis, which detailed the fatigue analysis of welds connecting safe-ends of four CNV CVCS nozzles to their respective valve bodies and nozzle forgings. These were performed for CNV6—RCS injection, CNV7—PZR spray, CNV13—RCS discharge, and CNV14—RPV high point degasification. The bounding nozzle was CNV6—RCS injection at the nozzle-to-top safe-end weld wet surface and lower transition zone of the nozzle. The fatigue usage was found to be less than 0.4 for the welds and nozzle necks and less than 1.0 for the nozzle transitions through base metal and meets ASME BPV Code requirements.

In summary, the staff found the calculations in various stages of completion (some calculations were final with assumptions that needed to be verified, and some were not final calculations) but determined that the calculations were in accordance with the ASME BPV Code, Section III, and 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.

3.8.2.4.5 Structural Acceptance Criteria

DCA Part 2, Tier 2, Section 3.8.2.5, describes the CNV structural integrity acceptance criteria limits, which are developed in accordance with ASME BPV Code, Section III, Subarticles NB-3200 and NF-3200, for plate-type and shell-type supports for the CNV support. In DCA Part 2, Tier 2, Tables 3.8.2-2 and 3.8.2-3 show the ASME BPV Code limits for the defined load combination. The CNV is also fabricated, installed, and tested according to ASME BPV Code, Section III, Subsections NB and NF.

The staff reviewed DCA Part 2, Tier 2, Tables 3.8.2-2 and 3.8.2-3, and finds them acceptable because the structural acceptance criteria comply with those identified in DSRS Acceptance Criterion 3.8.2.II.5, as the total stresses and loads are defined in accordance with ASME BPV Code, Section III.

3.8.2.4.6 Materials, Quality Control Programs, and Special Construction Techniques

DCA Part 2, Tier 2, Section 3.8.2.6, describes the CNV materials, which conform to the requirements of ASME BPV Code, Subarticle NB-2000. The CNV fabrication conforms to the requirements of ASME BPV Code, Subarticles NB-4000 and NF-4000. The CNV uses no special construction techniques. The quality control program involving materials, welding procedures, and nondestructive examination of welds conforms to ASME BPV Code, Subarticles NB-2000, NB-4000, and NB-5000. In DCA Part 2, Tier 2, Tables 6.1-1 and 6.1-2 show the materials of construction.

The staff reviewed DCA Part 2, Tier 2, Section 3.8.2.6, and found that it is in accordance with the guidance in DSRS Section 3.8.2.I.6 for materials, quality control, and special construction techniques.

3.8.2.4.7 Testing and Inservice Inspection Programs

DCA Part 2, Tier 2, Section 3.8.2.7, describes testing and ISI requirements for the CNV. For those CNV pressure boundary items defined as ASME BPV Code, Section III, Class 1, preservice examinations are in accordance with ASME BPV Code, Section III, Subarticle NB-5280, and ASME BPV Code, Section XI, Subarticle IWB-2200, using the examination methods of ASME BPV Code, Section V, except as modified by Subarticle NB-5111. These preservice examinations include 100 percent of the pressure boundary welds. The CNV pressure boundary welds are required to have a volumetric or surface examination performed in accordance with ASME BPV Code, Section XI, Subarticle IWB-2000, as stated in DCA Part 2, Tier 2, Section 6.2.1.6. DCA Part 2, Tier 2, Table 6.2-3, “Containment Vessel Inspection Elements,” lists the CNV inspection requirements. The staff reviewed DCA Part 2, Tier 2, Section 3.8.2.7, and finds it in accordance with the guidance in DSRS Section 3.8.2.II.7. SER Section 6.2.1.6 gives the staff evaluation of the ISI of the CNV. The staff finds that this satisfies the criteria of 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 Combined 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 accordance with the acceptance criteria set forth in DSRS Section 3.8.2, and 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.

3.8.4.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Sections 3.11, 3.12, and 3.13, present the Tier 1 information for this section. This includes the design descriptions and ITAAC for the RXB, the CRB, and RWB (seismic Category II structure), respectively.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.8.4 and Appendix 3B, "Design Reports and Critical Section Details," provide Tier 2 information on the design of seismic Category I structures, other than containment. The design of the RWB is not reviewed by the staff except for its influence, by its close proximity, on seismic Category I structures.

The applicant described structures; applicable codes, standards, and specifications; loads and load combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and ISI requirements. The applicant also described COL information items related to structural design aspects of seismic Category I structures.

The seismic Category I structures (other than containment) in the NuScale certified design application are the RXB and the CRB. These buildings are site independent and designed for the CSDRS and the CSDRS-HF, as described in DCA Part 2, Tier 2, Section 3.7.1. The predominant feature of the RXB is the UHS pool, which consists of the spent fuel pool, refueling area pool, and the reactor pool. The reactor pool contains bays to house up to 12 NPMs. The CRB sits on a separate foundation, approximately 10.4 m (34 ft) east of the RXB. A below-grade tunnel extends out from the CRB to the RXB. There is a 15-cm (6-in.) expansion gap between the end of the tunnel and the RXB walls. The RWB sits on a separate foundation, approximately 7.6 m (25 ft) west of the RXB (i.e., at the end opposite from the CRB). The applicant used a triple SSI model to include the impact of the adjacent buildings (CRB, RXB, and RWB) on the seismic demand of the seismic Category I structures. The applicant employed the SAP2000 computer code and the SASSI2010 computer code to obtain the static and seismic demands, respectively, for use in designing the RXB and CRB. In addition, the applicant performed separate analyses in ANSYS to determine added fluid loads and evaluate the effects of thermal loads on the RXB structure.

DCA Part 2, Tier 2, Appendix 3B, provides a design report for 15 critical sections in the RXB and 7 in the CRB. In accordance with DCA Part 2, Tier 2, Appendix 3B, the applicant selected these 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: In DCA Part 2, Tier 1, Tables 3.11-2, 3.12-2, and 3.13-1, for the RXB, RWB, and CRB, respectively, give the ITAAC for DCA Part 2, Tier 2, Section 3.8.4. 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 TR for this area of review is NuScale TR-0818-61384 P, Revision 2, "Pipe Rupture Hazards Analysis," issued July 2019.

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 DCA Part 2, Tier 2, Section 3.8.4, in accordance with DSRS Section 3.8.4. DSRS Section 3.8.4 describes acceptance criteria to meet the relevant requirements of the NRC’s regulations pertaining to the structural design of seismic Category I structures other than containment. Consistent with DSRS Section 3.8.4, the staff reviewed (1) the description of the structures, (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.

3.8.4.4.1 Description of the Structures

The seismic Category I structures (other than containment) in the DCA are the RXB and the CRB. In DCA Part 2, Tier 2, Sections 3.8.4.1.1 and 3.8.4.1.2 provide design descriptions of the RXB and CRB, respectively, and related RAI responses.

The RXB is a reinforced concrete shear wall building embedded approximately 26.2 m (86 ft) below grade level, supported on a single basemat foundation. It extends approximately 24.7 m (81 ft) above grade level for a total overall height of approximately 50.9 m (167 ft) from the top of the roof to the bottom of the basemat. The main seismic resisting system, in both N-S and E-W directions, is composed of exterior shear walls along the perimeter and interior shear walls for the reactor pool and crane walls. All main shear walls are 1.5 m (5 ft) thick and are tied together by rigid concrete diaphragms at different levels and by the roof. The exterior shear walls around the perimeter are continuous along the entire building height, from the basemat at elevation 7.3 m (24 ft), to their connection with the roof at elevation 49.7 m (163 ft). The exterior shear walls are braced by reinforced concrete pilasters embedded within the walls. There are five pilasters on the north and south walls, three pilasters on the east and west walls, and four corner pilasters. The wall pilasters are 1.5 m (5 ft) wide and extend 1.5 m (5 ft) out from the wall. The corner pilasters are 3.81 m (12.5 ft) wide and extend 0.76 m (2.5 ft) out from the wall. The RXB is 105.5 m (346 ft) wide (excluding pilasters) in the E-W direction and 45.87 m

(150.5 ft) wide (excluding pilasters) in the N-S direction and is centered on a below-grade 109.1 m by 49.53 m (358 ft by 162.5 ft) basemat. The RXB houses and provides protection to the NPMs and systems and components required for plant operation and shutdown. The predominant feature of the RXB is the UHS pool, which consists of the spent fuel pool, refueling area pool, and the reactor pool. The reactor pool contains bays to house up to 12 NPMs.

The CRB is a reinforced concrete building with an upper steel structure. It is located on a separate foundation from the RXB, approximately 10.4 m (34 ft) to the east and embedded approximately 16.8 m (55 ft) below grade. It extends approximately 12.5 m (41 ft) above grade level, for a total height of approximately 29.3 m (96 ft) from the top of the steel roof to the bottom of the basemat foundation. There are two pilasters along both the east and west walls and a single pilaster on the north and south walls. These pilasters are 0.9 m (3 ft) wide and extend 0.9 m (3 ft) out from the wall. In addition, there are four corner pilasters, 2.29 m (7.5 ft) wide and extending 0.46 m (1.5 ft) out from the wall. The CRB is 24.7 m (81 ft) wide (excluding pilasters) in the E-W direction and 36.47 m (119 ft, 8 in.) wide (excluding pilasters) in the N-S direction. It is centered on a below-grade 27.7 m by 39.52 m (91 ft by 129 ft, 8 in.) basemat. The CRB includes a below-grade tunnel that extends from the CRB to the RXB. There is a 15-cm (6-in.) expansion gap between the end of the tunnel and the corresponding connecting walls on the RXB. The CRB's primary function is to house the main control room and the technical support center. The SSCs on the upper steel structure have no safety-related functions and are categorized as seismic Category II.

The staff reviewed the descriptions of structures in DCA Part 2, Tier 2, Section 3.8.4, Appendix 3B, and Section 1.2.2, 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 to be 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 addressed in the DCA, 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 dynamic models for the seismic analyses of the structures and establish the relationship between adjacent structures. Additionally, the staff found the structural descriptions contained a sufficient level of detail to confirm the consistency of structural design aspects (e.g., structural member capacities and reinforcement configuration) in the design descriptions with the reference design codes. Moreover, the staff found the level of detail in the structural descriptions to be consistent with the level of detail of structural descriptions for past LWR applications. Based on the above, the staff concludes that the descriptions of structures in DCA Part 2, Tier 2, are acceptable.

The applicant also considered concrete pour pressure of 28.7 kPa (600 pounds per square foot (psf)) as the construction load on the RXB pool liner plate and its support structure in the load combinations and added COL Item 3.8-6 to account for the construction load of concrete pour pressure in Revision 3 of DCA Part 2, Tier 2, Section 3.8.4.1.7, "Reactor Building Pools." The staff also discusses the construction load of concrete pour pressure issue in Section 3.8.5.4.5.8, "Leak Detection," of this SER.

3.8.4.4.2 Applicable Codes, Standards, and Specifications

DCA Part 2, Tier 2, Section 3.8.4.2, lists the codes, standards, and specifications applicable to seismic Category I for 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.

The staff compared the codes, standards, and specifications listed by the applicant with the acceptance criteria in DSRS Section 3.8.4.II.2 and found that the listing addresses the codes, standards, and specifications acceptable to the staff in accordance with DSRS Section 3.8.4.II.2, with some exceptions. These exceptions involve guidance documents that are either not applicable to the NuScale DCA or are addressed under other DCA sections. Specifically, DSRS Section 3.8.4.II.2 addresses RG 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments”; RG 1.127, “Inspection of Water-Control Structures Associated with Nuclear Power Plants”; and RG 1.143, which DCA Part 2, Tier 2, Section 3.8.4.2, does not list. This is acceptable to the staff because RG 1.136 is specific to concrete containments and therefore not applicable to the NuScale design, which uses a steel CNV. Further, as stated in DCA Part 2, Tier 2, Table 1.9-2, RG 1.127 is not applicable to the standard design structures, as water control structures, if applicable, are designed for specific site conditions and are outside the scope of the NuScale design. Also, DCA Part 2, Tier 2, Section 3.8.4.2, is specific to seismic Category I, and RG 1.143 is applicable to the seismic Category II RWB, as stated in DCA Part 2, Tier 2, Section 3.2.1.4, and does not need to be listed. Further, in DCA Part 2, Tier 2, Table 1.9-2 and Section 3.2.1.4 both address conformance with RG 1.143. The staff’s review of the RWB is addressed in SER Section 3.8.4.4

Additionally, DCA Part 2, Tier 2, Sections 3.8.4.2.1, 3.8.4.3, and 3.8.4.5, indicate the use of AISC N690-12, for the design of seismic Category I structures. This is an updated version of that listed in DSRS Section 3.8.4.II.2, AISC N690-1994, including Supplement 2, dated October 6, 2004. The applicant used AISC N690-12 in the design of the NPM steel supports (i.e., the lug restraints and the skirt support). In describing its use of the 2012 version of AISC N690, relative to the 1994 version, the applicant compared the governing load combination for the NPM supports and showed the load combinations in each respective version of the code to be equivalent. Further, the applicant compared the allowable shear stress, a critical failure mode for these supports, provided by the two versions of the code. The use of the 2012 version yields higher estimates of allowable shear strength relative to the respective estimates based on the 1994 version. The staff compared the differences in the capacity estimates with the respective D/C ratio based on the applicant’s use of AISC N690-12. The staff’s review determined that the differences in the capacity estimates are reasonably covered by the conservatism in the NPM support design as demonstrated by the resulting D/C ratios. Based on the aforementioned equivalency between load combinations and sufficient conservatism in the NPM support design to reasonably cover the differences in the aforementioned capacity estimates, the staff finds the applicant’s use of AISC N690-12 to be acceptable.

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 DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, Section 3.8.4.3, presents the loads and load combinations for the RXB and CRB structural design and analysis and references ACI 349-06, RG 1.142, and AISC N690-12 as the basis for the loads and load combinations. Specifically, in DCA Part 2, Tier 2, Sections 3.8.4.3.1 through 3.8.4.3.22 discuss the loads, and Tables 3.8.4-1 and 3.8.4-2 discuss the load combinations for concrete and steel structures, respectively.

The staff reviewed and compared the loads and load combinations presented in the DCA with the referenced codes. The staff's review found that, in general, the load definitions and load combinations conform with the reference codes. The discussion below addresses the cases for which the staff needed additional information or clarifications.

DCA Part 2, Tier 2, Section 3.8.4.3.3, considered the earth's pressure (H) in the design of the RXB and CRB. The embedded exterior walls of these buildings are subjected to lateral soil pressure loads induced by (1) static soil pressures, including soil weight, hydrostatic pressure, and a surcharge load at grade level and (2) dynamic soil pressures considering standalone building structure SSI analysis cases and building SSSI analysis cases. The staff's review of the static earth pressures found that the applicant adequately determined them to be a function of unit weight of soil consistent with the engineering fill properties in DCA Part 2, Section 3.7.1; unit weight water; and at-rest pressure coefficient consistent with the soil angle of internal friction in DCA Part 2, Tier 1, Table 5.0-1, and therefore acceptable.

As stated in DCA Part 2, Tier 2, Section 3.8.4.3.3, the explicit modeling of the backfill soil in the SSI and SSSI analysis models inherently accounts for these dynamic soil pressures. The respective seismic demands for the embedded walls are conservatively established based on the envelope of the results from SSI and SSSI analysis cases that consider multiple time histories, multiple soil profiles, cracked and uncracked concrete stiffness properties, a standalone building model, and a combined triple building model. Because the seismic demands used in the design of the embedded walls are conservatively obtained as the envelope of multiple SSI and SSSI analysis cases that adequately considered a range of seismic input and soil and structure properties and used detailed 3-D FEMs of the structures with consideration of embedment, the staff finds the applicant's consideration of soil dynamic pressures to be reasonable and acceptable.

In DCA Part 2, Tier 2, Sections 3.8.4.3.8 and 3.8.4.3.9 describe the applicant's consideration of operating thermal loads (T_o), accident thermal loads (T_a), and accident pressure load, respectively, considered in the design evaluations of the RXB. The applicant's design evaluation for these loads is described in DCA Part 2, Tier 2, Sections 3.8.4.4.1, 3B.1.3, and 3B.2.8. The applicant calculated the strains in concrete, reinforcing steel bars, and liner plates for mechanical loads based on the resulting stresses from SAP2000, SASSI2010, and ANSYS analyses. Additionally, the applicant performed a series of thermal and structural analyses in ANSYS to determine the strains throughout the RXB for the operating temperature load, accident temperature load, and accident pressure load. In addition, the applicant combined the above-mentioned set of strains consistent with the load combinations involving these types of loads and applicable acceptance criteria and compared them with the strain limits set forth in ACI 349, ACI 349.1R-07 "Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures," and ASME BPV Code Section III, Division 2 (for the RXB pool liner). The staff reviewed the resulting strains for the load combinations involving operating temperature load, accident temperature load, and accident pressure loads, and confirmed that such strains are lower than the limits set forth in the design codes. Based on the conformance to the code strain limits, the staff concludes that the RXB has been shown to possess adequate capacity to withstand the demands from the design load combinations, including thermal loads and accident pressure loads, and finds the applicant's consideration of the effects of these loads in the RXB design to be acceptable.

In DCA Part 2, Tier 2, Sections 3.8.4.3.19, 3.8.4.3.20, and 3.8.4.3.21 describe the applicant's consideration of jet impingement load (Y_j), pipe break reaction loads (Y_r), and missile impact loads (Y_m), respectively. The applicant determined the Y_j and Y_r loads based on its PRHA described in TR-0818-61384-P. SER Section 3.6 evaluates the applicant's PRHA. To assess the effects of these loads on the RXB structure, the applicant performed a punching shear evaluation for the walls under jet impingement and jet reaction forces; that is, the RXB pool wall and exterior wall as per the pipe break locations addressed in TR-0818-61384-P. The evaluation results showed a D/C ratio of 0.02. The staff checked this D/C ratio against the D/C ratios and maximum strains based on the controlling load combinations for the RXB design and found the effects of the aforementioned Y_j and Y_r loads to be bounded by the RXB capacity. With respect to Y_m load effects, in accordance with TR-0818-61384-P, the applicant evaluated the pipe whip effects of MSS piping for several pipe lengths and angle configurations. The staff reviewed the applicant's evaluation and confirmed that, for the bounding pipe length and angle configuration, no scabbing occurs on walls impacted by the whipping pipe, thereby precluding the ejection of the wall material and subsequent impact of NPM and related systems. In addition, in accordance with COL Item 3.6-2 and COL Item 3.6-3, 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 Y_j, Y_r, and Y_m load effects in the RXB design to be acceptable.

3.8.4.4.4 Design and Analysis Procedures

DCA Part 2, Tier 2, Section 3.8.4.4, provides an overview of the design and analysis procedures for the RXB and CRB and refers to DCA Part 2, Tier 2, Section 3.8.4.5, for the design acceptance criteria. DCA Part 2, Tier 2, Section 3.8.4.5, indicates that the design criteria for reinforced concrete and steel structures are in accordance with ACI 349-06, with supplemental guidance by RG 1.142 and AISC N690-12, respectively. Further, DCA Part 2, Tier 2, Section 3.8.4, Appendix 3B, provides the design assessment for the critical sections within the RXB and CRB. Specifically, Appendix 3B addresses 15 critical sections in the RXB and 7 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 considered to be representative of the structural design.

3.8.4.4.4.1 Analysis Procedures

The applicant performed static analyses with SAP2000 and seismic analyses with SASSI2010 and ANSYS to determine the structural response to nonseismic loads and seismic loads, including consideration of fluid structure interaction effects, respectively. 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 in the public domain and have a sufficient history of use to justify their applicability. SER Section 3.7.2 provides specific details with respect to the staff's review of the V&V of particular modeling and analysis aspects performed with SASSI2010.

The applicant performed the aforementioned static and seismic analyses using detailed 3-D FEMs representing the primary structural members, including walls, beams, columns, pilasters, floors, and roofs. Additionally, these models included finite element representations of the NPMs. Both the static and dynamic analyses cases used to perform the structural response evaluations considered uncracked and cracked concrete conditions. In DCA Part 2, Tier 2, Section 3.7.1.2.2 and Table 3.7.1-7 specify the level of concrete cracking considered in the analyses. The staff reviewed the applicant's treatment of concrete cracking in its analyses and found it to be consistent with the criteria in DSRS Sections 3.8.4.II.4B and 3.7.2.II.3.C.iv. Specifically, as stated in DCA Part 2, Tier 2, Section 3.7.1.2.2 and Table 3.7.1-7, the applicant implemented concrete stiffness reduction factors in accordance with ASCE 43-05, which the staff finds acceptable, as established in DSRS Sections 3.8.4.II.4B and 3.7.2.II.3.C.iv. Additionally, in establishing both the static and seismic demands for use in the structural design, the applicant enveloped the results obtained from the analysis sets considering uncracked and cracked concrete conditions, which the staff finds to be conservative. Based on the consistency with the acceptance criteria in the DSRSs and the applicant's conservative approach to enveloping the aforementioned analysis results, the staff finds the applicant's consideration of concrete cracking acceptable.

As stated in DSRS Section 3.8.4.II.4.L, the design and analysis procedures for the RXB pool and the RXB are acceptable if they consider the multiple NPMs and their interaction in the reactor pool water. As described in DCA Part 2, Tier 2, Sections 3.8.4.3.1.3 and 3.7.2.1.3.2, the 12 NPMs are included in the RXB FEMs used for static and SSI analyses, respectively. The NPM FEMs used for these analyses are composed of mass and beam elements. These NPM mass and beam models are developed to have similar dynamic characteristics as a detailed 3-D NPM model. These models include the pool water by assigning lumped masses on the pool walls and foundation nodes that are in contact with the pool water. Additionally, as stated in DCA Part 2, Tier 2, Section 3.7.2.1.3.4, to fully account for hydrodynamic effects from all three directional components of earthquake input motions (i.e., in addition to the effects captured in the SSI analyses), the applicant performed detailed dynamic fluid structure interaction analyses with explicit fluid element representation of the pool water and detailed 3-D shell element NPM models. The staff's review finds that the applicant's consideration of the NPMs in the analysis models and their interaction in the reactor pool water meet the acceptance criteria in DSRS Section 3.8.4.II.4.L and are therefore acceptable.

3.8.4.4.4.2 Design Procedures

The staff's review of the design of the seismic Category I structures focused on the adequacy of the lateral force resisting system to limit story drift to acceptable levels and on verifying that the critical sections identified in DCA Part 2, Tier 2, Section 3.8.4, Appendix 3B, possess adequate capacity to withstand the design-basis demands. In DCA Part 2, Tier 2, Tables 3.7.2-28 and 3.7.2-29 present the displacement results for the RXB and CRB, respectively. The staff reviewed the displacement results in these tables, determined drift values from ground level to the roof elevation, and compared the calculated drift values with the allowable drift limits in Table 5-2 of ASCE 43-05. The staff's review confirmed that the calculated drift values for both the RXB and CRB are less than the allowable limits. On this basis, the staff concludes that the lateral force-resisting systems for the RXB and CRB are sufficiently stiff to control the drift of the building within the limits specified by ASCE 43-05 and are, therefore, acceptable.

DCA Part 2, Tier 2, Section 3.8.4, Appendix 3B, presents the applicant's design reports and critical section details for the RXB and CRB. In general, this appendix provides additional details with respect to the applicant's design procedures, a summary of the demand and capacity evaluations, and reinforcement arrangements for the reinforced concrete critical sections. Consistent with the critical section criteria described above (see the first paragraph under SER Section 3.8.4.4.4), the applicant addressed 15 critical sections for the RXB and 7 for the CRB, which include walls, slabs, pilasters, floor beams, buttresses, NPM bay structural members, and RXB and CRB basemats. SER Section 3.8.5 gives the staff's evaluation of the RXB and CRB basemats. The seismic demands for the critical sections were obtained from the SSI analyses described in DCA Part 2, Tier 2, Section 3.7.2, which considered uncracked and cracked concrete conditions, a standalone RXB building model, a triple building model (i.e., RWB, RXB, and CRB), four different soil profiles, two design response spectra (CSDRS and CSDRS-HF), and six sets of response spectra compatible time histories. The nonseismic demands were obtained from static analyses, which considered uncracked and cracked concrete conditions and single and triple building models. Further, as stated in COL Items 3.8-2 and 3.8-4, the COL applicant will perform site-specific verifications to confirm the standard design acceptability for use at a designated site. Specifically, the COL applicant will confirm that the standard design seismic demands bound the respective site-specific demands based on unique soil conditions and SSE at the designated site and consideration of the effects of construction aid elements that may be left in place after construction. The staff finds the aforementioned determination of 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.

In DCA Part 2, Tier 2, Sections 3B.1.1.3 and 3B.1.2.2 and Tables 3B-66 to 3B-94 present the code provisions implemented by the applicant in its design process for the reinforced concrete critical sections. The staff's review confirmed that the applicant implemented the capacity equations from ACI 349 in the design of the reinforced concrete critical sections. Further, the staff's review of the applicant's summary of demand and capacity evaluations verified that all critical sections have sufficient capacity to withstand the design-basis demands. Additionally, during the seismic/structural audit in December 2018 (ADAMS Accession No. ML19098A162), the staff confirmed the consistency of the calculation with the information described in DCA Part 2 for the critical sections and examined a sample of structural members other than the critical sections. Specifically, the staff examined the walls at gridlines A.7, B, D, and D.3 and confirmed that the design checks were consistent with those performed for the critical sections and possessed greater capacity than the design-basis demands (i.e., D/C ratios less than 1 per design code). Further, the staff verified the applicant's in-plane shear check for the RXB walls and confirmed that in all cases the in-plane shear capacity was greater than the in-plane shear demand.

3.8.4.4.4.2.1 Reactor Building Roof

The RXB roof is a 1.2-m (4-ft)-thick slab composed of a flat section at elevation 55.2 m (181 ft) between gridlines A.7 and D.3 and two inclined roof sections, sloping down from elevation 55.2 m (181 ft) to elevation 49.7 m (163 ft) from gridline A.7 to gridline A and from gridline D.3 to gridline E at the north and south sides of the building, respectively. There are stiffener walls underneath the sloping sections of the roof located at gridlines 2, 3, 4, 5, and 6 between gridlines A to A.7 and D.3 to E. DCA Part 2, Tier 2, Table 3B-18, addresses the design evaluation results for longitudinal reinforcement, concrete compressive strength requirements, and out-of-plane requirements for both concrete and transverse reinforcement. Additionally, the applicant's design evaluation addressed the verification for in-plane shear capacity. In performing the aforementioned design evaluations, the applicant implemented the capacity equations from ACI 349-06. The staff reviewed the design evaluation results and confirmed that the roof possesses greater capacity than the design basis demands. Further, the staff verified the locations in the roof with the bounding D/C ratios and confirmed such locations to be in general agreement with the expected locations of maximum positive and negative moments for the roof slab. Based on the aforementioned conservative determination of design-basis demands, the use of the equations from ACI 349 and respective demonstration of greater capacity than the design basis demands, and consistency with the staff's expectations with respect to the locations of maximum positive and negative moments, the staff concludes that the RXB roof is designed to retain its structural integrity when subjected to the design-basis demands and is, therefore, acceptable.

3.8.4.4.4.2.2 NuScale Power Module Lug Restraints

DCA Part 2, Tier 2, Appendix 3B, Section 3B.2.7.4, describes the applicant's design evaluation of the NPM lug restraints. The staff's review focused on confirming the adequacy of the design-basis demand and capacity for the NPM lug restraints. DCA Part 2, Tier 2, Table 3B-28, lists the maximum reaction forces for the NPM supports, obtained from the ANSYS NPM seismic analysis cases documented in TR-0916-51502-P, and the respective SASSI seismic analysis cases described in DCA Part 2, Tier 2, Section 3.7.2. In its design evaluation of the lug restraints, the applicant used a demand greater than the maximum reaction forces obtained from the aforementioned analyses. On this basis, the staff finds the demand used in the design of the NPM lug restraints to be conservative and acceptable. Additionally, DCA Part 2, Tier 2, Appendix 3B, Table 3B-57, presents the applicant's design evaluation results for the individual modes of failure for components of the lug restraints. The staff's review of the applicant's design evaluation results confirmed that the lug restraint possesses greater capacity than the design demands. Further, during the seismic/structural audit in December 2018, the staff examined the design report for the NPM lug restraint and confirmed that the design evaluation for the bearing of the shear lugs against the concrete (i.e., the controlling mode of failures as described in DCA Part 2, Tier 2, Table 3B-57) was performed in accordance with ACI 349. The staff also confirmed that the bearing stress demand, capacity, and respective D/C ratio in the applicant's design report were consistent with the respective information in DCA Part 2, Tier 2. Based on the use of a conservative demand 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 are, therefore, acceptable.

3.8.4.4.4.2.3 Wall at Gridline 1

The wall at gridline 1 is a 1.5-m (5-ft)-thick exterior structural wall on the west side of the building. It extends along the entire building height from the basemat to the roof. DCA Part 2, Tier 2, Table 3B-2, summarizes the design evaluation results for longitudinal reinforcement, concrete compressive strength requirements, and out-of-plane requirements for both concrete and transverse reinforcement. Additionally, the applicant's design evaluation addressed the verification for in-plane shear capacity. In performing these evaluations, the applicant implemented the capacity equations from ACI 349. The staff reviewed the aforementioned design evaluation results and confirmed that the wall at gridline 1 possesses greater capacity than the design basis demands. Additionally, the staff reviewed the design compliance with the provisions in ACI 349, Section 21.7.6, related to boundary elements of reinforced concrete structural walls. As stated in ACI 349, Section 21.7.6.1, boundary elements are not required for walls and piers with wall height over wall length ratios less than or equal to 2.0. The staff's review confirmed that the wall height over wall length ratio for the wall at gridline 1 is less than 2.0 and concluded, consistent with ACI 349, Section 21.7.6.1, that boundary elements are not required. In addition, to confirm the determination, based on wall dimensional properties, that wall boundary elements are not required in the wall design, during the seismic/structural audit in

December 2018, the staff requested the applicant to provide the compressive stresses at the bottom of the wall at gridline 1. The staff compared the compressive stresses provided by the applicant with alternative threshold stress criteria for inclusion of wall boundary elements and found them to be less than the threshold stress limit, thereby confirming that boundary elements do not have to be included in the wall design. Further, the staff reviewed the reinforcement arrangement presented in DCA Part 2, Tier 2, Figures 3B-8 and 3B-9. The staff's review confirmed that the reinforcement arrangement meets the applicable requirements in ACI 349, Sections 7.6 and 7.7, related to spacing limits for reinforcement and concrete protection for reinforcement, respectively. Based on this conservative determination of design-basis demands, the use of the ACI 349 equations and respective demonstration of greater capacity than the design basis demands, and compliance with the aforementioned ACI 349 provisions, the staff concludes that the wall at gridline 1 is designed to retain its structural integrity when subjected to the design-basis demands and is, therefore, acceptable.

As described above, DCA Part 2, Tier 2, Section 3.8.4.4, provides an overview of design and analysis aspects for the seismic Category I RXB and CRB, and DCA Part 2, Tier 2, Appendix 3B, summarizes the design assessment for critical sections in these buildings. Additionally, DCA Part 2, Tier 2, addresses the design of the RXB pool liner and key structural supports, including the rails for the RBC and fuel-handling machine (FHM), and the reactor flange tool (RFT) stand. The discussion below gives the staff's review of the RXB pool liner and aforementioned structural supports.

3.8.4.4.4.2.4 Reactor Building Pool Liner

DCA Part 2, Tier 2, Table 3.2-1, categorizes the RXB pool liner as a seismic Category I component. As stated in DCA Part 2, Tier 2, Section 3.8.4.1.7, the RXB pool liner is designed to ensure that the stresses and strains in the liner plate remain below the respective limits based on ASME BPV Code Section III, Division 2, and lateral deflection limit for containment liners given in Reference 3.8.4-10 in DCA Tier 2, Section 3.8.4. In DCA Tables 3B-60 and 3B-61 the applicant provided strain results for the liner. The staff reviewed the strain results in these tables and compared them with the limits in Table CC-3720-1 of ASME BPV Code Section III, Division 2. The staff's review confirmed that the calculated strain values are less than the allowable limits. Also, during the seismic/structural audit in December 2018, the staff reviewed the applicant's design report for the RXB pool liner and connection to the RXB pool walls, including stress, strain, and deflection results. The staff confirmed the acceptance criteria presented in the report to be consistent with the criteria described in the DCA. Additionally, per COL Item 3.8-6, a COL applicant will verify that the construction loads applied to the pool liner and its support structure will not exceed that considered in the DCA (for additional discussion, see SER Sections 3.8.5.4.5.6 and 3.8.5.4.5.8). Based on the above, the staff concludes that the liner plate and connection are designed to retain their integrity when subjected to the design-basis demands and are, therefore, acceptable.

3.8.4.4.4.2.5 Reactor Building Cooling and Fuel-Handling Machine Rails

In DCA Part 2, Tier 2, Sections 3.8.4.1.13 and 3.8.4.1.14 address the structural design criteria for the steel rails and anchor plates of the RBC and FHM, respectively. The steel rails and anchor plates meet the AISC N690-12 and ACI 349-06 design criteria. In addition, the loads and load cases, including extreme load cases, consider SSE loads. The extreme load cases are consistent with criteria in ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," issued 2004; that is, the code of reference for the RBC and FHM design. Based on the applicant's use of AISC N690, ACI 349, and ASME

NOG-1, the staff considers the applicant's approach for the design of the RBC and FHM rails acceptable because it is in accordance with the applicable code and standard prescribed in DSRS Section 3.8.4.II.4 and acceptable to the staff as described in SER Section 3.8.4.4.5.

3.8.4.4.4.2.6 Reactor Flange Tool Stand

In DCA Part 2, Tier 2, Sections 9.1.5.2.3 and 3.8.4.1.15 describe the use of an RFT for de-tensioning and tensioning the reactor flange closure bolts and as a structural support for the lower reactor pressure vessel (RPV) during refueling operations. As stated in DCA Part 2, Tier 2, Table 3.2-1, the RFT support is categorized as seismic Category I. The staff's review in this section of the SER focuses on evaluating the structural integrity of the RFT support when subjected to the design-basis demands. The seismic analyses performed for the RFT support are described in TR-0916-51502-P. SER Section 3.9.2 evaluates the seismic analysis in TR-0916-51502-P. As described in DCA Part 2, Tier 2, Section 3.8.4.1.15, and shown in Figure 3.8.4-34, the RFT support comprises an RFT stand (a cylindrical shape), an RFT base embed plate; an upper support structure consists of an upper support ring, which is braced (or supported) horizontally by four equally spaced steel wide flange beams that are pin supported by steel plates that are anchored to the concrete walls. As described in DCA Part 2, Tier 2, Section 3.8.4.5 and Tables 3.8.4-21 and 3.8.4-22, the floor and wall embed plates are designed in accordance with ACI 349 and AISC N690, and the RFT stand and upper support structures in accordance with Subsection NF of ASME Section III, Division 1. These tables also provided the D/C ratios resulting from the design evaluation of the RFT support. The staff reviewed the design evaluation results and confirmed that the RFT support possesses greater capacity than the design basis demands. In addition, in accordance with COL Item 3.8-5, the COL applicant will verify and demonstrate that the site-specific demands for the RFT support are bounded by the certified design or augment the design otherwise. Based on the applicant's design evaluation, the staff's review, and the site-specific verifications to be performed by the COL applicant, the staff finds that the RFT support is designed to retain its structural integrity when subjected to the design-basis demands and is therefore acceptable.

3.8.4.4.4.2.7 Radioactive Waste Building

DCA Part 2, Tier 1, Section 3.12.1, states that the RWB is a reinforced-concrete structure with a concrete roof supported on a steel frame. DCA Part 2, Tier 2, Section 3.8.4.1.3, also states that the RWB is a seismic Category II structure, located approximately 7.6 m (25 ft) west of the RXB. As stated in DCA Part 2, Tier 2, Section 3.8.4.1.3, there are no safety-related SSCs in the RWB. Further, DCA Part 2, Tier 2, Section 3.2.1.4, states that the RWB is also classified as RW-IIa because of its radioactive material content and is designed in accordance with the RG 1.143 design criteria for RW-IIa. The staff's review finds that the RWB is designed to maintain its structural integrity under the design-basis loads because it meets the RG 1.143 design criteria for RW-IIa.

3.8.4.4.5 Structural Acceptance Criteria

In accordance with DSRS Section 3.8.4.II.5, the staff's review focused on verifying the consistency of the applicant's structural acceptance criteria with the structural design criteria in ACI 349, with additional guidance provided by RG 1.142 for concrete structures and AISC N690 for steel structures.

DCA Part 2, Tier 2, Section 3.8.4.5, states that the limits for allowable stresses, strains, deformations, and other design criteria for the reinforced concrete structures are in accordance with ACI 349/349R and its appendices, as modified by the exceptions specified in RG 1.142.

Structural acceptance criteria for the steel components are in accordance with AISC N690. Further, this section refers to DCA Part 2, Tier 2, Tables 3.8.4-1 and 3.8.4-2, for the load cases for the RXB and CRB. As stated in DCA Part 2, Tier 2, Section 3.8.4.3, such load combinations are based on ACI 349, as modified by RG 1.142 and AISC N690-12.

The staff reviewed the structural acceptance criteria in the DCA Part 2, Tier 2, Section 3.8.4.5, for application to the concrete and steel seismic Category I structures. 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 N690-12, to be implemented conservatively as described in SER Section 3.8.4.2. On this basis, the staff finds the information in DCA Part 2, Tier 2, Section 3.8.4.5, on the structural acceptance criteria to be acceptable.

3.8.4.4.6 Materials, Quality Control, and Special Construction Techniques

DCA Part 2, Tier 2, Section 3.8.4.6.1, indicates that the principal construction materials for structures are concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. DCA Part 2, Tier 2, Table 3.8.4-10, provides materials properties for these materials used for structural design.

3.8.4.4.6.1 Concrete

DCA Part 2, Tier 2, Section 3.8.4.6.1.1, states that structural concrete used in the seismic Category I RXB and CRB conforms to ACI 349, as supplemented by RG 1.142, and ACI 301, "Specification for Structural Concrete for Buildings," issued 2010. Concrete mixes are designed in accordance with ACI 211.1, "Standard Practice for Selecting Proportions for Normal, Heavyweight, and Mass Concrete," issued 1991. Further, DCA Part 2, Tier 2, Section 3.8.4.6.1.1, lists the codes applicable to the concrete mix constituents as follows:

- 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, DCA Part 2, Tier 2, Section 3.8.4.6.1.1, addresses codes and standards used for concrete construction, including placement, inspection, and testing. These include ACI 301, ACI 304R, ACI 305.1, ACI 306.1, ACI 347, ACI SP-2, and ASTM C94.

3.8.4.4.6.2 Reinforcing Steel

DCA Part 2, Tier 2, Section 3.8.4.6.1.2, states that reinforcing steel consists of deformed billet steel bars conforming 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).

3.8.4.4.6.3 Connections

DCA Part 2, Tier 2, Section 3.8.4.6.1.3, addresses connection materials, including steel bolts and studs, anchor bolts, and weld electrodes as follows:

- 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.

With respect to quality control, DCA Part 2, Tier 2, Section 3.8.4.6.2, refers to DCA Part 2, Tier 2, Chapter 17, for the details of the QAP.

The staff's review confirmed that the aforementioned material specifications are within the scope of the primary design codes; that is, ACI 349 and AISC N690 or other referenced codes and standards are within the scope of the primary design codes. Therefore, the staff finds these material specifications to be acceptable.

3.8.4.4.6.4 Partition Walls in the Reactor Building

The RXB has interior steel partition walls that are not part of the RXB's lateral force resisting system. These walls are designed as steel box-type walls filled with nonstructural concrete to provide radiation protection. These walls are nonseismic Category I and are designed consistent with the seismic interaction criteria in DSRS Section 3.7.2.II.8. Specifically, these walls are analyzed and designed to prevent their failure under SSE conditions. In its analysis and design evaluations for these walls, the applicant used CSDRS-based 4-percent damped floor response spectra at elevation 30 m (100 ft), which the applicant established as the bounding seismic input for partition walls across the RXB elevations. The design of the steel partition anchorages is based on Appendix D to ACI 349-06 and 349.2R-07. Based on the implementation of seismic interaction criteria consistent with DSRS Section 3.7.2.II.8, the use of conservative seismic design input, and use of the seismic Category I design code for the partition wall anchorages, the staff finds the applicant's approach for the design of the steel partition walls to be acceptable.

3.8.4.4.7 Testing and Inservice Surveillance Requirements

DCA Part 2, Tier 2, 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 and AISC N690. Further, DCA Part 2, Tier 2, states that a COL applicant that references the NuScale Power Plant DC 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. This is COL Item 3.8-1. 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, DCA Part 2, Tier 2, 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.

3.8.4.4.7.1 Masonry Walls

DCA Part 2, Tier 2, Section 3.8.4.1.11, states that masonry walls are not used in the RXB or in the CRB. Hence, staff review in accordance with SRP Section 3.8.4, “Other Seismic Category I Structures,” Acceptance Criterion II.7, is not required.

3.8.4.5 Combined License Information Items

SER 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 DCA Part 2, Tier 2, Table 1.8-2.

Table 3.8.4-1: NuScale COL Information Items for Section 3.8.4

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.8-1	A COL applicant that references the NuScale Power Plant design certification 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 RG 1.160. Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements and differential displacements.	3.8.4.7
COL Item 3.8-2	A COL applicant that references the NuScale Power Plant design certification will confirm that the site independent RXB and CRB are acceptable for use at the designated site.	3.8.4.8
COL Item 3.8-4	A COL applicant that references the NuScale Power Plant design certification will evaluate and document construction aid elements such as steel beams, Q-decking, formwork, lugs, and other items that are left in place after construction but that were not part of the certified design to verify the construction aid elements do not have an appreciable adverse	3.8.4.8

Item No.	Description	DCA Part 2, Tier 2, Section
	effect on overall mass, stiffness, and seismic demands of the certified building structure. The COL applicant will confirm that these left in place construction aid elements will not have adverse effects on safety-related SSCs per Section 3.7.2.	
COL Item 3.8-5	A COL applicant that references the NuScale Power Plant design certification will verify that the reactor flange tool (RFT) and embed plates are evaluated using site-specific seismic analysis, and generate seismic loads to the reactor pressure vessel and fuel assemblies that are bounded by the certified design. The design of the structural members will be confirmed by assessing demand-to-capacity ratios for the load combinations in Table 3.8.4-23. The design of the embed plates will be confirmed by assessing demand-to-capacity ratios for the load combinations in Table 3.8.4-1 and Table 3.8.4-2, and applicable design codes in Table 3.8.4-12. In addition, the core plate in-structure response spectra for the RFT location shown in Figure B-34 through Figure B-39 of TR-0916-51502 (NuScale Power Module Seismic Analysis) shall be confirmed against the site specific spectra. If either the demands on the structural members or the embed plates exceed their capacity, or core plate motions do not maintain justifiable margin to limits for the fuel assembly, the COL applicant will address and augment the design per the criteria specified in FSAR Section 3.8.4, and the fuel assembly-imposed load limitations.	3.8.4.1.15
COL Item 3.8-6	A COL applicant that references the NuScale Power Plant design certification will verify that the construction loads applied to the pool liner plate and its support structure do not exceed 600 psf per American Concrete Institute (ACI)-347, Guide to Formwork for Concrete.	3.8.4.1.7

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 describes the review of areas relating 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 design of seismic Category I structures other than the containment.

3.8.5.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Sections 3.11 and 3.13, present the Tier 1 information for this section. This includes the design descriptions and ITAAC for the RXB and the CRB, respectively.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.8.4, and Appendix 3B, "Design Reports and Critical Section Details," provide Tier 2 information on the design of seismic Category I structures, other than containment.

The applicant described structures; applicable codes, standards, and specifications; loads and load combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and ISI requirements. The applicant also described COL information items related to structural design aspects of seismic Category I structures.

The seismic Category I foundations in the NuScale certified design application are the RXB and the CRB. The CRB sits on a separate foundation, approximately 10.4 m (34 ft) east of the RXB. A below-grade tunnel extends out from the CRB to the RXB. There is a 15-cm (6-in.) expansion gap between the end of the tunnel and the RXB walls. The RWB sits on a separate foundation, approximately 7.6 m (25 ft) west of the RXB. The applicant employed the SAP2000 computer code and the SASSI2010 computer code to obtain the static and seismic demands, respectively, for use in designing the RXB and CRB foundations. In addition, the applicant performed separate analyses in ANSYS to determine the stability checks (sliding) on the RXB and CRB foundations.

DCA Part 2, Tier 2, Appendix 3B, provides a design report for 15 critical sections in the RXB and 7 in the CRB. In accordance with DCA Part 2, Tier 2, Appendix 3B, the applicant selected these 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: In DCA Part 2, Tier 1, Tables 3.11-2 and 3.13-1 for the RXB and CRB, respectively, provide the ITAAC items for seismic Category I structures. These ITAAC are evaluated in Section 14.3 of this SER.

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 Section 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 1.142, Revision 2
- RG 1.160, Revision 3
- RG 1.206, Revision 0

3.8.5.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, 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 the structural design of seismic Category I structures other than the containment. Consistent with DSRS Section 3.8.4, the staff reviewed (1) the description of the structures, (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. The staff also evaluated the use of "headed reinforcing bars," in Section 3.8.5.4.5.4.

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 is in accordance with DSRS Acceptance Criterion 3.8.5.II.1.

DCA Part 2, Tier 2, 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 Power Plant. The applicant identified the RXB and CRB as seismic Category I. A tunnel connects these buildings.

DCA Part 2, Tier 2, Section 3.8.5.1, describes the dimensions and steel reinforcement patterns of the RXB and CRB basemats as follows.

3.8.5.4.1.1 Reactor Building Foundation

The RXB basemat dimensions are 109.4 (359 ft) in the E-W direction, 49.7 m (163 ft) in the N-S direction, with a minimum thickness of 3 m (10 ft). The top of concrete elevations of the foundation, the refueling pool area, the elevator area, and the sumps area are elevations 7.3 m, 5.8 m, 5.2 m, and 6.1 m (24 ft, 19 ft, 17 ft, and 20 ft), respectively.

The steel reinforcing bars at the perimeter of the RXB basemat extend 4.6 m (15 ft) from the centerlines of the exterior wall and consist of six layers of #11 bars centered at 30 cm (12 in.) each way (N-S and E-W) at top and bottom surfaces, with two-legged stirrups of #6 bars centered at 30 cm (12 in.) each way.

The steel reinforcing bars at the interior sections of the basemat consist of four layers of #11 bars centered at 30 cm (12 in.) each way (N-S and E-W) at top and bottom surfaces, with one-legged stirrups of #6 bars centered at 30 cm (12 in.) each way.

3.8.5.4.1.2 Control Building Foundation

The CRB basemat dimensions are 40 m (130 ft) in the E-W direction, 27.7 m (91 ft) in N-S direction, with a thickness of 1.5 m (5 ft).

The CRB reinforcement pattern at the perimeter of the basemat consists of four layers of #11 bars centered at 30 cm (12 in.) each way (N-W and E-W) at top and bottom, with two-legged stirrups of #6 bars centered at 30 cm (12 in.) each way, and at the center regions are three layers of #11 bars centered at 30 cm (12 in.) each way at top and bottom with two-legged stirrups of #6 bars 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 also met DSRS Acceptance Criterion 3.8.5.II.1.

The staff's Phase 2 regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML18324A551) reviewed the applicant's documents to obtain additional information on the NuScale design of the seismic Category I structures. During the audit, the staff also reviewed the following structural concrete drawings to ensure that the reinforcement patterns of RXB and CRB basemats in DCA Part 2, Tier 2, Section 3.8.5, are consistent with the applicant's structural concrete drawings:

- RXB drawings:
  - NP-12-00-F010-CD-1697-S54, Revision 5, "RXB Structural Concrete Drawing"
  - NP-12-00-F010-CD-1697-S09, Revision 5, "RXB Structural Concrete Drawing"
  - NP-12-00-F010-CD-1697-S11, Revision 5, "RXB Structural Concrete Drawing"
  - NP-12-00-F010-CD-1697-S23, Revision 5, "RXB Structural Concrete Drawing"
  - NP-12-00-F010-CD-1697-S54, Revision 5, "RXB Structural Concrete Drawing"
- CRB drawing:
  - NP-12-00-F170-CD-2330-S44, Revision 4, "CRB Structural Concrete Drawing"

Based on the staff's review of the structural concrete drawings, the staff confirmed the applicant's description of the RXB and CRB basemat reinforcement patterns and that the thicknesses of the basemats in DCA Part 2, Tier 2, are as shown in the structural concrete drawings listed above. During the audit, the applicant described the rationale for excluding geometric discontinuations (vertical basemat depressions) within the RXB and CRB basemats (e.g., RXB refueling platform, RXB sumps (12 places), RXB and CRB elevator shaft) from its analytical models (SASSI2010, SAP2020, and ANSYS). The applicant clarified that it considered flat-bottom basemats in the analytical analyses to obtain conservative results in stability evaluations without affecting the results of the settlement and design evaluation because, as shown in the structural concrete drawings, the RXB and CRB basemats maintain the minimum required thicknesses with reinforcement patterns. Based on the above explanation, the staff concludes that the consideration of flat-bottom RXB basemats in the analytical analyses are acceptable since the sliding displacement results in Table 3.8.5-11, "Reactor Building Sliding Displacements for Soil Type 7, 8 and 11 (Dead Weight + Buoyancy)," and Table 3.8.5-12, "Control Building Sliding and Uplift Displacements for Soil Type 7 and 11," for RX and CRB, respectively, are very small (2.8 millimeters (mm) (0.11 in.) for RXB and 1.12 mm (0.044 in.) for CRB) and would have a negligible effect on the RXB and CRB vertical basemat depressions The staff concludes that the nonlinear stability evaluations for RXB and CRB are conservative without vertical basemat depressions and the effects of small sliding displacements are negligible on the vertical basemat depressions.

3.8.5.4.2 Applicable Codes, Standards, and Specifications

The staff reviewed the applicable codes, standards, and specifications 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.2.

DCA Part 2, Tier 2, Section 3.8.4.2, describes the codes, standards, and specifications used for the design and construction of the RXB and CRB.

In DCA Part 2, Tier 2, Section 3.8.4.2.1, "Design Codes and Standards," and Section 3.8.4.2.2, "Regulatory Guides," list industrial design codes and standards and RGs applicable to the design and construction of seismic Category I structures (RXB and CRB). The applicant will use the latest endorsed edition of the ASTM standards at the time of the construction. Therefore, the applicant did not provide the editions of the ASTM standards in DCA Part 2, Tier 2, Section 3.8.4.2.

3.8.5.4.3 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.

DCA Part 2, Tier 2, Section 3.8.4.4.3, describes loads and load combinations for the design and construction of the RXB and CRB. In DCA Part 2, Tier 2, Tables 3.8.4-1 and 3.8.4-2 provide the load cases to be designed for the RXB and CRB, respectively. SER Section 3.8.4.4.3 evaluates loads and load combinations of the foundations for the RXB and CRB.

DCA Part 2, Tier 2, Section 3.8.4.3.22, "Other Loads," describes buoyancy forces (B), construction loads, and operation with less than 12 NPMs:

- DCA Part 2, Tier 2, Section 3.8.4.3.22.1, "Buoyant Force (B)," explains that the buoyant force is the upward pressure exerted on the bottom of the foundation during a saturated condition. The applicant correctly described the buoyant force as equal to the volume of the building below grade multiplied by the density of water.

- The applicant determined the buoyancy forces by multiplying the volume of the building below grade by the density of the water. The applicant correctly calculated the buoyancy forces for the RXB and CRB as 1,376.7 MN (309,495 kips) and 180.2 MN (40,500 kips), respectively. The buoyant forces are exerted in the upward direction on the bottom of the basemats during saturated conditions.

- DCA Part 2, Tier 2, Section 3.8.5.6.6, "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. There would not be any 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 12 NPMs. DCA Part 2, Tier 2, Section 3.7.2.9.1, "Effects of Operation with Less than Twelve NuScale Power Modules," and Section 3.7.2.9.1.5, "Conclusion of Study," report that the difference in results between operation with 12 NPMs and operation with fewer NPMs in place is small and within the capacity of the building design. However, the applicant also issued COL Item 3.7-10, which calls for the COL applicant to perform a site-specific configuration analysis that includes the RXB with applicable configuration layout of the desired NPMs.

The staff's Phase 2 regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML18324A551), reviewed calculation EC-F012-3683, Revision 2, "Design of RXB Pool Liner," to determine the justifications of structural integrity of the RXB pool liner. The stainless steel RXB liner was designed to the requirements of ASME BPV Code, Section III, Division 2. The stainless steel RXB liner is used as a permanent form during construction and loading because the load combinations also considered concrete pour. The consideration of construction loads of concrete pour pressure of 28.7 kPa (600 psf), based on ACI 347, applied to the pool liner is addressed in Section 3.8.5.4.5.8 of this SER.

3.8.5.4.3.1 Load Combinations for Stability Assessment:

The applicant correctly considered five load combinations for the assessment of basemat stability for flotation, uplift, sliding, and overturning, in accordance with DSRS Acceptance Criterion 3.8.5.II.3:

A. $D + H + E_{OBE}$
B. $D + H + W$
C. $D + H + E_{SSE}$
D. $D + H + W_t$
E. $D + B$

The applicant defined the dead weight of a structure as "D"; the buoyant force as "B"; the lateral static soil pressure as "H"; seismic loads as "E_{SSE}"; and loads generated by the design-basis tornado causing tornado wind pressure, tornado-created differential pressure, and tornado-generated missiles as "W_t."

The applicant did not analyze for the load combinations A, B, and D because, according to DCA Part 2, Tier 2, Section 3.8.4, the OBE is one-third of the SSE, and wind loads are bounded by the SSE. Thus, the applicant concluded that the load combinations C and E are bounding for the stability assessments for the RXB and CRB structures.

The applicant described the loads as dead load, buoyant force, and seismic loads used for the stability of RXB and CRB and also provided the values for the dead weights and buoyant forces.

Based on the review, the staff finds that the loads and load combinations C and E are bounding for the stability assessments for the RXB and CRB structures and are acceptable because they are in accordance with DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.2 Lateral Soil Force and Seismic Loads

DCA Part 2, Tier 2, Section 3.8.5.3.1, "Lateral Soil Force and Seismic Loads," states that the RXB and CRB are embedded structures; therefore, surrounding soil imposes lateral soil pressures to the embedded structure. The applicant calculated the soil pressure using the backfill soil with a density of 2,100 kilograms per cubic meter (kg/m^3) (130 pounds per cubic foot (lb/ft^3)) and assumed a soil friction angle of 30°. The applicant calculated the coefficient of friction (COF) between soil and basemat as 0.58 from COF = tan(30°) = 0.58.

In DCA Part 2, Tier 1, Table 5.0-1 and Table 2.0-1, "Site Parameters," provide a minimum static COF of 0.58 for all interfaces between basemat and soil. DCA Part 2, Tier 2, Section 3.8.5.4.1.2, "RXB Basemat Analysis Model Description," uses the COF of 0.50 between the RXB walls and soil. DCA Part 2, Tier 2, Table 3.8.5-1, "RXB Stability Evaluation Input Parameters," gives the COFs as 0.50 between walls and soil and 0.58 between basemat bottom surface and soil for the RXB basemat. DCA Part 2, Tier 2, Section 3.8.5.4.1.4, uses a COF of 0.50 between the CRB walls and soil, and a COF of 0.58 for the bottom surface of the CRB basemat and soil.

DCA Part 2, Tier 2, Section 3.8.4.3.3, describes the values of total lateral static effective soil forces on walls. DCA Part 2, Tier 2, Section 3.8.5.3.1, provides the equation to determine the total lateral static effective soil forces on walls.

In DCA Part 2, Tier 2, Tables 3.8.5-1 and 3.8.5-9, list the surcharge load (σ_h) as 12 kPa (250 psf), which is used in the design calculations for the RXB and CRB embedded walls. Based on the review of the site layout in DCA Part 2, Tier 2, Figure 1.2-1, the staff is not clear whether the surcharge loads are to be different for the RXB walls because of the effect of adjacent buildings (e.g., turbine generator buildings, RWB). The applicant provided the technical basis as an addition to the other surcharge loadings resulting from the SASSI2010 SSI analysis.

DCA Part 2, Tier 2, Section 3.8.5.3.1, calculates the lateral soil forces for the RXB and CRB. The forces on the RXB walls are calculated as 208,920 kN (46,967 kips) for the north and south walls and 95,321 kN (21,429 kips) for the east and west walls. The staff performed an independent check of the calculations for the CRB and determined that the applicant calculated the forces correctly. In DCA Part 2, Tier 2, Tables 3.8.5-2 and 3.8.5-9 provide the static effective soil forces for the RXB and soil pressure for the CRB, respectively.

DCA Part 2, Tier 2, Table 3.8.5-3, "Seismic Base Reactions," gives the RXB seismic reaction forces obtained at the base springs from 68 load combinations from two different RXB models, two concrete conditions (cracked and uncracked with 7-percent damping), four soil types (7, 8, 11, and 9), and six time histories (Capitola, Chi-Chi, El Centro, Izmit, Yermo, and Lucerne). DCA Part 2, Tier 2, Table 3.8.5-3, also presents the CRB seismic reaction forces, which are obtained from two concrete conditions, two soil types (7 and 9), and six time histories. The maximum seismic reaction forces at the bases come from the triple building model (RWB+RXW+CRB) and from different time histories, but they are all from Soil Type 7. DCA Part 2, Tier 2, Table 3.8.5-3, provides the maximum seismic base reactions in global directions (see DCA Part 2, Tier 2, Figure 3.7.2-3, for global coordinates) as F_X (E-W) = 1,538.4 MN (345,847 kips), F_Y (N-S) = 1,268.8 MN (285,248 kips), and F_Z (vertical) = 1,190.5 MN (267,641 kips).

Based on the review, 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.

3.8.5.4.3.3 Friction-Resistant Loads

DCA Part 2, Tier 2, Section 3.8.5.3.2, describes the friction-resistant loads. The friction-resistant loads against the sliding of RXB and CRB consist of (1) total sliding frictional resistance on the foundation surface from effective vertical load ($D_{effective}$) and (2) total sliding frictional resistance on embedded wall surfaces from static soil pressure.

The friction-resistant loads against overturning consist of total restoring moment from frictional resistance on embedded wall surfaces from effective static soil pressure. The applicant described the effective soil pressure as the soil pressure induced by the ground water table.

Based on the review, the staff found the applicant's description in the DCA acceptable for describing the friction-resistant loads for RXB and CRB by (1) total sliding frictional resistance on the foundation surface from effective vertical load ($D_{effective}$) and (2) total sliding frictional resistance on embedded wall surfaces from static soil pressure. The applicant's description also met DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.4 Effective Vertical Load

DCA Part 2, Tier 2, Section 3.8.5.3.3, describes the effective vertical load. The effective vertical load is an important stabilizing force for stability evaluations of the buildings with two components of the dead weight of the building and buoyancy load from the water table at grade. DCA Part 2, Tier 2, Section 3.8.5.3.3, calculates the effective dead weights ($D_{effective}$) of the RXB and CRB by subtracting the dead weight of the buildings (D_{RXB} and D_{RXB}) from the buoyancy forces (B_{RXB} and B_{CRB}) and lists them as 1,368.7 MN (307,702 kips) and 23,460 kN (5,274 kips), respectively.

Based on the review, the staff finds the applicant's approach acceptable for determining the effective dead weight of RXB and CRB by subtracting the total weight of the buildings from the buoyancy loads. The applicant's description also meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.5 Design and Analysis Procedures

DSRS Section 3.8.5 provides review guidance pertaining to the design and analysis procedures of foundations. DCA Part 2, Tier 2, Section 3.8.4.1, "Description of Foundations," describes the basemat reinforcement pattern of the RXB foundation. DCA Part 2, Tier 2, Appendix 3B, describes the structural design and analysis of the RXB and CRB. By letter dated December 20, 2018 (ADAMS Accession No. ML18354B330), the applicant also addressed

(1) the capacity of sections, forces and moments at critical locations, and design checks, (2) boundary conditions for each foundation model, (3) soil stiffness conditions, and (4) settlement evaluations in DCA Part 2, Tier 2, Sections 3.8.5.1, 3.8.5.6.4, 3B.2.3.1; Tables 3.8.5-7c, 3.8.5-7d, and 3.8.5-19; and Appendix B Tables 3B-62 through 3B-65 and Figures 3B-86 through 3B-89. DSRS Section 3.8.5.II.4 provides review guidance on maximizing the bending moments used in the design of foundations the applicant provides from "stiff and soft spots." In DCA Part 2, Tier 2, Section 3.8.5.4 and Table 1.8-2, the applicant provided a COL Item 3.8-3 asking for the COL applicant to identify local "stiff and soft spots" in the foundation soil and address these in the design of foundations, as necessary.

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 and ensured 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.

3.8.5.4.3.6 Foundation Basemat Analysis

The basemats are analyzed and designed for static and seismic loads. In some areas, the linear analyses did not provide acceptable results; therefore, the applicant performed nonlinear analyses with acceptable results.

The staff's Phase 2 regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML18324A551) reviewed the applicant's related documents to obtain additional information on the NuScale design of the seismic Category I structures. During the audit, the staff performed spot reviews of the following applicant's calculations:

- Calculations related to the stability checks (uplifting, sliding, and overturning) for RXB and CRB: EC-F010-3629, Revision 5, "Flotation, Sliding and Over Turning Stability Evaluations for NuScale RXB," and EC-F170-3932, Revision 2, "Flotation, Sliding and Over Turning Stability Evaluations for NuScale CRB." During the audit, the staff performed spot checks of these calculations. The staff confirmed the use of nonlinear analyses and further discusses the stability checks based on the applicant's responses to RAIs in the following sections of this SER.

- Calculations related to the settlement checks (maximum vertical settlement, maximum tilting settlement, maximum allowable differential settlement between buildings, and maximum angular distortion) for the RXB and CRB (tunnel area): EC-F010-4135, Revision 3, "NuScale RXB Responses Including Foundation Differential Settlement Effect from Triple Building Model." During the audit, the staff performed spot checks of the calculation related to settlement and held technical discussions related to settlement evaluations with the applicant's engineers. The staff was not able to locate any evaluation related to the "angular distortion" for the RXB and CRB basement in the applicant's calculations. This evaluation is required since the static loading configuration is irregularly distributed on the RXB basemat. Therefore, the staff requested the applicant to provide an assessment of the "angular distortion" of the basements. During the audit, the applicant provided the maximum local flexural tilt angles of RXB basemat. The applicant also provided a figure showing a vertical settlement contour of the RXB basemat under static loading conditions and calculated local tilt angles of 0.00744 degrees and 0.00784 degrees using horizontal (E-W) and diagonal (N-W) distances, respectively. The staff concluded that the calculated "angular-distortion" angles (local tilt angles) are small and are acceptable. Even though the applicant did not provide any evaluation related to “angular distortions” for the CRB basemat, the staff finds it acceptable as the CRB basemat static loading is about evenly distributed, because it should result in very small values of “angular distortions” in the CRB basemat. The applicant addressed this issue in its response dated December 20, 2018 (ADAMS Accession No. ML18354B330), and the issue is discussed below.

- Calculations related to the bearing pressure checks for RXB and CRB basemats: EC-F010-3870, Revision 3, “Seismic Soil-Structure Interaction Analysis of NuScale RXB for Structure Response,” and EC-F170-3758, Revision 2, “Seismic Soil-Structure Interaction Analysis of NuScale CRB for Structure Response.” The staff confirmed that, DCA Part 2, Tier 2, Section 3.8.5.6.7, “Basement Soil Pressure along Basemat Edges (Toe Pressure),” describes the results from this calculation for the CRB toe pressure, as discussed below in a letter dated August 23, 2018 (ADAMS Accession No. ML18235A280). In addition, DCA Part 2, Tier 2, Table 3.8.5-15, “Average Soil Bearing Pressures (Toe Pressures) along Edges of CRB Basemat,” provides the results of average toe pressures for the CRB, which are less than the minimum soil-bearing pressure capacity of 3.59 MPa (75 kilopounds per square foot (ksf)), as specified in DCA Part 2, Tier 2,, Table 2.0-1 and Section 2.5.4, “Stability of Subsurface Materials and Foundations.”

- Calculations related to the designs of the RXB and CRB basemats: EC-F010-4170, Revision 1, “RXB Phase 3 Design Evaluation,” and EC-F170-3555, Revision 2, “Structural Design Phase 3 Evaluation for NuScale CRB.” The staff reviewed the tables related to the D/C ratios for RXB and CRB basemats. Although the staff determined the D/C ratios for RXB and CRB basemats are calculated to be less than 1.0, for the cases where the localized D/C ratios (in a single element) would exceed the ratio of 1.0, the applicant described the adjacent elements that are considered to average demand forces and moments to determine new realistic D/C ratios for that cross-section (with multiple elements). The staff found that the applicant’s approach of averaging demand forces and moments over wall or slab sections is acceptable because it is a realistic engineering practice to consider adjacent finite elements’ demand forces and moments when calculating D/C ratio exceedances over a single finite element. The applicant also described this approach in DCA Part 2, Tier 2, Appendix 3B, Section 3B.1.1.1, “Averaging Demand Forces and Moments.”

- Calculations that consider the reduction of 50 percent of soft-soil stiffness (Soil Type 11) in the design of the RXB and CRB basemats: EC-F010-4135, Revision 3, “NuScale RXB Responses Including Foundation Differential Settlement Effect from Triple Building Model.” Based on the review, the staff confirmed that the applicant considered 50-percent stiffness reduction of soft-soil stiffness (Soil Type 11) in the calculation. Furthermore, the applicant’s response dated August 16, 2018 (ADAMS Accession No. ML18228A859), below, states that the design conservatively considered the reduction of 50 percent of the soft-soil profile (Soil Type 11) and determined the differential basemat settlements of RXB and CRB basemats.

- Standalone and triple building models used in the RXB and CRB foundation designs: EC-F010-3870, Revision 3, and EC-F170-3758, Revision 2. Based on its review, the staff confirms that the applicant used two layers of solid elements in the RXB standalone and triple building models in the SASSI2010 and SAP2000 models. The applicant also addressed this issue in its response dated June 29, 2018 (ADAMS Accession No. ML18180A404), below, and provided markups to DCA Part 2, Tier 2, Section 3.7.2.1.1.1, “SAP2000,” and Section 3.7.2.1.3.5, “Control Building,” and the associated tables, tabulating software (SASSI2010, SAP2000), buildings included in the model, number of layers and types of elements used in the basemat model, and results used for the designs of the RXB and CRB basemats.

- The staff reviewed the reports describing the governing load combination 10 from DCA Part 2, Tier 2, Table 3.8.4-1 (Equation 9-6 of ACI 349), used for the design of the RXB and CRB foundations: EC-F010-4170, Revision 1, and EC-F170-3555, Revision 2. Based on the review, the staff confirms the applicant used load combination 10 (Equation 9-6 of ACI 349) to design the basemats in those reports. Therefore, the applicant’s description meets DSRS Acceptance Criterion 3.8.4.II.3.

- Report related to design criteria, ES-0303-3677, Revision 2, “Civil and Structural Design Criteria”: The staff confirmed that the tabulation of the COF values in DCA Part 2, Tier 2, Table 2.0-1, and calculations in DCA Part 2, Tier 2, correctly reflected the use of COFs.

- Calculation EC-F012-3683, Revision 2: The staff reviewed the calculation to determine the justifications of structural integrity of the RXB pool liner. The applicant stated that the stainless steel RXB liner was designed to the requirements of ASME BPV Code, Section III, Division 2. Further, the stainless steel RXB liner is used as a permanent form during construction and loading because concrete pour was also considered in the load combinations. The consideration of construction loads of concrete pour pressure of 28.7 kPa (600 psf) based on ACI 347 being applied to the pool liner is addressed in Section 3.8.5.4.5.8 in this SER.

- The staff reviewed the calculation EC-F010-1731, Revision 3, “Structural Analysis for NuScale Building Using SAP2000,” to determine how the applicant considered the cracked concrete stiffness properties in the models to account for any concrete cracking during a seismic event. In this report, the applicant described how to reduce the stiffness components to perform the seismic analyses under cracked concrete condition. The applicant provided tables tabulating cracked stiffness rigidity equation of “axial (E_cA_g), shear (G_cA_w) and flexural (E_cI_g).” The applicant considered two approaches using two variables (element thickness and Young’s modulus) to determine the effective cracked stiffness values for each reinforced concrete element: (1) effective stiffness of reinforced concrete elements if thickness is reduced and (2) effective stiffness of reinforced concrete elements if modulus of elasticity is reduced. As described in the report, the applicant selected the approach that reduced the concrete element thickness in its seismic analyses. As discussed in a letter dated February 14, 2019 (ADAMS Accession No. ML19045A493), DCA Part 2, Tier 2, Table 3.7.1-7a, “Effective Stiffness Changes of Cracked Reinforced Concrete Finite Element Model Members,” and Section 3.7.1.2.2, “Structural Damping,” clearly identify the cracked concrete properties used in the FEAs. The staff concluded that the applicant’s approach meets DSRS Acceptance Criterion 3.7.1.II.4 and is therefore acceptable.

3.8.5.4.3.6.1 Analysis of Reactor Building Basemat

DCA Part 2, Tier 2, Table 3.8.5-1, “RXB Stability Evaluation Input Parameters,” provides the design input parameters to perform the stability (flotation, sliding, and overturning) evaluation of the RXB. The staff reviewed the design input parameters in DCA Part 2, Tier 2, Table 3.8.5-1, and concluded that the information listed is acceptable to perform the stability evaluation of the RXB.

3.8.5.4.3.6.2 Reactor Building Basemat Analysis Model Description

DCA Part 2, Tier 2, Section 3.8.5.4.1.2, describes the RXB basemat model. The applicant described the forces and moments in all structural elements determined from static and seismic demands using standalone and combined SAP2000 and SASSI2010 RXB models.

The applicant applied out-of-plane pressure loads extracted from static and dynamic analyses, including buoyancy pressure, to determine internal shear and moments for the design of the RXB basemat. In DCA Part 2, Tier 2, Figures 3.8.5-2 to 3.8.5-7 show the static and seismic pressure contours and bending moments in the RXB basemat.

Based on its review, the staff issued the RAIs described below, requesting the applicant to provide additional information related to the RXB basemat models.

DCA Part 2, Tier 2, Section 3.8.5.4.1.2, states, "The SAP2000 model was created modeling the RXB basemat with solid elements in order to calculate forces and moments in the basemat." DCA Part 2, Tier 2, page 3.8-59, states, "Figure 3.8.5-1 shows the SAP2000 model." The applicant described the RXB basemat in the SAP2000 models (standalone and triple building) as composed of two layers of 1.5-m (5-ft)-thick solid elements with fixed-based boundary conditions to determine static bearing pressures and settlements. The applicant also described that there are two SASSI2010 models (standalone and triple building) with rigid soil springs connecting the RXB to the surrounding soil, used to address seismic load combinations to determine the seismic bearing pressures. The applicant further stated that the two 1.5-m (5-ft)-thick layers of concrete solid elements were replaced by a single layer of shell element in the standalone SAP2000 RXB model (as shown in DCA Part 2, Tier 2, Figure 3.8.5-1), where the pilasters on the building perimeter and walls within the footprint are connected to the shell elements acting as inverted supports to the RXB basement, and the determined static and seismic bearing pressures applied as out-of-plane pressure loads to determine internal moments and shear for the design of RXB basemat. In DCA Part 2, Tier 2, Revision 2, Figures 3.8.5-1 to 3.8.5-7 show static and seismic RXB base pressure contours and static and seismic RXB bending moments.

The staff finds that the applicant's modeling approach is acceptable because it uses two layers of 1.5-m (5-ft)-thick concrete solid elements to model the RXB basemat in both the standalone RXB and triple building model (RXB, CRB, and RWB) using the SAP2000 and SASSI2010 software to determine static and seismic bearing pressures. The applicant also described, and the staff finds acceptable, that two 1.5-m (5-ft)-thick layers of concrete solid elements were replaced by a single layer of shell elements in the standalone RXB model using the SAP2000 software, and the applicant then applied the calculated static and seismic bearing pressures as out-of-plane pressure loads to determine internal moments and shear for the design of the RXB basemat. The applicant's markups are consistent with this description, addressing the RXB model in DCA Part 2, Tier 2, Section 3.8.5.4.1.2.

The applicant described the development of a partial RXB basemat model in the SAP2000 software, using the bottom part of the entire RXB model, where (1) basemat elements are changed from solid elements to shell elements, (2) all structural components 3 m (10 ft) above the basemat are deleted, (3) walls, pilasters, and columns are restrained in all six directions at 3 m (10 ft) above the basemat, and (4) the envelope soil-bearing pressures, because of static and dynamic loads, are applied as uniform pressures to the basemat. DCA Part 2, Tier 2, Section 3.7.2.1.1, “Computer Programs,” describes that the RXB and CRB basemats are designed using combinations of different models from three commercially available computer programs (SAP2000, ANSYS, SASSI2010), and by extracting the structural responses from the building models and then applying them to the separate basemat models to determine structural design forces and moments for the basemats. DCA Part 2, Tier 2, Table 3.7.2-49, “Building Models Used for RXB Basemat Design,” Table 3.7.2-50, “Basemat Model Used for RXB Basemat Design,” Table 3.7.2-51, “Building Models Used for CRB Basemat Design,” and Table 3.7.2-52, “Basemat Model Used for CRB Basemat Design,” describe in tabular format the software used (SASSI2010, SAP2000), buildings included in the model, number of layers and types of element used in the basemat model, and results used for the designs of RXB and CRB basemats.

DCA Part 2, Tier 2, Table 3.8.5-5, “Factors of Safety-RXB Stability,” gives the values of factor of safety (FOS) of RXB stability from linear analysis. The linear analysis resulted with an FOS less than 1 for the RXB sliding. Therefore, the applicant performed a nonlinear analysis for RXB sliding using ANSYS and provided DCA Part 2, Tier 2, Table 3.8.5-6, “RXB ANSYS Model Summary,” which lists the types and numbers of elements used in the RXB ANSYS model. The soil domain is uncoupled from the RXB building by creating two coincident joints or nodes in the ANSYS finite element mesh to permit sliding under seismic conditions. The coincident nodes are the nonlinear contact region, one belonging to the RXB and one belonging to the backfill soil, as shown in DCA Part 2, Tier 2, Figure 3.8.5-9. The applicant defined the nonlinear element between the coincident nodes as CONTA178 element, as shown in Figures 3.8.5-13 and 3.8.5-14, and also provided Figure 3.8.5-15, “Nonlinear Contact Element between Backfill and Surrounding Soil,” which illustrates the CONTA178 element where forces are transferred between the end node-i and node-j when the gap is closed and consequently transmitting compression only. The respective acceleration time histories from SASSI2010 for Soil Types 7, 8, and 11, were applied uniformly to all the boundary nodes in the ANSYS model. In DCA Part 2, Tier 2, Figures 3.8.5-17 through 3.8.5-25 show the input acceleration time history for Soil Types 7, 8, and 11.

Based on the information in DCA Part 2, Tier 2, and confirmatory review performed during the regulatory audit (ADAMS Accession No. ML18324A551), the staff concludes that the applicant provided sufficient information to describe the RXB SAP2000, SASSI2010, and ANSYS models. The staff determined that this information addressed the modeling and software used in the design of the RXB basemat and met DSRS Acceptance Criterion 3.8.5.II.4.E and therefore is acceptable.

3.8.5.4.3.6.3 Analysis of Control Building Basemat

DCA Part 2, Tier 2, Section 3.8.5.4.1.3, “Analysis of Control Building Basemat,” describes the analyses performed for the CRB.

The applicant obtained the static load results of the static forces and moments in the basemat from the standalone and the combined CRB SAP2000 models.

The applicant used the basemat solid element stresses obtained from the SASSI analysis as a result of the dynamic loads. The applicant determined the axial and shear forces by multiplying the axial and shear stresses by the solid element thickness and the bending moments using a separate SAP2000 shell element basemat model.

The applicant determined the bending moments by considering the CRB tunnel basemat as a simple-supported, one-way slab spanning between the exterior and middle tunnel walls. The applicant conservatively averaged the end moments of the exterior and middle walls and added to the simple-supported moments at the center of the span. The applicant used the resultant moment for both global X and Y axes.

In DCA Part 2, Tier 2, Revision 2, Figures 3.8.5-2a and 3.8.5-3a show the CRB basemat contour pressure for static and seismic load combinations. In DCA Part 2, Tier 2, Revision 2, Figures 3.8.5-4a and 3.8.5-5a show the CRB basemat static M_{yy} and M_{xx} from the standalone SAP2000 mode, and Figures 3.8.5-6a and Figure 3.8.5-7a show the CRB basemat seismic M_{yy} and M_{xx}.

Based on its review, the staff concludes that DCA Part 2, Tier 2, Revision 2, Section 3.8.5.4.1.3, provides sufficient information to describe types of analyses performed to determine the internal forces in the CRB basemat. The applicant's description also meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.6.4 Linear Analysis

DCA Part 2, Tier 2, Section 3.8.5.4.1.3, describes the linear analysis for the CRB. The acceptance criteria for flotation/uplift, sliding, and overturning are based on an FOS determined from the ratio of the driving force to the resisting force. DCA Part 2, Tier 2, Section 3.7, "Seismic Design," discusses how the applicant performed these analyses statically using the maximum forces from the combinations of soil profiles, time histories, and cracked and uncracked conditions discussed. However, the applicant concluded that the FOS performed for the CRB yielded unacceptable results (less than 1.1 FOS) for uplift stability. Therefore, the applicant performed a nonlinear analysis to determine the uplift, sliding, and overturning of CRB.

DSRS Section 3.8.5 provides review guidance on the stability of foundations. DCA Part 2, Tier 2, Table 3.8.5-1, gives the COF between walls and soil and between basemat bottom surface and soil. DCA Part 2, Tier 2, Table 3.8.5-9, "CRB Stability Evaluation Input Parameters," tabulates values of COF between wall and soil and between basemat and soil (static analysis and dynamic analysis). The staff performed a Phase 2 regulatory audit on December 3–7, 2018 (ADAMS Accession No. ML18324A551), which also included a review of additional information related to the list of COF values. The staff confirmed that the values of COF "between soil and basemats" and "between soil and walls" of the RXB and CRB are correctly provided in DCA Part 2, Tier 2, Table 2.0-1 and Section 3.8.5, and meet the DSRS Acceptance Criterion 3.8.5.II.4.E and therefore are acceptable.

3.8.5.4.3.6.5 Control Building Basemat Nonlinear Analysis Model Description

DCA Part 2, Tier 2, Section 3.8.5.4.1.4, describes the nonlinear analysis for the CRB. DCA Part 2, Tier 2, Table 3.8.5-13, "Control Building Sliding and Uplift Displacements for Soil Type 7 and 11," provides the model summary showing quantity of elements in the ANSYS structural analysis model, including joints, frame elements, shell elements, solid elements, and links/supports. The applicant referred to DCA Part 2, Tier 2, Figure 3.8.5-41, "SAP2000 Model for Settlement," showing that the coordinate system for the nonlinear analysis is represented by the CRB SAP2000 model—the X axis points to the east, the Y axis points to the north, and the Z axis points vertically upward.

The applicant made the following changes to the nonlinear ANSYS CRB model with fixed-based boundary conditions:

- The applicant established coincident joints and nodes for the CRB and soil in the finite element mesh to provide the independence of the CRB structure.

- The applicant created a nonlinear frictional contact region with the coincident nodes, as shown in DCA Part 2, Tier 2, Figure 3.8.5-26, “Nonlinear Contact Region between CRB and Soil,” with a COF of 0.50 (between the CRB walls and soil).

- The applicant considered three time histories for each soil type (Soil Type 11 backfill combined with the surrounding Soil Types 7 and 11) by uniformly applying the time histories from the typical skin node to the CRB and backfill soil nodes, as shown in DCA Part 2, Tier 2, Figure 3.8.5-27, “CRB Time Histories from SASSI Applied to ANSYS Model Boundary,” which are in contact with the in situ soil. The applicant selected the SASSI time histories for the Capitola input case because that case produced the largest horizontal base reactions, as shown in DCA Part 2, Tier 2, Table 3.8.5-13. In DCA Part 2, Tier 2, Figures 3.8.5-28 through 3.8.5-33 show the input acceleration time histories for Soil Types 7 and 11.

- DCA Part 2, Tier 2, Figure 3.8.5-34, “CRB Skin Nodes on Backfill Outer Boundaries for Applying SASSI Time Histories,” and Figure 3.8.5-35, “CRB Foundation Bottom Skin Nodes for Applying SASSI Time Histories,” use node-to-node CONTA178 elements between the CRB and surrounding soil. The elements directly under the CRB foundation and on the sides of the CRB have COFs of 0.55 and 0.50, respectively, defined to resist sliding.

- The applicant calculated the buoyancy pressure as 202.7 kPa (29.399 psi) at the bottom of the CRB basemat.

- The applicant considered the Poisson’s ratio effect in the static soil pressure profile from the dead weight of the backfill soil on the CRB walls as a conservative measure. DCA Part 2, Tier 2, Figure 3.8.5-37, shows the static pressure profile. The applicant also provided DCA Part 2, Tier 2, Figures 3.8.5-38 and 3.8.5-39, where the Poisson’s ratio effect produces a complex pressure distribution depending on the local flexibility of the walls for Soil Types 11 and 7, respectively.

DCA Part 2, Tier 2, Table 3.8.5-10, “CRB SAP2000, SASSI2010, and ANSYS Model Summary,” gives the quantities of types of joints, elements, and restraints of the CRB SAP2000, SASSI2010, and ANSYS models. DCA Part 2, Tier 2, Figure 3.8.5-40, “CRB SAP2000 Model with Backfill Soil,” also provides an isometric view of the CRB SAP2000 model and describes the coordinate system as the X axis pointing to the east, the Y axis pointing to the north, and the Z axis pointing vertically upward.

DCA Part 2, Tier 2, Revision 2, Section 3.8.5.4.1.4, lists the new COL Item 3.8-3, addressing the site-specific “stiff and soft spots” foundation soil, as addressed above.

Based on the information provided in DCA Part 2, Tier 2, and confirmatory review performed during the audit (ADAMS Accession No. ML18324A551), the staff concludes that the applicant provided sufficient information to describe the CRB basemat model with conditions. In addition, the applicant also listed COL Item 3.8-3, addressing the site-specific “stiff and soft spots” foundation soil in this section. Therefore, the applicant’s response also meets DSRS Acceptance Criterion 3.8.5.II.4.J.

3.8.5.4.4 Evaluation Criteria for Stability Analysis

3.8.5.4.4.1 Flotation and Uplift Stability Analysis Approach

DCA Part 2, Tier 2, Section 3.8.5.5.1, "Flotation and Uplift Stability Analysis Approach," describes the calculations of flotation for static conditions and uplift under the seismic conditions and provides equations for determining the factors of safety of flotation and uplift. DCA Part 2, Tier 2, Section 3.8.5.3, "Loads and Load Combinations," describes the load combination E for flotation and load combination C for uplift as follows:

$$FOS = F_{resisting} / F_{driving}$$
$$FOS_{flotation} = D / B$$
$$FOS_{uplift} = (D + F) / (B + R_z)$$

The applicant described the resisting force ($F_{resisting}$) for the buried portion of the structure only, which includes the components of weight of the building (D) and vertical friction from static soil pressure (F).

The applicant described the driving forces ($F_{driving}$) for uplift from the ground water and seismic motion, which includes the components of the buoyant force (B) from ground water or flood water at grade, and upward inertia (R_z).

Based on the review, the staff concludes that the applicant correctly described equations for determining the FOS for flotation and uplift stability assessments for the RXB and CRB basemats and meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.4.2 Sliding Stability Analysis Approach

DCA Part 2, Tier 2, Section 3.8.5.5.2, "Sliding Stability Analysis Approach," describes the sliding stability analysis under seismic conditions. As described in DCA Part 2, Tier 2, Section 3.8.5.3, the applicant used the load combination C to perform the sliding stability evaluation.

The applicant determined the stability evaluation by comparing the total resisting sliding forces and the total driving sliding forces for the E-W movement (Global X-direction), and for N-S movement (Global Y-direction). The RXB and CRB are deeply embedded structures; therefore, the applicant considered the frictional resistance by the interactions of soil and structure on the exterior walls and basemats.

The driving sliding force is the inertia force from seismic conditions, and the total inertia force equals the sum of all reaction forces on walls and basemat in the N-S and E-W sliding.

DCA Part 2, Tier 2, Section 3.8.5.3.3, determined that, for sliding stability evaluation, the effective dead weights ($D_{effective}$) of the RXB and CRB are important stabilizing forces.

The applicant calculated friction resistance ($R_{sliding}$) forces at the RXB basemat and CRB basemat by multiplying the effective dead weight ($D_{effective}$) with the COF of soil under the basemats. The applicant determined the $R_{sliding}$ forces for RXB and CRB as 793,861 kN (178,467 kips) and 13,607 kN (3,059 kips), respectively. The applicant described how to determine friction forces resisting the sliding of the RXB and CRB basemats. However, the applicant used the nonlinear analyses approach to assess the RXB and CRB sliding stability in DCA Part 2, Tier 2, Sections 3.8.5.4.1.2 and 3.8.5.4.1.3. The staff found the use of the nonlinear approach acceptable because it will provide realistic results in the sliding stability assessment of RXB and CRB basemats and meets DSRS Acceptance Criterion 3.8.5.II.4.N.

3.8.5.4.4.3 Overturning Stability Analysis Approach

DCA Part 2, Tier 2, Section 3.8.5.5.3, "Overturning Stability Analysis Approach," describes the overturning stability. DCA Part 2, Tier 2, Section 3.8.5.3, "Loads and Load Combinations," describes how the applicant used the load combination C to perform the overturning stability evaluation. The applicant correctly provided the following FOS equation for overturning:

$FOS_{overturning} = M_{restoring} / M_{overturning}$

The applicant determined the overturning evaluation by comparing the total resisting overturning moment and the total driving overturning moments for the N-S movement (moment about Global X-direction) and for the E-W movement (moment about Global Y-direction).

The applicant considered the RXB as a deeply embedded structure; therefore, the frictional resisting moments provided by the interactions between soil and structure on the exterior walls and basemat are considered to be resistant to overturning. The applicant also considered the restoring moment because of the effective vertical load in the evaluation. Therefore, the applicant described three components resistant to overturning: (1) friction on parallel walls, (2) friction on perpendicular walls, and (3) effective dead weight. The applicant also described in detail the calculations of RXB resultant resisting moments ($M_{N-S} = M_1+M_2+M_3$ and $M_{E-W} = M_4+M_5+M_6$) from friction in N-S overturning and E-W overturning.

Based on its review, the staff concludes that the applicant's assessment is acceptable for the formulations of resultant resistance moments from frictions and buoyant dead weight for the RXB resistance in N-S and E-W overturning. The assessment is consistent with DSRS Acceptance Criterion 3.8.5.II.4.E.

3.8.5.4.4.4 Bearing Pressure Approach

DCA Part 2, Tier 2, Section 3.8.5.5.4, "Bearing Pressure Approach," describes the average and localized bearing pressures. The applicant calculated the average static bearing pressure by dividing the building weight by the building footprint area. The applicant calculated the seismic basemat bearing pressure by the algebraic summation of reaction time histories in the springs below the basemat. The algebraic summation yields three time histories of total basemat reactions in the three directions from each seismic input.

The applicant calculated the localized soil pressure under each building's basemat using the forces in the rigid springs, which are used to connect the RXB and CRB basemats to the excavated free-field soil. The applicant determined the vertical force in a spring by dividing by the tributary area of the spring to obtain the localized nodal soil pressure.

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, for determining the static bearing pressures by dividing the buildings' weight by the footprint areas of basemats, and the seismic bearing pressures by dividing the algebraic summations of vertical reactions in the rigid springs (three orthogonal time history seismic demands divided by the tributary areas of rigid springs).

3.8.5.4.4.5 Settlement Approach

DCA Part 2, Tier 2, Section 3.8.5.5.5, "Settlement Approach," describes the foundation settlements. The applicant used a SAP2000 FEM to determine the effect of foundation differential movements; DCA Part 2, Tier 2, Figure 3.8.5-41 shows the SAP2000 model for settlement evaluation. The applicant used the soft-soil profile (Soil Type 11) 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's analyses include both the cracked and uncracked concrete conditions. The applicant determined that settlements from the static loads are negligible.

DCA Part 2, Tier 2, Section 3.8.4, uses the governing load combination 10 from DCA Part 2, Tier 2, Table 3.8.4-1, "Concrete Design Load Combinations" (Equation 9-6 of ACI 349):

$$U = D + F + H + 0.8L + C_{cr} + T_o + R_o + E_{SSE}$$

The applicant also increased the dead weight of the building to account for the live and snow loads.

DCA Part 2, Tier 2, Section 3.8.5.5.5, states the following:

> [T]he soil stiffness values are further reduced by 50 percent to amplify the effect of differential movements or settlements. The size of the soil included in the model is so large that the static displacements induced by the static loads of the structures become negligible on the edges of the free field soil model.

By letters dated October 17, 2017 (ADAMS Accession No. ML17290B267), April 4, 2018 (ADAMS Accession No. ML18094B106), and August 16, 2018 (ADAMS Accession No. ML18228A859), the applicant clarified that the soil model had extended the reduced soil stiffness values for the entire free-field soil, as shown in DCA Part 2, Tier 2, Figure 3.8.5-41. This was done to determine the static demand forces for the RXB and CRB foundation designs, and to determine maximum differential displacements within each building foundation. In its August 16, 2018 letter, the applicant also provided moment and shear capacities of RXB (3.05-m (10-ft, 0-in.)-thick basemat) and CRB (1.52-m (5-ft, 0-in.)-thick basemat) foundations. The applicant referred to DCA Part 2, Tier 2, Appendix 3B, Table 3B-36, which gives the moment and shear capacities of a 1.52-m (5-ft)-thick CRB basemat, and DCA Part 2, Tier 2, Tables 2B-37 and 3B-38, which show the demand forces and moments for the perimeter and interior of the CRB basemat, respectively. Furthermore, the staff's Phase 2 regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML18324A551), reviewed calculation ER-F010-4135, Revision 3, to determine whether the applicant's conservatively performed calculation considered the reduction of 50 percent of soft-soil profile (Soil Type 11) in the design of the RXB and CRB basemats and to determine the differential foundation settlements. The staff also reviewed DCA Part 2, Tier 2, Section 3.8.5.5.5, which described the extent of the reduction in soil stiffness in the soil model and the determination of static-demand forces and differential settlements for the RXB and CRB basemats.

Based on its review of the information submitted by the applicant and audit of the associated calculation, 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 of the SAP2000 triple building model 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 applicant's responses also meet 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 DCA Part 2, Tier 2, Section 3.8.5.6.1, "RXB Stability," the applicant determined the FOS for 16 cases for the RXB stability. DCA Part 2, Tier 2, Table 3.8.5-5, shows the cases, which include enveloping seismic loads from two RXB models (a standalone RXB model and an integrated RWB+RXB+CRB triple building model), two concrete conditions (cracked and uncracked with 7-percent damping), and Soil Types 7, 8, 9, and 11. The tabulated results in this table indicate that the linear analysis for RXB sliding stability did not yield an FOS greater than 1.0.

The applicant combined the seismic results calculated by SASSI2010 with the static results calculated by SAP2000, as provided in the following three methods, to account for the phasing issue of seismic results:

(1) by the algebraic sum by assuming all seismic results are positive: static + seismic
(2) by the algebraic sum by assuming all seismic results are negative: static – seismic
(3) by the absolute sum of static and seismic results: abs (static) + seismic

For the in-plane forces FXX and FYY, the first two combination methods give the actual ranges of the in-plane forces; shear forces and bending moments are obtained by the third, or the absolute sum, method.

The combined static and seismic results are assembled based on the reinforcing pattern described above. The reinforcement pattern has separate reinforcement for the perimeter of the RXB basemat (approximately 4.6 m (15 ft) from the edge of the basemat) and the interior of the RXB basemat.

DCA Part 2, Tier 2, Table 3.8.5-5, gives the FOS for the RXB stability; however, the RXB sliding and CRB uplift were not within the allowable limit of 1.1. Therefore, the applicant performed a nonlinear analysis for the RXB to show that sliding was insignificant. The applicant also performed nonlinear sliding, overturning, and uplift analyses for the CRB to show that sliding, overturning, and uplift are insignificant.

The applicant stated that the bearing capacity of the soil should provide an FOS of 3.0 for the static bearing pressure and an FOS of 2.0 for dynamic bearing pressure. In addition, the maximum allowable differential settlement for the RXB and CRB is 2.5 cm (1.0 in.) total, or 1.3 cm per 15 m (1/2 in. per 50 ft) in any direction at any point in either structure.

The staff confirmed that the maximum allowable tilt settlement for the RWB, RXB, and CRB shall be 2.5 cm (1 in.) total or 1.3 cm per 15 m (1/2 in. per 50 ft) in any direction at any point in any of the structures, and the maximum allowable differential settlement for the RWB, RXB, and CRB shall be 1.3 cm (1/2 in.) between the RXB and CRB, and the RXB and RWB in DCA

Part 2, Tier 1, Table 5.0-1, and DCA Part 2, Tier 2, Table 2.0-1, for the RXB, CRB, and RWB basemats.

The staff reviewed the stability evaluation and criteria in DCA Part 2, Tier 2, Section 3.8.5.6.1, for RXB stability. The staff found them to be in accordance with the guidance given in DSRS Section 3.8.5.II.5. On this basis, the staff finds the information in DCA Part 2, Tier 2, Section 3.8.5.6.1, on the RXB stability to be acceptable.

3.8.5.4.5.1.1 Reactor Building Uplift

DCA Part 2, Tier 2, Section 3.8.5.6.1.1, "RXB Uplift," describes the RXB uplift. DCA Part 2, Tier 2, Section 3.8.5.5.1, provides the FOS equations. DCA Part 2, Tier 2, Table 3.8.5-5, gives the FOS for the RXB flotation for each of the 16 cases considered, including cracked and uncracked concrete conditions; Soil Types 7, 8, 9, and 11; and the RXB model and the triple building model. The applicant calculated an acceptable FOS for overturning for each of the cases.

Dynamic Reactor Building Uplift Ratio

In DCA Part 2, Tier 2, Section 3.8.5.6.1.1.1, "Dynamic RXB Uplift Ratio," the applicant stated that the linear SSI analysis methods would be acceptable if the ground contact ratio is equal to or greater than 80 percent. The ground contact ratio can be calculated from the linear SSI analysis using the minimum basemat area that remains in compression with the soil. The applicant calculated the seismic total vertical base reactions by algebraic summation of all nodal vertical reactions of the nodes of RXB basemat. DCA Part 2, Tier 2, Table 3.8.5-4, "Seismic Vertical RXB Base Reactions and Dead Weight," gives the maximum seismic vertical reactions for the cracked and uncracked concrete conditions for the two models and shows that the seismic reactions are much less than the total dead weight reaction of 2,097.28 MN (471,487 kips). Therefore, the applicant concluded that the net reactions are always in compression.

In DCA Part 2, Tier 2, Figures 3.8.5-42 through 3.8.5-47 show that the total basemat vertical reaction time histories are under compression for the cracked and uncracked concrete conditions. Therefore, the applicant concluded that the vertical reactions are in compression, and the RXB basemat is 100 percent in contact. The staff reviewed DCA Part 2, Tier 2, Figures 3.8.5-42 through 3.8.5-47, and confirmed that, for all cases, vertical reactions are in compression because of the seismic time histories of "Capitola and Lucerne" for Soil Types 7, 8, 9, and 11. Therefore, they meet DSRS Acceptance Criterion 3.7.2.II.4 of "assuring the ground contact ratio is equal to or greater than 80 percent."

The staff also confirmed that the applicant correctly considered three acceleration components, X (E-W), Y (N-S), and Z (vertical), from each of the CSDRS and CSDRS-HF seismic inputs used in the calculations of RXB and CRB uplift ratios, and as described in DCA Part 2, Tier 2, Section 3.8.5.6.1.1.1 and Section 3.8.5.6.2.1.1, "Dynamic CRB Uplift Ratio."

3.8.5.4.5.1.2 Reactor Building Sliding

DCA Part 2, Tier 2, Section 3.8.5.6.1.2, "RXB Sliding," describes the RXB sliding. DCA Part 2, Tier 2, Section 3.8.5.5.1, provides the FOS equation as follows:

$$FOS_{sliding} = F_{resisting} / F_{driving}$$

DCA Part 2, Tier 2, Table 3.8.5-5, shows that the linear evaluations for the RXB sliding are less than the acceptable FOS. Therefore, the applicant performed a nonlinear sliding analysis to show that sliding is insignificant.

Nonlinear Analysis

The applicant provided the following figures in DCA Part 2, Tier 2, Section 3.8.5, describing the model and providing sliding at the corners of the RXB basemat:

- Figure 3.8.5-52 shows the designations used (A–D) for the locations on the RXB basemat where lateral displacements (sliding) were assessed between two end nodes of CONTA178 elements.

- Figures 3.8.5-53 through 3.8.5-60 show the E-W and N-S sliding displacements for Soil Type 7 for the four foundation locations (A, B, C, and D).

- Figures 3.8.5-61 through 3.8.5-68 show the E-W and N-S sliding displacements for Soil Type 11 for the four foundation locations (A, B, C, and D).

- Figures 3.8.5-69 through 3.8.5-76 show the E-W and N-S sliding displacements for Soil Type 8 for the four foundation locations (A, B, C, and D).

In DCA Part 2, Tier 2, Section 3.8.5.6.1.2, the applicant stated, "…a nonlinear sliding analysis has been performed to show that sliding is insignificant." In letters dated October 17, 2017, and February 21, 2018 (ADAMS Accession Nos. ML17290B267 and ML18052B566), the applicant described the nonlinear analysis methods for RXB and CRB in DCA Part 2, Tier 2, Sections 3.8.5.4.1.2 and 3.8.5.4.1.4. The applicant tabulated the sliding results in DCA Part 2, Tier 2, Table 3.8.5-11, where the maximum sliding displacement is 2.8 mm (0.11 in.). The applicant concluded that the displacements at these magnitudes would not cause any structural damage to RXB and CRB structures.

Based on its review, the staff finds the applicant's approach acceptable because the applicant performed detailed nonlinear sliding analyses using ANSYS, which would provide more realistic results as described in DCA Part 2, Tier 2, Sections 3.8.5.4.1.2 and 3.8.5.4.1.4, for the RXB and CRB, respectively. The staff also reviewed the tabulated results in DCA Part 2, Tier 2, Tables 3.8.5-11 and 3.8.5-12, and concluded that the results would not show any structural damage to RXB and CRB structures. Furthermore, the staff's Phase 2 regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML18324A551), reviewed calculations EC-F010-3629, Revision 5, and ER-F170-3932, Revision 2. Based on its review of the calculations, the staff confirmed that the applicant developed the ANSYS models for the RXB and CRB to perform nonlinear analyses by uncoupling the soil domain from the buildings, where they are connected with contact elements (CONTA178), to permit sliding under seismic conditions. The staff confirmed that the calculated results for sliding displacements are small and therefore meet DSRS Acceptance Criterion 3.8.5.II.4.E. The staff also concluded that the sliding displacements of deeply embedded structures would be a minor consideration under seismic conditions. Therefore, the staff finds that the information in DCA Part 2, Tier 2, Section 3.8.5.6.1.2, is acceptable.

3.8.5.4.5.1.3 Reactor Building Overturning

DCA Part 2, Tier 2, Section 3.8.5.6.1.3, "RXB Overturning," describes the RXB overturning. DCA Part 2, Tier 2, Section 3.8.5.5.3, provides the FOS equation as follows:

$FOS_{overturning} = M_{restoring} / M_{overturning}$

DCA Part 2, Tier 2, Table 3.8.5-5, provides the results of FOS for overturning for each of the 16 cases considered, including cracked and uncracked concrete conditions, Soil Types 7, 8, 9, and 11, and for the RXB model and the triple building model. The applicant concluded, for each of the cases, that it had met an acceptable FOS for overturning.

Based on its review, the staff found the applicant’s tabulated FOS results acceptable for each attributed case for the RXB overturning in DCA Part 2, Tier 2, Table 3.8.5-5, and determined that they meet DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.2 Control Building Stability

DCA Part 2, Tier 2, Section 3.8.5.6.2, “CRB Stability,” describes the CRB stability. Because the linear stability analyses gave unsatisfactory results for the CRB stability analyses, the applicant performed nonlinear evaluation for the uplift, sliding, and overturning stability analyses of the CRB. The staff’s Phase 2 regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML18324A551), reviewed calculation ER-F170-3932, Revision 2, for the CRB stability evaluation. Based on its review, the staff confirmed that the applicant developed the ANSYS models for the RXB and CRB to perform nonlinear analyses by uncoupling the soil domain from the buildings, which are connected with contact elements (CONTA178), to permit sliding under seismic conditions. The staff confirmed that the calculated results for the stability evaluations are small.

The staff concluded that the stability displacements of deeply embedded structures would be a minor consideration under the seismic conditions, and therefore, they meet DSRS Acceptance Criterion 3.8.5.II.4.E.

3.8.5.4.5.2.1 Control Building Uplift

DCA Part 2, Tier 2, Section 3.8.5.6.2.1, “CRB Uplift,” describes the CRB uplift. In DCA Part 2, Tier 2, Figures 3.8.5-49 and 3.8.5-50 show the vertical reaction force and displacement at location A, which is designated in DCA Part 2, Tier 2, Figure 3.8.5-48, “CRB Foundation Time History Location Designations.” The CRB is in an uplifted state at location A for an infinitesimal duration of time—just before the 10-second mark—resulting in zero reaction forces. The applicant determined that the maximum uplift at location A is less than 0.4 mm (1/64 in.) and concluded that the potential uplift is insignificant.

Based on its review, the staff agrees with the applicant’s conclusion of insignificant CRB uplifting and finds that the CRB is acceptable for a deeply embedded structure and therefore meets DSRS Acceptance Criterion 3.8.5.II.4.E.

<u>Dynamic Control Building Uplift Ratio</u>

DCA Part 2, Tier 2, Section 3.8.5.6.2.1.1, describes the dynamic CRB uplift ratio assessment. The linear SSI analysis methods are acceptable if the ground contact ratio is equal to or greater than 80 percent. The ground contact ratio can be calculated from the linear SSI analysis using the minimum basemat area that remains in compression with the soil. The applicant calculated the seismic total vertical base reactions by algebraic summation of all nodal vertical reactions of the nodes of the CRB basemat. In addition, the applicant summarized the maximum seismic vertical reactions for the cracked and uncracked concrete conditions in DCA Part 2, Tier 2,

Table 3.8.5-15. The applicant's base vertical reaction results for the uncracked condition are similar to those for the cracked concrete condition.

DCA Part 2, Tier 2, Table 3.8.5-14, "Seismic Vertical CRB Base Reactions and Dead Weight," gives the results of seismic vertical and dead weight and shows that the seismic reactions are much less than the total dead weight reaction of 203,190 kN (45,680 kips). Therefore, the applicant concluded that the net reactions are always in compression. DCA Part 2, Tier 2, Figures 3.8.5-77 through 3.8.5-82, show that the total basemat vertical reaction time histories are under compression for the cracked and uncracked concrete conditions. Therefore, the applicant concluded that the vertical reactions are in compression, and therefore, the basemat is 100 percent in contact. The staff reviewed DCA Part 2, Tier 2, Figures 3.8.5-77 through 3.8.5-82, and confirmed, in all cases, that the vertical reactions are in compression because of the seismic time histories of "Capitola and Lucerne" for Soil Types 7, 8, 9, and 11 and therefore meet DSRS Acceptance Criterion 3.7.2.II.4.

3.8.5.4.5.2.2 Control Building Sliding

DCA Part 2, Tier 2, Section 3.8.5.6.2.2, "CRB Sliding," describes the CRB sliding evaluation. DCA Part 2, Tier 2, Revision 2, Figure 3.8.5-51, "Lateral Relative Displacements (Sliding) at Location A," shows that the maximum relative sliding is 0.152 mm (0.006 in.) at location A, depicted in DCA Part 2, Tier 2, Figure 3.8.5-48. DCA Part 2, Tier 2, Table 3.8.5-12, summarizes the sliding and uplifting results, in which the magnitudes of these displacements are insignificant.

Based on its review, the staff finds that the applicant's results are acceptable to show insignificant sliding displacements as the CRB is a deeply embedded structure and, therefore, is consistent with DSRS Acceptance Criterion 3.8.5.II.4.E.

3.8.5.4.5.2.3 Control Building Overturning

DCA Part 2, Tier 2, Section 3.8.5.6.2.3, "CRB Overturning," describes the CRB overturning evaluation. DCA Part 2, Tier 2, Table 3.8.5-12, shows that the deeply embedded CRB experiences less than 2.5 mm (1/10 in.) of overturning horizontal displacement and less than 0.4 mm (1/64 in.) of total vertical uplift displacement. The applicant concluded that the magnitudes of these displacements are insignificant, and therefore, the potential for CRB overturning is insignificant.

Based on its review, the staff finds that the applicant's conclusion is acceptable to show that displacements are insignificant for any possibility of CRB overturning as the CRB is a deeply embedded structure; therefore, the conclusion is consistent with DSRS Acceptance Criterion 3.8.5.II.4.E.

3.8.5.4.5.3 Average Bearing Pressure

DCA Part 2, Tier 2, Section 3.8.5.6.3, "Average Bearing Pressure," describes the static and dynamic bearing pressures for the RXB and CRB basemats. DCA Part 2, Tier 2, Section 3.8.5.6.1, states that the bearing capacity of the soil should provide an FOS of 3.0 for the static bearing pressure and an FOS of 2.0 for dynamic bearing pressure.

The applicant defined the static bearing pressure by dividing the dead load of the building by the footprint area of the building. Based on its review, the staff finds the applicant's approach acceptable, since it is a generally accepted engineering practice of formulating to calculate the bearing pressure.

The dead weight of the RXB is 2,611.76 MN (587,147 kips), and the calculated footprint is 5,404.63 m^2 (58,175 ft^2). This results in an average static pressure of 0.484 MPa (10.1 ksf) and an FOS of 6.9 to the minimum soil bearing capacity of 3.59 MPa (75 ksf), specified in DCA Part 2, Tier 2, Table 2.0-1.

The dead weight of the CRB is 218,145 kN (49,041 kips), with a base area of 1,096 m^2 (11,800 ft^2). This results in an average static bearing pressure of 0.193 MPa (4.0 ksf) and an FOS of 19 to the minimum soil bearing pressure of 3.59 MPa (75 ksf) shown in DCA Part 2, Tier 2, Table 2.0-1.

The applicant also provided the dynamic bearing pressures of 0.220 MPa (4.6 ksf) and 0.110 MPa (2.3 ksf) for the RXB and CRB basemats, respectively, in DCA Part 2, Tier 2, Section 3.8.5.6.3. DCA Part 2, Tier 2, Figure 3.8.5-3, "Seismic Base Pressure Contours from SASSI2010 Analysis in the RXB Basemat (psi)," shows the seismic bearing pressure contours from SASSI2010.

By letters dated October 17, 2017 (ADAMS Accession No. ML17290B267), April 4, 2018 (ADAMS Accession No. ML18094B106), and August 23, 2018 (ADAMS Accession No. ML18235A280), the applicant described that the average dynamic pressures are obtained by the postprocessing approach, as described in DCA Part 2, Tier 2, Section 3.7.2.4.1. The applicant considered the tunnel basemat as a simple-supported, one-way slab spanning between the exterior and middle tunnel walls. The applicant used the maximum resultant bending moment, and conservatively added the twisting moment to the bending moments, for both X and Y directions. The applicant also described the CRB tunnel basemat assessments in DCA Part 2, Tier 2, Sections 3.8.5.4.1.3, "Analysis of Control Building Basemat," and 3.8.5.6.4, "Settlement," and provided the static and seismic bending moment (M_{xx} and M_{yy}) figures and tabulated settlement results from the analysis in DCA Part 2, Tier 2, Figures 3.8.5-4a, 3.8.5-5a, 3.8.5-6a, and 3.8.6-7a and Tables 3.8.5-17, "CRB Tunnel Foundation Corner Displacements," and 3.8.5-18, "CRB Tunnel Differential Settlement over 50 Feet and Tilt Angle." The applicant also provided Figures 3.8.5-2a and 3.8.5-3a to show the CRB basemat contour pressures for static and seismic loads in DCA Part 2, Tier 2. Furthermore, the staff, during its Phase 2 regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML18324A551), reviewed calculation ER-F170-3858, Revision 2, "Seismic Soil-Structure Interaction Analysis of NuScale CRB for Structural Responses." The applicant described the results from this calculation for the CRB toe pressure in DCA Part 2, Tier 2, Section 3.8.5.6.7. The applicant also provided the results of average toe pressures for the CRB in DCA Part 2, Tier 2, Table 3.8.5-15, which are less than the minimum soil-bearing pressure capacity of 3.59 MPa (75 ksf), as specified in DCA Part 2, Tier 2, Revision 2, Table 2.0-1 and Section 2.5.4.

Based on its review, the staff finds the applicant's approach acceptable because, during the audit, the staff reviewed the associated calculations and confirmed that the applicant had correctly performed calculations to determine average dynamic bearing pressures for the RXB and CRB. Additionally, the applicant described and provided figures showing the CRB basemat contour pressures for static and seismic loads. The applicant elaborated on the CRB tunnel basemat assessments for obtaining the moments for design and settlement values for the CRB tunnel. Based on the staff's review, the application meets DSRS Acceptance Criterion 3.8.5.II.4 and therefore is acceptable.

3.8.5.4.5.4 Settlement

In DCA Part 2, Tier 2, Table 3.8.5-7a, "Displacement at Bottoms of Foundations of Uncracked Triple Building Model," and Table 3.8.5-7b, "Displacement at Bottoms of Foundations of Cracked Triple Building Model," provide the displacement values for selected nodes in the RXB and CRC foundations. DCA Part 2, Tier 2, Figure 3.8.5-10, "Edge and Center Nodes at Bottom of Foundations Selected for Building Settlement Assessment," shows the locations of these nodes. In DCA Part 2, Tier 2, Tables 3.8.5-7a and 3.8.5-7b show that the values for vertical settlement, tilt, and differential displacements are small.

The applicant determined that the RXB settles approximately 4.45 cm (1.75 in.) on the west end and approximately 5.1 cm (2 in.) on the east end, and the differential settlement of 6.4 mm (0.25 in.) is less than 2.5 cm (1 in.), as stated in DCA Part 2, Tier 2, Section 3.8.5.6.1. The RXB has negligible N-S tilt.

The applicant determined that the CRB settles approximately 4.45 cm (1.75 in.) on the west end and approximately 2.5 cm (1.0 in.) on the east end, and the differential settlement of 1.9 cm (0.75 in.) is less than 2.5 cm (1.0 in.), as stated in DCA Part 2, Tier 2, Section 3.8.5.6.2. The CRB tilts toward the RXB and has negligible N-S tilt. The applicant determined that the differential settlement between the two buildings is 6.4 mm (1/4 in.). DCA Part 2, Tier 2, Revision 2, Table 3.8.5-17, tabulates the displacements at the four corners of the CRB tunnel foundation for the cracked concrete condition. The applicant determined that the maximum settlement is approximately 5.1 cm (2.0 in.). DCA Part 2, Tier 2, Table 3.8.5-18, provides the rotation of the tunnel foundation as 0.0361 degrees. The tunnel foundation has negligible differential settlement in the N-S direction, and the more than 15-m (50-ft)-long differential settlement in the E-W direction is 0.91 cm (0.36 in.).

Based on its review, the staff finds that the applicant's tabulated results of settlements are acceptable in DCA Part 2, Tier 2, Tables 3.8.5-7a, and 3.8.5-7b, for the RXB and CRB foundations, and the maximum settlement tilt and tilt angle in DCA Part 2, Tier 2, Tables 3.8.5-17 and 3.8.5-18, are acceptable for the CRB tunnel foundation.

By letters dated December 20, 2018 (ADAMS Accession No. ML18354B330), and March 28, 2019 (ADAMS Accession No. ML19087A330), the applicant stated that it intended to use "headed reinforcing bars" that deviated from the reinforcing bars with standard hooks in the foundation design. The applicant also described that the headed reinforcing bars have equivalent or superior performance compared to the reinforcing bars with standard hooks and offer less congestion, which facilitates concrete consolidation and faster installation times, reducing placement costs. Although headed bars are not addressed in the provisions of ACI 349-06, the staff determined that the headed reinforcement bars were first introduced in the 2008 Edition of ACI 318, "Building Code Requirements for Structural Concrete and Commentary," followed by the 2013 Edition of ACI 349. Based on the review, the staff finds that the applicant's supplemental response is acceptable since the applicant provided an appropriate level of information describing the application of headed reinforcing bars in DCA Part 2, Tier 2, Section 3B1.1.3, "Wall and Slab Design Approach," and referred to ACI 318-08 in DCA Part 2, Tier 2, Section 3B4 "References." Therefore, the staff considers determined settlements to be acceptable since they are consistent with DSRS Acceptance Criterion 3.8.5.II.4.H.

3.8.5.4.5.5 Thermal Loads

The staff reviewed the temperature gradient across the CRB foundation 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 DCA Part 2, Tier 2, Section 3.8.5.6.5, "Thermal Loads," the applicant stated that during normal operation, a linear temperature gradient across the RXB foundation may develop, but the applicant did not perform a thermal analysis. The applicant considered the thermal loads to be minor loads and self-relieving because of concrete cracking and moment distribution.

Based on its review, the staff finds the applicant's description of thermal loads acceptable, since they are minor loads and self-relieving because of concrete cracking and moment distribution. 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 DCA Part 2, Tier 2, Section 3.8.5.6.6, "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 much 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 much 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.

At its Phase 2 regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML18324A551), the staff reviewed calculation EC-F012-3683, Revision 2, to determine the justifications of structural integrity of the RXB pool liner. The stainless steel RXB liner was designed to the requirements of ASME BPV Code, Section III, Division 2. The applicant explained that the stainless steel RXB liner is used as a permanent form during the construction, and the load combinations also consider loading from concrete pour. However, this issue was resolved and closed per the discussions provided in Section 3.8.5.4.5.8 of this SER.

3.8.5.4.5.7 Basemat Soil Pressures along Basemat Edges (Toe Pressure)

DCA Part 2, Tier 2, Section 3.8.5.6.7, describes the toe pressures. The static dead weight reaction at an edge node is added to the seismic reaction of the node to calculate the total reaction. The applicant calculated the bearing pressure by dividing the total reaction by the tributary area of the node. In addition, the bearing capacity of the soil should provide an FOS of 3.0 for the static bearing pressure and an FOS of 2.0 for the dynamic bearing pressure in DCA Part 2, Tier 2, Section 3.8.5.6.1.

In DCA Part 2, Tier 2, Tables 3.8.5-13 and 3.8.5-15 provide the average toe pressures for the RXB and CRB, respectively. The values shown in these tables result in an FOS greater than 2.0, where the minimum soil bearing pressure capacity is 3.59 MPa (75 ksf), as specified in DCA Part 2, Tier 2, Table 2.0-1.

The staff confirms that in DCA Part 2, Tier 2, Tables 3.8.5-13 and 3.8.5-15 include the applicant's results of average toe pressures for the RXB and CRB and show that they all result in an FOS greater than 2.0.

The applicant performed analyses to determine the edge bearing pressures (or toe pressures) along the edges of the basemats of the RXB and CRB. DCA Part 2, Tier 2, Section 3.8.5.6.7, shows the applicant's analyses to determine the edge bearing pressures (or toe pressures) along the edges of the seismic Category I structure basemats (RXB and CRB). DCA Part 2, Tier 2, Section 3.8.5.5.5, considers further reducing the soil stiffness values by 50 percent to counter the effects of settlements. In DCA Part 2, Tier 2, Tables 3.8.5-7a and 3.8.5-7b tabulate the settlement values of uncracked and cracked, respectively, using the soft-soil profile (soil Type 11) properties in the triple building foundation model. DCA Part 2, Tier 2, Figure 3.8.5-10, also shows the locations of critical edge and center nodes at the bottom of the foundations. Furthermore, the staff in its Phase 2 regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML18324A551), reviewed calculation ER-F010-4135, Revision 3, and confirmed the settlement results for uncracked and cracked concrete conditions using the triple building model for soft-soil profile (Soil Type 11) and determined that the applicant conservatively reduced the soft-soil profile by 50 percent to determine the settlement values of the RXB and CRB basemats.

Based on its review, the staff finds the applicant's approach acceptable because the applicant analytically determined settlement values from uncracked and cracked concrete conditions, respectively, using the triple building model for the conservative soft-soil profile (Soil Type 11), and showed the locations of edge and center nodes. The staff also confirmed that the tabulated settlement values are below the maximum total value of 10.2 cm (4.0 in.), as provided in DCA Part 2, Tier 2, Revision 2, Table 2.0-1. The applicant's response also meets DSRS Acceptance Criterion 3.8.5.II.4 and therefore is acceptable.

3.8.5.4.5.8 Leak Detection

The staff reviewed the design details to prevent and monitor potential leakage from the pool and potential leakage into the RXB from ground water outside the RXB 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.4.O.

DCA Part 2, Tier 2, Section 3.8.5.6.8, "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.

DCA Part 2, Tier 2, Section 3.8.5.6.8, states, "A leak chase system is provided in the RXB basemat to detect any leakage from the reactor pool." However, the applicant did not describe the leak chase system in detail. In its response dated October 17, 2017 (ADAMS Accession No. ML17290B267), the applicant referred to DCA Part 2, Tier 2, Section 9.1.3.2.5, which describes the pool leakage detection system. As described in that section, the pool leakage detection system consists of embedded in-concrete floor leakage and perimeter leakage channels, channel drainage lines, leak collection headers, leakage rate measuring lines, and valves. The channels collect leakage from the pool liner plates and direct it to a sump or to collection header piping that leads to a sump in the radioactive waste drain system. The leakage collected in the radioactive waste drain system sumps is routed to the liquid radioactive waste system for further processing. DCA Part 2, Tier 2, Section 3.2, classifies the pool leakage detection system, radioactive waste drain system, and liquid radioactive waste system as not safety related and not risk significant (B2).

The staff conducted a Phase 2 regulatory audit from June 14, 2018 through July 12, 2018, to review the applicant's documents and to obtain additional information on the responses to numerous related RAIs to ensure the prevention of possible effects of leaking borated UHS pool water through the liner seam-welds into the safety-related reinforced concrete structures (walls and foundation). The staff communicated its review scope in the audit plan dated June 12, 2018 (ADAMS Accession No. ML18158A164). The audit process allowed the staff to access supporting documentation that it identified as potentially significant to the review, such as figures, system diagrams, and nondocketed information, in the applicant's electronic reading room, and hold discussions with the NuScale technical personnel. The staff's review of the documents led to a concern about the ability of the current leak chase channel arrangement to prevent the leaking of UHS pool water through the liner seam-welds into the safety-related reinforced concrete structures. The staff provided feedback and requested supplemental information through the audit report (ADAMS Accession No. ML19098A162).

Furthermore, the staff at its Phase 2 regulatory audit from December 3 to 7, 2018 (ADAMS Accession No. ML18324A551) reviewed calculation EC-F012-3683, Revision 2, to determine the justifications of structural integrity of the RXB pool liner. Based on the review, the applicant correctly described that the stainless steel RXB pool liner was designed to the requirements of ASME BPV Code, Section III, Division 2. The applicant also described that the stainless steel RXB pool liner is used as a permanent form during the construction and, appropriately, the load combinations considered the loading from concrete pour. The applicant included COL Item 3.8-6, in DCA Part 2, Tier 2, Section 3.8.4.1.7, to direct the COL applicant to verify that the construction load of concrete pour pressure does not exceed 28.7 kPa (600 psf), to the pool liner plate and its support structure per ACI 347, "Guide to Formwork for Concrete."

Based on audits performed and the review of DCD Part 2, Tier 2, the staff determined the applicant's approach is acceptable because the applicant analytically qualified the integrity of the stainless steel RXB pool liner in accordance with ASME BPV Code, Section III, Division 2, and considered concrete pour pressure of 28.7 kPa (600 psf) as the construction load in the load combinations as well as adding COL Item 3.8-6 to account for the construction load of concrete pour pressure. Thus, the applicant meets DSRS Acceptance Criterion 3.8.4.II.1.

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.

The applicant described the materials, quality control and special construction techniques in DCA Part 2, Tier 2, Section 3.8.5.7, "Materials, Quality Control, and Special Construction Techniques." DCA Part 2, Tier 2, Section 3.8.4.6, 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 DCA Part 2, Tier 2, Section 3.8.4.6, with regard to their application to the RXB and CRB foundations. DCA Part 2, Tier 2, Section 3.8.4.6.1, describes the principal construction materials for seismic Category I structures as concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. DCA Part 2, Tier 2, Table 3.8.4-10, provides the material properties of concrete and steel. DCA Part 2, Tier 2, Section 3.8.4.6.1.1, states that the minimum compressive strength of concrete at below grade is 34 MPa (5,000 psi). DCA Part 2, Tier 2, Section 3.8.4.6.1.1, also states that the concrete ingredients are cement, aggregates, admixtures, and water. In DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.8.5.6.6, describes the foundations of seismic Category I structures as poured-in-place reinforced concrete structures, with concrete and steel reinforcing bars as the primary materials used in construction.

DCA Part 2, Tier 2, 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 material, 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 DCA Part 2, Tier 2, Section 3.8.5.6, to be acceptable.

3.8.5.4.7 Testing and Inservice Surveillance 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.

DCA Part 2, Tier 2, Section 3.8.4.7, describes the testing and ISI requirements for the RXB and CRB foundations. 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. The applicant proposed COL Item 3.8-1, which states that a COL applicant that references the NuScale Power Plant DC will describe the site-specific program(s) 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 DCA Part 2, Tier 2, Section 3.8.4.7, to verify performance of the testing, monitoring, and maintenance of structures in accordance with 10 CFR 50.65 and RG 1.160 for other seismic Category I structures. The adequacy of testing, monitoring, and maintenance of the foundations for other seismic Category I structures is in accordance with 10 CFR 50.65 and RG 1.160, as addressed in DSRS Section 3.8.4.

The as-built RXB and CRB design commitments include the following ITAAC in DCA Part 2, Tier 1, Section 3.11, "Reactor Building," and Section 3.13, "Control Building," respectively: (1) internal flooding barriers to provide confinement so that the impact from internal flooding is contained within the RXB and CRB flooding area of origin and (2) protection against external flooding in order to prevent the flooding of safety-related SSCs within the RXB and CRB, as listed in DCA Part 2, Tier 1, Table 3.11-2 and Table 3.13-1.

3.8.5.5 Combined License Information Items

The applicant added COL Item 3.8-3 in DCA Part 2, Tier 2, Section 3.8.5.4.1.4 and Table 1.8-2, stating that a COL applicant that references the NuScale Power Plant DC will identify local stiff and soft spots in the foundation soil and address these in the design of foundations, as necessary.

Table 3.8.5-1: NuScale COL Information Item for Section 3.8.5

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.8-3	COL applicant that references the NuScale Power Plant design certification will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.	3.8.5.4.1.4

3.8.5.6 Conclusion

The staff concludes that the design of NuScale's RXB and CRB foundations is acceptable and meets the 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 assumptions and procedures used for the inclusion of transients in the design and fatigue evaluations of ASME BPV Code Class 1 and core support components, the computer programs used in the design and analyses of seismic Category I components and their supports, and experimental and inelastic analytical techniques.

The staff reviewed DCA Part 2, in accordance with SRP Section 3.9.1, Revision 4, "Special Topics for Mechanical Components," issued December 2016. The applicant's DCA Part 2 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 both those designated as ASME BPV Code, Section III, Class 1, 2, 3, or core supports, and those not covered by the ASME BPV Code. Alternatives to the guidance may be proposed if adequate justification is provided. The staff reviewed DCA Part 2 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

- descriptions of any experimental stress analysis programs to be used in lieu of theoretical stress analyses

- descriptions of the analysis methods to be used if the applicant elects to use elastic-plastic stress analysis methods in the design of any components

- 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

DCA Part 2, Tier 1: There is no Tier 1 information for this area of review.

DCA Part 2, Tier 2: 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)

DCA Part 2, Tier 2, Section 3.9.1.1, does not cover the seismic loading and other mechanical loading on each component. DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.1.3, "Experimental Stress Analysis," states that experimental stress analysis is not used for the NuScale design.

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.9.1, "Special Topics for Mechanical Components," in accordance with SRP Section 3.9.1. The staff also reviewed portions of DCA Part 2, Tier 2, Section 5.1, "Summary Description," and Section 5.2, "Integrity of Reactor

Coolant Boundary." The staff reviewed DCA Part 2, Tier 2, Section 3.9.3, in regard to seismic loading and analytical methods used to evaluate ASME BPV Code, Section III, Service Level D, limits. The staff also reviewed DCA Part 2, Tier 1, Section 2.1; Section 2.2, "Chemical and Volume Control System"; and Section 2.3, "Containment Evacuation System," in regard to applicable component and piping design commitments and related ITAAC. The staff's evaluation focused on determining whether there is adequate assurance of a mechanical component performing its safety-related function under all postulated service conditions, including normal operation and transients, in accordance with SRP Section 3.9.1, and seismic events, as defined in SRP Section 3.7.2, "Seismic System Analysis." The staff also reviewed DCA Part 2, Tier 2, Section 3.8.4, "Other Seismic Category I Structures," and Section 3.12.4, "Piping Modeling Technique," as they relate to the design load combinations in Tables 3.9-3 through 3.9-14. In particular, the OBE load is required to be part of the fatigue CUF calculation. If the OBE magnitude is smaller than one-third of the SSE, in accordance with 10 CFR Part 50, Appendix S, the SSCs of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public that must remain functional and within applicable stress, strain, and deformation limits can be satisfied without the applicant performing explicit response or design analysis. The OBE is defined to be one-third of the SSE and therefore is eliminated from explicit response or design analyses, in accordance with 10 CFR Part 50, Appendix S, Section IV. However, the fatigue analysis of SSCs accounts for the effects of the OBEs.

DSRS Section 3.7.3 states that the fatigue evaluation should assume one SSE and five OBE events. DCA Part 2, Tier 2, Section 3.12.5.5, "Fatigue Evaluation of ASME BPV Code Class 1 Piping," states that a minimum of one SSE and five OBE events were included in the fatigue analysis of ASME Code Class 1 piping components. In DCA Part 2, Tier 2, Section 3.12.5.5, the applicant stated that this loading is equivalent to an alternate load of two SSEs with 10 maximum stress cycles each, for a total of 20 full cycles; or 312 cycles with an amplitude of one-third of the SSE in accordance with DCA Part 2, Tier 2, Section 3.7.3.2 and Section 3.12.5.5, when derived in accordance with IEEE 344-2004, Annex D.1. The staff concluded that the applicant's position on fatigue consideration is acceptable because it is consistent with DSRS Section 3.7.3. 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.3, gives load combinations and their acceptance criteria for mechanical components and associated supports, and DCA Part 2, Tier 2, 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 severe or 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 or occurrences as listed in DCA Part 2, Tier 2, Table 3.9-1, and magnitude of temperature and pressure changes. DCA Part 2, Tier 2, Table 3.9-1, contains the description of events and occurrence cycles. The applicant also explained that the OBE is a Service Level B loading and is included in the fatigue analyses as discussed in DCA Part 2, Tier 2, Section 3.9.1.

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. The design basis assumes 90 cycles of turbine trip without bypass and 180 cycles of turbine trip with bypass.

The number of cycles selected for the turbine trip without bypass and turbine trip with bypass transients are based on the predicted NuScale probabilistic risk assessment initiating event frequencies, nuclear operating experience, and comparisons to recent DCAs. As DCA Part 2, Tier 2, Table 19.1-8, "Level 1 Internal Probabilistic Risk Assessment Initiating Events," shows, the mean initiating event frequency for a general reactor trip is 1.3 per module critical year, which equates to 78 events over the 60-year design life. This includes the event frequency for a turbine trip as a portion of this general reactor trip frequency. Therefore, considering the number of total reactor trips from full power for NuScale to be 125 times in DCA Part 2, Tier 2, Table 3.9-1, the staff concludes that 90 cycles is reasonable for the turbine trip without bypass. The applicant discussed the basis for selecting 180 cycles for the turbine trip with bypass during a public teleconference on December 12, 2017. As a result, the applicant provided supplemental information in a letter dated December 27, 2017 (ADAMS Accession No. ML17361A301), indicating that, for NuScale bounding design purposes, the applicant assumed 270 occurrences of a turbine trip. As such, the applicant selected 180 cycles for the turbine trip with bypass. The applicant also indicated that it considered dynamic fluid loads, such as those that could be generated during a rapid turbine stop valve closure, in the piping load combinations, as discussed in DCA Part 2, Tier 2, Section 3.12.5.3, "Loadings and Load Combinations," and in the component and component support load combinations, as discussed in DCA Part 2, Tier 2, Section 3.9.3.1.1, "Loads for Components, Component Supports, and Core Support Structures." Based on its review, the staff finds that the applicant specified a conservative number of cycles for these transients.

DCA Part 2, Tier 2, Section 3.9.1.1.1, "Service Level A Conditions," discusses power ascent and descent between 0 and 15 percent of full power. Above 15 percent of full power, the control systems are placed in automatic mode. DCA Part 2, Tier 2, Section 3.9.1.1.1, specifies that the power ascent from hot shutdown and power descent to hot shutdown rate is limited to 0.5 percent of full power per minute. These two transients are identified as Transient 3 and Transient 4, respectively, in DCA Part 2, Tier 2, Section 3.9.1.1. DCA Part 2, Tier 2, Table 3.9-1, describes the numbers of transient occurrences.

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 DCA Part 2 and noted deviations from the previously accepted practice. The NuScale design includes 700 occurrences of power ascension and 300 occurrences of power descent. The applicant indicated that considering the rate of coolant temperature changes during these transients, the power ascent causes larger temperature gradients and stresses in the affected NPM regions than the power descent. The applicant assumed 700 power ascent and 300 power descent events because it is more realistic to have a higher number of power ascents than descents, as a result of plant trips. Although the selected number of power descent events is less than the 500 used by a previously approved DC applicant, in this case, the applicant justified the difference by using an additional 200 power ascent events in the analyses, which generate higher stresses in NPM components through faster temperature changes. This approach is acceptable because, for the NuScale design, the ascent events result in larger temperature gradients and therefore result in a conservative analysis of the affected NPM regions. The staff concludes that the applicant's use of higher cycles for the higher stress conditions combined with a lower number of power descents is conservative for the fatigue design of Class 1 components and is therefore acceptable.

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). Inadvertent actuation of the PZR spray is one such operational occurrence. DCA Part 2, Tier 2, Table 3.9-1, assumes 15 cycles for inadvertent PZR spray instead of 30 cycles as is typical for a standard pressurized-water reactor (PWR) plant for this transient.

An inadvertent PZR spray transient occurs when the PZR spray control valve fails to control and changes to an open position. This valve in the NuScale CVCS design is an air-operated valve (AOV). The applicant stated that according to the 2015 results of the industry average parameter estimates for component reliability, which are updates to those originally provided in NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," issued February 2007 (ADAMS Accession No. ML070650650), the mean frequency for "Air Operated Valve Fails To Control" is 2.28 x 10^{-7} per hour, while the baseline mean frequency in NUREG/CR-6928 for the AOV failed to control is 3.0 x 10^{-6} per hour. Over a 60-year design life, this equates to 0.12 cycles (or 1.58 baseline cycles). The staff noted that because of improved technology, the mean frequency of failure is reduced from 1.58 to 0.12 cycles per plant life. The staff finds 15 cycles over 60 years is sufficiently bounding for the design analysis.

The staff reviewed the remaining transients described in DCA Part 2, Tier 2, Section 3.9.1.1 and Table 3.9-1, and finds the applicant's identification of the design transients and the number of transients is appropriate for the design life. Therefore, the staff finds the applicant's characterization of design transients acceptable.

3.9.1.4.2 Computer Programs Used in Analyses

The staff reviewed DCA Part 2, Tier 2, 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. In addition to meeting the requirements in 10 CFR Part 50, Appendix B, the applicant stated that for computer programs used for design control and verification, NuScale commits to complying with ASME NQA-1-2008, "Quality Assurance Requirements for Nuclear Facility Applications," and ASME NQA-1a-2009 Addenda, Requirement 3, Sections 100–900, and the standards for computer software in ASME NQA-1-2008 and ASME NQA-1a-2009 Addenda, Part II, Subpart 2.7, and Subpart 2.14, for QA requirements for commercial-grade items and services. The applicant also stated that delegated responsibilities may be performed under an approved supplier's or principal contractor's QAP, in which case the supplier is responsible for the control of computer programs used. From March 20, 2018, through

April 27, 2018, the staff conducted a regulatory audit of the computer codes in support of its reviews of DCA Part 2, Tier 2, Section 3.9.1, to ensure that the computer programs used for the design of seismic Category I structures and components meet the requirements of GDC 1 and 10 CFR Part 50, Appendix B, with respect to the development, procurement, testing, and maintenance of computer programs in accordance with the QAP described in DCA Part 2, Tier 2, Chapter 17, for a 60-year NuScale plant life.

DCA Part 2, Tier 2, 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 plant contractors used RspMatch2009, SAP2000, SASSI2010, SHAKE2000, EMDAC, and Simulink. The review procedures in SRP Section 3.9.1, Section III.2.B, state the following:

> The submitted computer test problem solutions recommended in subsection II.2.C of this SRP section are reviewed and compared to the test solutions. Satisfactory agreement of computer and test solutions, usually within +/- 5 percent error band, verifies the quality and adequacy of the computer programs for the functions for which they were designed.

With respect to NRELAP5, DCA Part 2, Tier 2, Section 3.9.1.2, indicates that the development, use, verification, validation, and code limitations for this program are discussed in Section 15.0.2 for application to transient and accident analyses, and TR-1016-51669, Revision 1, “NuScale Power Module Short-Term Transient Analysis,” issued July 2019 (ADAMS Accession Nos. ML19211D414 (proprietary) and ML19211D411 (nonproprietary) (DCA Reference 3.9-8) for application to short-term transient dynamic mechanical loads, such as pipe breaks and valve actuations. NuScale technical report TR-1016-51669 is incorporated by reference as part of the NuScale Power Plant DCA in Part 2, Tier 2, Section 1.6 and Table 1.6-2.

COL Item 3.9-14 specifies that a COL applicant that references the NuScale Power Plant DC will develop an evaluation methodology for the analysis of secondary-side instabilities in the SG design. This methodology will address the identification of potential density wave oscillations (DWOs) in the SG tubes, and qualification of the applicable portions of the RCS integral RPV and SG given the occurrence of DWOs, including the effects of reverse fluid flows within the tubes. In this safety evaluation, the NRC does not finalize the DC review of the evaluation methodology for the analysis of secondary-side instabilities in the SG design for the NuScale nuclear power plant. Because the NuScale application neither describes nor analyzes the details of the DWO transients in the SG tubes, or qualification of the SGs given the occurrence of DWOs, including the effects of reverse fluid flows within the tubes and SGIFRs for the NuScale nuclear power plant, the NRC staff does not have a basis for making findings regarding the potential for secondary-side instabilities in the SGs or the adequacy of an evaluation methodology to address this condition. The NRC staff determined that COL Item 3.9-14 proposed by the applicant was not sufficient for the NRC staff to conclude its review in this area. Therefore, the NRC staff recommends that the Commission include language in the proposed rule stating that the NRC is not making a finding with regard to the secondary-side instabilities in the SGs design or the associated evaluation methodology. Specifically, 10 CFR Part 52, Appendix G for the Design Certification for the NuScale SMR, Section VI, “Issue Resolution,” will state that the evaluation of secondary-side instabilities in the SG, particularly with regard to the potential for DWOs and the associated evaluation methodology, are not considered resolved within the meaning of § 52.63(a)(5) by the design certification. Additionally, and Section IV, “Additional Requirements and Restrictions,” will require the COL applicant to complete the analysis of secondary-side instabilities in the SGs, taking into account the potential DWOs in the SG tubes and qualification of the SGs given the occurrence of DWOs, including the effects of reverse fluid flows within the tubes and SGIFRs, and provide an associated evaluation methodology.

As indicated in the audit plan (ADAMS Accession No. ML18074A079) and the audit report (ADAMS Accession No. ML18324A562), the staff conducted the audit for verification and validation (V&V) of computer programs, including the test problem benchmarking methods, solutions, and summary of the results used for computer program qualification, and concluded that the V&V requirements were met for the computer codes used by the applicant. The staff performed a vendor inspection from October 1–5, 2018 (ADAMS Accession No. ML18318A261), of the V&V of the computer codes used by contractors and concluded that the criteria were met.

Appendix B to 10 CFR Part 50 requires provisions to assure that design documents include and specify appropriate standards, including design methods and computer programs, for the design and analysis of seismic Category I ASME BPV Code Class 1, 2, 3, and core support structures and non-ASME BPV Code structures.

The staff conducted an audit for NuScale computer codes for ASME BPV Code Class 1, 2, and 3 components from March 20, 2018 through April 27, 2018 (ADAMS Accession No. ML18074A079), to confirm that the computer programs used for the NuScale design as listed in DCA Part 2, Tier 2, Section 3.9.1.2, comply with 10 CFR Part 50, Appendix B, and ASME NQA-1 and to evaluate the computer programs used to perform the fatigue analysis. For calculating the cumulative usage factor (CUF) and the factor for environmental fatigue (F_{en}), the applicant did not have a computer code. The fatigue analysis is evaluated in this SER in Sections 3.9.3 and 3.12. As a result of the audit, as shown in the audit report (ADAMS Accession No. ML18324A562), the staff concluded that the computer codes, as listed in DCA Part 2, Tier 2, Section 3.9.1.2, for the NuScale ASME BPV Code Class 1, 2, and 3 components, comply with 10 CFR Part 50, Appendix B, and ASME NQA-1. As noted above, this conclusion does not apply to the evaluation methodology for the analysis of secondary-side instabilities in the SG design. A potential future COL applicant will be responsible for providing information regarding the evaluation methodology used to analyze secondary-side instabilities for the NRC staff's review and approval.

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 DCA Part 2, Tier 2, 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 related to seismic instrumentation from DCA Part 2, Tier 2, Table 1.8-2.

Table 3.9.1-1: NuScale COL Information Items for Section 3.9.1

Item No.	Description	DCA Part 2 Tier 2, Section
COL Item 3.9-12	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. 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
COL Item 3.9-14	A COL applicant that references the NuScale Power Plant design certification will develop an evaluation methodology for the analysis of secondary-side instabilities in the steam generator design. This methodology will address the identification of potential density wave oscillations in the steam generator tubes, and qualification of the applicable portions of the reactor coolant system integral reactor pressure vessel and steam generator given the occurrence of density wave oscillations, including the effects of reverse fluid flows within the tubes.	3.9

3.9.1.6 Conclusion

On the basis of the evaluations and limitations documented in SER Sections 3.9.1.1–3.9.1.4, 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.

3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components

3.9.2.1 Introduction

This section of the SER reviews the analytical methodologies, testing procedures, and dynamic analyses that the applicant used to ensure the structural and functional integrity of the piping systems, mechanical equipment, reactor vessel internals (RVIs), and their supports under vibratory loadings, including those caused by fluid flow and postulated seismic events.

This section addresses six main areas of review:

(1) piping vibration, thermal expansion, and dynamic effects testing

(2) seismic analysis and qualification of seismic Category I mechanical equipment

(3) dynamic response analysis for RVIs under operational flow transients and steady-state conditions

(4) preoperational flow-induced vibration (FIV) testing of RVIs

(5) dynamic system analysis of the RVIs under faulted conditions

(6) correlations of RVI vibration tests with the analytical results

3.9.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, provides the information associated with this section on designing the ASME BPV Code Class core support components in the NuScale Power Module (NPM) to the provisions of ASME BPV Code, Section III, Subsection NG, "Core Support Structures."

In DCA Part 2, Tier 1, Section 2.1.1, the applicant made the following commitment:

> TF-3 is the test facility designed to study fluid elastic instability, vortex shedding, and turbulence due to primary side flow in helical steam generator tubes. Testing consists of modal testing in air and in water, and primary side flow testing with extensive instrumentation to detect vibration.
>
> Prototypes of the SG assembly will undergo TF-3 testing and meet the acceptance criteria in accordance with the Initial Test Program Steam Generator Flow-Induced Vibration Test. The results of the testing will be reviewed and approved in accordance with the NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report prior to loading fuel in the first ever NPM. This one-time testing satisfies TF-3 testing requirements for subsequent NPMs built in accordance with the approved design.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," describes the dynamic testing and analysis of systems, components, and equipment. In DCA Part 2, Tier 2, Section 3.9.2.1, "Piping Vibration, Thermal Expansion, and Dynamic Effects," and Section 14.2, "Initial Plant Test Program," describe the vibration and thermal expansion testing of piping.

DCA Part 2, Tier 2, Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals under Service Level D Conditions," and DCA Part 2, Tier 2, Appendix 3A, "Dynamic Structural Analysis of the NuScale Power Module," describe the dynamic analysis of the NPM. DCA Part 2, Tier 2, Appendix 3A, references Technical Report TR-0916-51502, Revision 2 (ADAMS Accession Nos. ML19094A021 (proprietary) and ML19093B850 (nonproprietary)), for additional details. A system-level NPM seismic analysis was performed first on a 3-D 360-degree NPM ANSYS model that consists of six NPM submodels representing the containment, reactor vessel, lower RVI, upper RVI, control rod drive mechanism (CRDM), and top support structure (TSS). The six submodels were connected by coupling nodes, constraint equations, contact elements, and fluid coupling to form the 3-D NPM model. A soil-structure interaction (SSI) analysis was performed using a simplified NPM model with a peak ground acceleration of 0.5g in the horizontal direction and 0.4g in the vertical direction. The calculated acceleration time histories from the SSI analysis were used as boundary conditions at the pool floor and walls, as well as the NPM supports for seismic analysis of the NPM entire pool model that contains the 3-D NPM and entire pool water. The 12 analysis cases considered cracked and uncracked concrete of the reactor building (RXB), variation of NPM stiffness, and location of the NPM. From the results of the 12 cases, maximum seismic loads (i.e., displacements, in-structure response spectra (ISRS), forces and moments at NPM component interfaces, and cross sections) were generated for the NPM component-level stress analysis in the Service Level D Condition. The load combination in the Service Level D Condition component stress analysis involved the safe-shutdown earthquake (SSE) load, blowdown load of the main steam pipe break (MSPB) and feedwater pipe break (FWPB), and design-basis pipe break (DBPB) load. The DBPB load included the load from the spurious actuation of the reactor vent valves (RVVs) and reactor recirculation valves (RRVs), as well as from a chemical and volume control system (CVCS) pipe break. The square root of the sum of the squares (SRSS) method was used to combine the loads resulting from an SSE and the highest load of the MSPB, FWPB, or DBPB. TR-1016-51669, Revision 1 (ADAMS Accession Nos. ML19211D414 (proprietary) and ML19211D411 (nonproprietary)), documents the blowdown analysis of a DBPB, MSPB, and FWPB. The thermal-hydraulic code NRELAP5 and the ANSYS code calculated the short-term transient structural loads. NRELAP5 was used to generate thermal-hydraulic boundary condition inputs for the ANSYS structural dynamic analysis model that calculated the short-term transient structural loads within the NPM. An ANSYS model was developed for (1) calculating the asymmetric cavity pressurization load between the containment vessel (CNV) and the reactor pressure vessel (RPV) and (2) calculating the blowdown load inside the RPV. For DBPB, six cases were analyzed. Maximum forces and moments at NPM component interfaces and cross sections were generated for each of the six analysis cases. The bounding values of the maximum forces and moments of the six analysis cases were used to combine with the SSE load for the NPM component Service Level D stress evaluation.

The applicant has submitted TR-0716-50439, Revision 2, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report” (CVAP Technical Report), issued July 2019 (ADAMS Accession Nos. ML19212A777 (proprietary) and ML19212A776 (nonproprietary)), which is referenced in DCA Part 2, Tier 2, 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 Unit Operation.” In addition, the applicant has submitted licensing document TR-0918-60894, Revision 1, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan (MIP) Technical Report,” issued August 2019 (ADAMS Accession Nos. ML19214A250, ML19214A251, and ML19214A253 (proprietary) and ML19214A248 (nonproprietary)). DCA Part 2, Tier 2, Section 14.2, also describes the steam generator (SG) prototype testing and the NPM initial startup vibration testing.

The CVAP describes the screening procedures and provides the results of the FIV analyses of (1) RVIs and structures, (2) helical coil SG (HCSG) components, and (3) primary and secondary reactor coolant system (RCS) piping, up to the NPM disconnect flange.

Components with small margins of safety against FIV effects were identified for validation testing. In particular, an HCSG mockup, called the Società Informazioni Esperienze Termoidrauliche (SIET) test facility-3 (TF-3), will be built and tested in accordance with TR-0918-60894, Revision 1. In addition, a final validation test for leakage flow instability (LFI) of the HCSG inlet flow restrictor (IFR) under forward secondary flow conditions is planned. The initial startup testing for FIV effects on the prototype NPM will be accomplished through a set of sensors to detect any unexpectedly strong FIV of the RVIs, CNTS steam piping, and HCSG.

ITAAC: DCA Part 2, Tier 1, Table 2.8-2, “Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria,” ITAAC 1, requires that the seismic Category I equipment listed in DCA Part 2, Tier 1, Table 2.8-1, “Module Specific Mechanical and Electrical/I&C Equipment,” including its associated supports and anchorages, withstands design-basis seismic loads without the loss of its safety-related function(s) during and after an SSE. DCA Part 2, Tier 1, Table 2.8-1, lists seismic Category I RVI components.

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

Technical Reports:

- TR-0716-50439, Revision 2, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report”

- TR-0916-51502, Revision 2, “NuScale Power Module Seismic Analysis”

- TR-0918-60894, Revision 1, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report”

- TR-1016-51669, Revision 1, “NuScale Power Module Short-Term Transient Analysis”

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 acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.20, Revision 3, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” issued March 2007, 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

DCA Part 2, Tier 2, 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 identified in DCA Part 2, Tier 2, Table 3.2-1, (2) other high-energy piping systems inside seismic Category 1 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.

DCA Part 2, Tier 2, Table 3.2-1, lists systems and their QG that corresponds to the ASME BPV Code classification. The test program, as described in DCA Part 2, Tier 2, Section 14.2, verifies that the Class 1, Class 2, Class 3, and other high-energy and seismic Category 1 piping systems meet functional design requirements and that piping vibrations and thermal expansions are within acceptable levels and will withstand dynamic effects resulting from operating transients.

DCA Part 2, Tier 2, Section 3.9.2.1, states that DCA Part 2, Tier 2, Section 14.2, describes the initial test program (ITP). The vibration, thermal expansion, and dynamic effect elements of this test program are performed during Phase I preoperational testing and Phase II initial startup testing. The Phase I preoperational tests are performed to demonstrate that the piping system components meet functional design requirements and that piping vibrations and thermal expansions and contractions are bounded by the analyses. If the design-basis parameters are not bounding compared to the measured values, corrective actions (i.e., reanalyzing with as-built values) are implemented, and the systems are retested. The Phase II initial startup testing is performed after the reactor core is loaded into a reactor module. These Phase II 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 analyses use the vibration level from the testing as input to subsequently verify that the design is acceptable.

DCA Part 2, Tier 2, Section 3.9.2.1.1, "Piping Vibration Details," states that vibration test specifications were developed, and that piping vibration testing and assessment were performed, in accordance with ASME Operation and Maintenance of Nuclear Power Plants, Division 2, 2012 Edition, Part 3. SRP Section 3.9.2, Revision 3, references the ASME OM Standards and Guides 2000 Edition. The NRC staff finds that the use of ASME OM Code, Division 2, 2012 Edition, is acceptable because the provisions for piping vibration and thermal expansion testing are equivalent.

DCA Part 2, Tier 2, Section 3.9.2.1, includes COL Item 3.9-13, for the COL applicant to assess and select piping systems for the vibration and thermal expansion 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 testing during initial startup testing. Additionally, ASME OM Code, Division 2, 2012 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.

DCA Part 2, Tier 2, Section 3.9.2.1.1.1, "Main Steam Line Branch Piping Acoustic Resonance," addresses the concern of potential fatigue failure of main steam (MS) line branch piping due to acoustic resonances (ARs). COL Item 3.9-10 addresses the detailed design of the MS piping by the COL applicant. This COL item ensures that 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 3.9-10 will ensure that the design of the piping systems will preclude significant ARs at closed pipe branches.

DCA Part 2, Tier 2, 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 Service Level A and B transient events. In addition, the tests verify that the component supports can accommodate the expansion of the piping for the service levels for these modes of operation. DCA Part 2, Tier 2, 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, 2012 Edition, Part 7. The staff finds that performing the piping thermal expansion testing according to OM Code, Division 3, 2012 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

DCA Part 2, Tier 2, Section 3.9.2.2, "Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment," references DCA Part 2, Tier 2, 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 sections.

3.9.2.4.3 *Dynamic Response Analysis of Reactor Vessel Internals under Operational Flow Transients and Steady-State Conditions*

3.9.2.4.3.1 *Design and Operation Summary*

In DCA Part 2, Tier 2, Section 1.2.1.1.2, "Operating Characteristics," and Section 3.9.5, "Reactor Vessel Internals," describe the NPM and its RVI components. Aspects of the design that are relevant to FIV are summarized here. The NPM comprises a reactor core, PZR, and two HCSGs within a cylindrical RPV, which is housed in a cylindrical steel CNV. The NPM operates with natural circulation primary coolant flow, reducing the strength of flow-induced forces compared to those in a typical PWR. The NPM has no pumps; therefore, there are no pump dynamic forces or inlet flow jets that impinge on reactor components. The NPM rests in a reactor pool 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., decay heat removal system (DHRS) and ECCS). A power plant comprises up to a maximum of 12 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 Category I nonprototypes per RG 1.20. A single NPM is smaller than currently operating PWRs, and outputs power up to 160 megawatts thermal. Unlike traditional PWRs with forced primary coolant circulation, the core flow rate is proportional to the plant's power. The HCSG is integral to the NPM and therefore is evaluated along with RVIs. Also, all piping systems and valves, including those outside the containment vessel and up to the NPM disconnect flange, are evaluated for FIV effects. FIV effects on the RVIs, HCSG, 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 steamlines and isolation valves may experience 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 CVCS and piping connection to the DHRS. The RPV is a 17.7-m-high, 2.7-m-diameter (58-ft-high, 9-ft-diameter) cylinder that is rated to operate at 12.76 MPa (absolute) (1,850 psia)). 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 head penetrations (less than 20-cm (8-in.)-diameter) accommodate the PZR spray, RVVs, reactor safety valves (RSVs), and CRDMs. 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 rests on the CSA. The upper riser assembly (URA) is suspended from the PZR baffle plate by an upper hanger brace and sits over the lower riser with a slip joint along a conical mating section. A bellows is included in the upper riser to allow for relative movement between the upper and lower risers (primarily caused by thermal expansion) at the slip joint and to minimize the likelihood of significant leakage flow between the hot and cold legs.

The dome of the RPV houses the PZR system. The primary coolant turns downward below the PZR baffle plate and passes the SG tubing 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 PZR 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. Support plates with several holes are attached to the risers to accommodate control rod drive (CRD) shafts and in-core instrumentation guide tubes (ICIGTs), which are inserted into the top of the RPV and extend downward 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 CRD shaft supports and the CRD shaft and ICIGT structures.

The once-through HCSGs consist of two independent bundles of tubing within the annulus between the riser and the wall of the RPV. The tubing for each bundle is welded to tubesheets at two integral feed and steam (near the top) plenums, thus forming a pressure boundary between the primary and secondary coolant. The tubing is held in place by arrays of tube support assemblies mounted on upper SG supports attached to the PZR baffle plate and interfaced with lower SG supports that are attached to the RPV. Nominal clearances are specified between the tubing supports and tubes. Inlet flow restrictors (IFRs) are mounted to a plate and inserted into all HCSG tubing inlets to mitigate the strength of low-frequency density wave oscillation (DWO) instabilities within the tubes during normal operation. Although the IFRs should improve the stability of the secondary coolant flow, NuScale has not yet demonstrated that DWO instabilities will not occur at or near full plant power conditions. DWO instabilities can cause reverse secondary coolant flow through the affected HCSG tubes and through the affected IFRs. Flow-induced forces resulting from DWO on the HCSG and IFRs have not yet been defined or quantified. As discussed further below, the NRC staff is unable to make a finding with regard to the potential for DWO instabilities in the secondary-side of the NuScale SGs and this issue will need to be addressed by a potential future COL applicant referencing the NuScale design.

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 PZR baffle plate and then downward though the annulus between the riser and RPV wall over the HCSG tubing array. After passing over the HCSG tubing 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 HCSG tubing, as preheated 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 HCSG tubing 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 25 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 HCSG tubing also functions in conjunction with the DHRS. The DHRS provides secondary-side reactor cooling for non-loss-of-coolant accidents (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 HCSG loops inside the RPV. The DHRS condensers are connected to the two SG loops, rejecting heat to the water in the reactor pool. The flow rates through the HCSG, during DHRS operation, are lower than those during normal operation; therefore, DHRS operating conditions do not need HCSG 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 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. The staff's evaluation of the FIV analyses and testing is described below.

3.9.2.4.3.2 Analytic Flow-Induced Vibration Evaluation

The NRC staff based this evaluation of the applicant's FIV, RVI, and HCSG analyses on (1) the CVAP analysis and MIP technical reports and (2) four audits of the applicant's internal documents, drawings, and test data that were conducted from May 16, 2017, through November 2, 2017; September 5, 2018, through October 4, 2018; March 4, 2019, through September 10, 2019; and June 24, 2019, through September 30, 2019. The staff used the audits to assess the details of the analyses. Detailed audit reports are available in ADAMS under Accession Nos. ML18023A091 (nonproprietary), ML18022A377 (proprietary); ML18333A221 (nonproprietary), ML18333A222 (proprietary); and ML19343C756 (nonproprietary). 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 and structures
- HCSG 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:

- HCSGs
    - tubing
    - tube support assemblies
    - lower SG supports

- upper riser assembly (URA)
    - upper riser section
    - riser slip joint
    - ICIGTs
    - CRD shaft
    - CRD shaft support
    - CRD shaft sleeve
    - upper riser hanger brace

- lower riser assembly

  - lower riser section
  - control rod assembly guide tubes (CRAGTs)
  - upper core plate

- CSA

  - core barrel
  - upper support block
  - core support block
  - reflector block
  - lower core plate
  - fuel pin interfaces

- other RVIs

  - pressurizer spray RVI
  - CVCS injection RVI
  - flow diverter
  - thermowells
  - component and instrument ports

Four small through-holes in the upper riser permit flow between the upper riser and steam generator 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 piping, nozzles, and MS isolation valves; also, CNTS drain branches
- HCSG steam plenum
- DHRS steam piping
- DHRS condensate piping
- HCSG tubing
- steam generator tube inlet flow restrictors (SGIFRs)

The NRC staff does not usually review SGs as part of an RVI CVAP. However, because the HCSG tubing is 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 components are susceptible to FIV based on low-flow conditions or robust structural designs, or both.

The applicant addressed the following FIV mechanisms:

- turbulent buffeting (TB)
- flutter and galloping (F/G)
- vortex shedding (VS)
- fluid-elastic instability (FEI)
- AR

- LFI

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. If "lock-in" were to occur, rapid failure of the associated SSC would be expected. However, if sufficient margin against the possibility of lock-in for FIV mechanisms exists, the response to those flow-induced loads is assumed to be negligible.

For much of its screening and analyses, the applicant relied heavily on the following references:

- ASME BPV Code, Section III, Nonmandatory Appendix N-1300, "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;
- a paper by S.S. Chen on FEI and VS in HCSG 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-0716-50439
    (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, or both, to substantiate its analysis procedures, the applicant has performed minimal benchmarking to date. The NRC staff assessed the significance of this lack of benchmarking during the CVAP review and the 2017 and 2018 audits (ADAMS Accession Nos. ML18023A091 and ML18333A221, respectively), as further discussed below.

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)

- assumed structural damping

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. The NRC staff's evaluation of the analysis methodologies and FIV mechanisms is 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 beam-like 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 but and are based on assumed reactor core and SG power density and loss coefficients. Therefore, spatial variations of the flow through the core and HCSG were not computed (only bulk velocities are available for those regions). CFD grid refinement studies verified flow velocity convergence throughout the primary coolant flowpath. 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 maximum 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 HCSG TB, HCSG 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, 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. Also, any nonconservative biases in the applicant's assumed VS and/or FEI velocities are countered by conservatisms in the constants applied to their empirical FIV models, such as the assumed Strouhal Number for VS and the Connors constants in the FEI critical velocity assessments. The evaluation of TB and VS/FEI analyses is discussed later in this section of the SER.

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 HCSG tubing. The secondary coolant enters the HCSG tubing 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 (DCA Part 2, Tier 2, 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 these forces to its models of the HCSG piping in the TF-2 test (DCA Part 2, Tier 2, Section 1.5.1.4, "Steam Generator

Thermal-Hydraulic Performance Testing—Fluid-Heated Facility”), where tubing vibration was measured in the presence of both primary and secondary flow. The estimated strains are lower than the background noise in the TF-2 spectral measurements, showing that the internal forces observed in TF-1 testing do not induce significant vibration (details are provided in NuScale TR-0918-60894). There are also no peaks visible in the TF-2 measurements that are indicative of strong internal flow excitation. The NRC staff finds that there is reasonable assurance that the secondary coolant flow will not cause adverse FIV effects on the SG tubes because the analyses have been shown to be conservative through benchmarking against the TF-1 and TF-2 test results. The analysis combines the highest measured spectra from TF-1 and TF-2 (the measured strong spectral peaks from the TF-1 internal flow and the measured spectra from the TF-2 external flow); therefore, the analysis is conservative.

3.9.2.4.3.4 Structural Mode Shapes and Resonance Frequencies

The ANSYS software suite, which includes structural finite element (FE) and CFD modeling tools, was used to estimate structural mode shapes and resonance frequencies of the NPM RVIs and HCSG. DCA Part 2, Tier 2, Section 3.9.1.2, states that ANSYS is a preverified 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 ASME BPV Code Appendix N. The secondary coolant mass densities were also applied to the HCSG tubing to compute the inservice 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 CRDSs, 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.

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 HCSG tubing FEI analysis but did not provide validated test data to substantiate this increased damping. The higher damping is assumed to be caused by interaction 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 tubing and tube supports at normal plant operating conditions. The higher assumed damping leads to higher estimated margins against FEI. The applicant stated during the 2017 and 2018 audits (ADAMS Accession Nos. ML18023A091 and ML18333A221, respectively) that upcoming SIET TF-3 modal testing (as described in TR-0918-60894) will substantiate the higher damping. The NRC staff finds that this modal testing, which the applicant committed to perform in DCA Part 2, Tier 1, Section 2.1.1, combined with flow testing to establish ranges of possible VS and FEI, will be adequate to validate the applicant’s assumed damping.

3.9.2.4.3.6 Turbulent Buffeting Analyses

The applicant evaluated TB-induced vibration of the overall RVI assembly, extending from the PZR baffle plate to the lower core supports, as well as of individual CRD shafts, ICIGTs, and CRAGTs. The applicant also analyzed the HCSG tubing and IFR. The CRD shafts and ICIGTs pass through holes in several CRD shaft supports spaced vertically throughout the RPV. The clearances between the ICIGTs and the through holes in the CRD shaft supports are very small, which provides lateral support for the ICIGTs due to the fluid “squeeze film” effect. The clearances for the CRD shafts are not small, but manufacturing and assembly variability will lead to contact at some of the through holes which will provide CRD shaft lateral support. However, most CRD shaft through-hole locations are assumed to have inactive supports, which leads to lower resonance frequencies; the NRC staff confirms that this is conservative and acceptable.

Since TB loads are spatially and temporally random, random forced response analysis methods are used. Vibration and alternating material stresses are very small due to the low primary coolant flow speeds. Fatigue and wear due to impacts and continuous sliding were estimated for components with nonnegligible TB-induced vibration amplitudes and, in particular, for cases with high relative motion between components and neighboring supports. 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 fretting stress estimates for all evaluated components are negligible. However worst case sliding wear estimates for the CRD shaft through-hole locations, if assumed to occur at a single location over the full plant life, are significant, sometimes exceeding the structural wall thickness. However, since this amount of wear is based on extremely conservative assumptions (in particular, all wear at a single location throughout the life of the plant), it is unlikely to occur. Also, the wear would occur mainly on the neighboring support plates, which are constructed of softer material than the CRD shafts. Therefore, the CRD shaft supports are more susceptible to wear than the CRD shafts. Finally, significant wear in the support plate holes would not lead to any safety issues. Nevertheless, the applicant included the CRD shaft supports in the long-term ISI plan per DCA Part 2, Tier 2, Table 5.2-7, “Reactor Vessel Internals Inspection Elements.” Any significant wear in the support holes, adjacent to any CRD shaft, will trigger an inspection of the CRD shaft itself.

The empirical forcing functions used to assess TB of the CRAGTs are highly uncertain. The NRC staff’s review of the CRAGT geometry and the core region, confirms that the flow will pass both within and outside the CRAGTs, with flow possibly repeatedly entering and exiting the CRAGTs through many holes in the sides of the tubes. This type of flow is not well understood, and no bounding forcing models are available in the ASME BPV Code or open literature. In a letter dated July 25, 2018 (ADAMS Accession No. ML18206A815 (nonproprietary) and ML18206A816 (proprietary)), the applicant provided two means of establishing the long-term structural integrity of the CRAGTs. First, similar designs in other plants have been operating for many years under much higher flow speeds with no observed degradation. Second, the applicant estimated a very low wear of 3.5 percent of wall thickness using conservative forcing functions. Finally, in accordance with DCA Part 2, Tier 2, Table 5.2-7, the CRAGT is included in the ISI program to ensure that unforeseen degradation does not progress to the point of component failure over the life of the plant. The NRC staff finds the CRAGT wear analysis to be acceptable because the inservice reliability of similar designs in stronger flow fields in operating plants and conservative analysis results provide reasonable assurance that FIV will not adversely affect the NuScale CRAGTs.

In DCA Part 2, Tier 2 Section 3.9.5.1, the applicant evaluated the effects of the upper riser through holes on TB. The holes are small enough so that structural modes of the upper riser are not affected significantly. Also, although the holes 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. The applicant has established that the potential static and dynamic jet flow loads on SG tube are minor, and much less than that due to downward primary coolant cross flow and from thermal expansion. The NRC staff conducted an audit of the applicant's evaluations (see audit report ADAMS Accession No. ML20160A224) and 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 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 HCSG are based on appropriate modeling procedures, assumptions, and inputs, and are reasonable and conservative. No significant TB-induced degradation 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 forcing 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 HCSG tubing), even higher vibrations and damage could occur. All components subjected to cross-flow were screened for susceptibility to VS. The only components, after screening, that warrant additional evaluation against VS/lock-in are the lower regions of HCSG tubing. All other components were designed to ensure that the VS frequencies are well below any structural resonance frequencies.

Only the lower HCSG tubing is subject to VS/lock-in since the primary coolant flows downward, and the lower tubing has no downstream structures to break up the shed vortices. However, all HCSG tubing may experience FEI at and above critical flow velocities. The applicant acknowledged that these components need validation testing to ensure margin against these mechanisms, and reported the following margins in TR-0716-50439, Revision 2, in accordance with ASME BPV Code, Appendix N, VS/lock-in criteria:

- lower HCSG tubing VS/lock-in: [[ ]] percent
- HCSG tubing FEI: [[ ]] percent

The margins are for primary coolant flow rates at 100-percent power ([[ ]] margin implies plant power would need to increase significantly above 100 percent to induce FEI in the HCSG). The NRC staff notes that these margins are based on conservative assumptions. The actual best estimate margins of safety, listed in TR-0918-60894, Revision 1, are [[ ]] for lower HCSG tubing VS/lock-in and [[ ]] for HCSG tubing FEI.

The methods used by the applicant are consistent with those in the ASME BPV code, the analysis inputs are generally conservative, and positive margins against VS and FEI will be demonstrated in TF-3 testing (see discussion below) prior to initial plant startup. The components evaluated for VS and FEI will be monitored during initial startup testing, and those components are part of the ISI program per DCA Part 2, Tier 2, Table 5.2-7. The NRC staff therefore finds that there is reasonable assurance of no significant VS- or FEI-induced vibration and structural damage for the life of an 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 coincide 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 have also been observed in nuclear power plants and 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 containment system (CNTS). Analysis margins against the first-order AR are shown in parentheses for each component.

- primary coolant
  - RCS injection to RVV/RRV reset lines ([[ ]] percent)
  - RRV nozzle ([[ ]] percent)
  - CNTS CVCS drain valve branches ([[ ]] percent)
  - flowmeter port ([[ ]] percent)
- secondary coolant
  - CNTS MS drain valve branches ([[ ]])
  - CNTS FW drain valve branches ([[ ]] percent)
  - SG system pressure-relief valve branches ([[ ]] percent)
  - main steam isolation valve (MSIV) upstream and downstream bypass lines ([[ ]])
  - DHRS piping tees
    - steamline tee ([[ ]])

- condensate line tee ([[ ]] percent)

By adhering 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. The only components with less than 100-percent margin against first-order AR are in the secondary coolant system: the DHRS steamline tees ([[ ]]), MSIV bypass lines ([[ ]]), and CNTS MS drain valve branches ([[ ]]). However, these components might experience second order instability AR at less than the full plant power level:

- DHRS steamline tees (possible at [[ ]] plant power)
- MSIV upstream and downstream bypass lines (possible at [[ ]] plant power)
- CNTS MS drain valve branches (possible at [[ ]] plant power)

As noted in DCA Part 2, Tier 2, Table 14.2-108, "NuScale Power Module Vibration Test # 108," these locations will be monitored during the initial startup testing (i.e., 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, there is significant estimated margin against the first-order AR, and all components with less than 100-percent margin for AR will be tested during the initial startup. Therefore, the NRC staff finds that there is reasonable assurance against significant AR-induced vibration and that if AR occurs, it will be detected so that changes could be implemented to preclude damage.

In DCA Part 2, Tier 2, 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 acoustic resonance. The NRC staff conducted an audit of the applicant's evaluations (see audit report ADAMS Accession No. ML20160A224) and finds that the riser holes will not cause acoustic resonances because there is flow through the holes and the flow instability frequencies and the upper riser acoustic frequencies are well separated.

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 flowpaths 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 CVAP technical report 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
- SG IFR for forward secondary coolant flow
- slip joint between upper and lower riser
- CRAGT at CRAGT support plate

The NRC staff evaluated the LFI evaluation procedures in the applicant's cited references and performed sample confirmatory calculations. The procedures are reasonable and validated against test data, and the NRC staff's confirmatory calculations are consistent with the applicant's (provided during a 2019 audit (ADAMS Accession No. ML19340A015)).

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.

There is no analysis for the array of SGIFRs that diffuses secondary coolant inlet flow and mitigates DWO instabilities within the HCSG tubing. Instead, various flow restrictor designs were tested under forward flow conditions in a special fixture (described in TR-0918-60894, which includes the SGIFR design concept LFI test results). The NRC staff examined these test results in detail during a 2018 audit (ADAMS Accession No. ML18333A221). The final SGIFR design is based on a concept that showed no signs of LFI or any other significant FIV. However, final validation testing of the current SGIFR design for forward flow is still planned prior to the NPM initial startup and is described in Section 5.3, "Steam Generator Inlet Flow Restrictor Validation Testing," of TR-0918-60894.

Despite the high estimated margins against LFI for forward flow, all screened components will be instrumented 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 DCA Part 2, Tier 2, Table 14-2-108 and Section 6.0 of TR-0918-60894). 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, all components evaluated for LFI will be tested during initial startup, and those components are part of the ISI plan. Therefore, the NRC staff finds that there is reasonable assurance of no significant LFI-induced vibration and structural damage under forward secondary coolant flow conditions for the life of an NPM.

3.9.2.4.3.11 Density Wave Oscillation Instability

NuScale has not submitted sufficient information to rule out the possibility of secondary coolant DWO instabilities, nor to define flow-induced loading of HCSG tubing and IFRs due to those instabilities. A DWO instability in the secondary coolant within the HCSG tubing could lead to slow oscillations of the boundary between the inlet subcooled liquid and the outlet steam. For mild DWO instability, the subcooled liquid will flow backwards through the IFRs. In strong DWO instabilities, the boiling boundary (sometimes called a “slug”) and even the steam can flow backwards through the IFRs. Although the IFRs will limit the conditions under which DWO instabilities can occur, those conditions remain unknown. Also, the following loading mechanisms associated with DWO instabilities are not yet defined or quantified:

- LFI in the IFRs due to reverse flow (either liquid or steam) (the staff notes that the cantilevered IFRs are less stable under reverse flow since the cantilevered end is opposite the flow entrance)
- pressure drops cycling through the IFRs
- cavitation loads on the IFRs and tubing interior walls for reverse steam flow
- oscillating strains in the tube walls induced by thermal gradients
- sliding between the tubes and tube supports induced by thermal gradients

No methodologies have yet been described for evaluating the loads due to these mechanisms, nor for estimating the effects of these loads on the long-term structural integrity of the HCSG tubes and IFRs. Possible effects on the tubes and IFRs include the following:

- fatigue of bolted joints and, as a result, loose IFR parts caused by LFI and/or pressure drop loading cycles
- cavitation erosion of SG tube walls and IFRs that could further worsen stability
- high-cycle fatigue failure of the tubing and tubing welds
- wear of the SG tubing as it slides past support tubes

TF-2 testing and several relevant cases in the open literature demonstrate clearly that the period of a secondary coolant DWO instability cycle is much longer than the periods of structural resonances in the HCSG system and tubing. Therefore, DWO instabilities will not couple strongly with any system resonances. However, the other loading mechanisms listed above have not yet been addressed, and their effects on long-term structural integrity are unknown. This aspect of the structural integrity of the HCSG tubes and SGIFRs due to DWO instability has not been reviewed by the NRC staff and therefore is not resolved. Because the NuScale application neither describes nor analyzes the potential for DWOs in the SG tubes, or qualification of the SGs given the occurrence of DWOs, including the effects of reverse fluid flows within the tubes and SGIFRs for the NuScale nuclear power plant, the NRC staff does not have a basis for making findings regarding the potential for secondary-side instabilities in the SGs or the adequacy of an evaluation methodology to address this condition. The NRC staff determined that COL Item 3.9-14 proposed by the applicant and described in Section 3.9.1.4.2, was not sufficient for the NRC staff to conclude its review in this area. Therefore, the NRC staff recommends that the Commission include language in the proposed rule stating that the NRC is not making a finding with regard to the secondary-side instabilities in the SG design or the associated evaluation methodology. Specifically, 10 CFR Part 52, Appendix G for the Design Certification for the NuScale SMR, Section VI, “Issue Resolution,” will state that the evaluation of secondary-side instabilities in the SGs, particularly with regard to the potential for DWOs and the associated evaluation methodology are not considered resolved within the meaning of § 52.63(a)(5) by the design certification. Additionally, Section IV, “Additional Requirements and Restrictions,” will require the COL applicant to complete the analysis of secondary-side instabilities in the SGs, taking into account the potential DWOs in the SG tubes and qualification of the SGs given the occurrence of DWOs, including the effects of reverse fluid flows within the tubes and SGIFRs, and provide an associated evaluation methodology.

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 HCSG 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 following:

- HCSG tubing (SIET TF-1 secondary flow testing, SIET TF-2 primary and secondary flow testing)
- SGIFR design concepts (for LFI under forward secondary coolant flow conditions)

TR-0918-60894, Revision 1, describes the results of the benchmark tests. Several candidate SGIFR designs were tested for LFI and TB. The chosen design did not experience any significant vibration, even at flow rates significantly higher than expected during normal NPM operation. Some peaks exist in TF-2 HCSG tubing vibration spectra, along with unexpected strong forces induced by two-phase secondary flow within the tubing (TF-1 testing). However, the variation of TF-2 vibration peaks with increasing flow is not indicative of VS or FEI behavior. Also, the unexpected TF-1 forces, when applied to models of the HCSG piping, 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.

Since there is no benchmarking of the VS or FEI analyses of the HCSG, and since the TF-2 tests were conducted on a nonprototypic HCSG and support system, the upcoming SIET TF-3 validation tests to be performed by the COL applicant are critical to confirm the adequacy of the HCSG design. Also, since the SGIFR final design deviates slightly from the concept chosen during the benchmarking testing, final SGIFR validation FIV testing will confirm the lack of significant FIV during normal operation. Finally, no benchmark testing was performed to assess the possibility of AR in the steam system. Instead, as discussed above, the components with less than 100-percent estimated margin against AR will be instrumented during the initial startup testing. These planned tests are evaluated in Section 3.9.2.4.4 below.

3.9.2.4.4 Flow-Induced Vibration Testing and Inspection of Reactor Vessel Internals

DCA Part 2, Tier 2, Section 3.9.2.4, COL Item 3.9-1, states that a COL applicant that references the NuScale Power Plant DC 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 the NRC staff. The NRC staff finds COL Item 3.9-1 to be acceptable because the COL item description is consistent with the provision in RG 1.20 regarding submitting the CVAP, including providing test procedures that contain acceptance criteria.

The planned measurement program details for validation testing of the final SGIFR design and of the HCSG in the SIET TF-3 facility are provided in TR-0918-60894, Revision 1. A second report (to be submitted after the DC review) will include the post-measurement evaluations and will be submitted after the completion of the first validation testing and will be updated after the completion of each subsequent validation test, as well as after the initial startup testing.

Validation testing will be performed on the following:

- the final SGIFR design (for LFI under forward secondary coolant flow)
- HCSG tubing with prototypic supports (SIET TF-3 for modes, damping, VS, and FEI)

According to Section 4.0, “Vibration Measurement Program,” of TR-0716-50439, the initial startup testing 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

The acceptance criteria are defined for the SGIFR and HCSG validation testing. Development of the detailed acceptance criteria for the initial startup testing is deferred to the COL applicant. To ensure that the acceptance criteria are appropriate and that corrective actions will be taken if the criteria are violated, the applicant committed (in DCA Part 2, Tier 2, Section 14.2), to meet Criterion XI, “Test Control,” of 10 CFR Part 50, Appendix B. The NRC staff finds that COL Item 3.9-1 provides reasonable assurance that the COL applicant will establish appropriate test acceptance criteria and the Appendix B requirements ensure that the test acceptance criteria are met. The NRC staff’s evaluations of each test are provided below.

3.9.2.4.4.1 Final Design Testing of the Steam Generator Tube Inlet Flow Restrictors

Validation testing of the final SGIFR design under forward secondary coolant flow will be performed before the initial startup testing to obtain early validation of the SGIFR design, obtain more data than would be possible in the first module initial startup testing, and evaluate more operating conditions. A facility like the one used to screen early SGIFR design concepts will be constructed but with expanded capabilities for vibration and flow testing. The applicant specified requirements for the test facility, instrumentation, data acquisition, and operation, along with a list of planned tests, in Section 5.3, “Steam Generator Inlet Flow Restrictor Validation Testing,” of TR-0918-60894. The fundamental bending modes of the SGIFR will be measured, along with FIV over a wide range of forward flow rates that significantly exceed those of normal plant operation. Testing will be at prototypic temperatures, but lower than prototypic pressures, which the NRC staff finds to be acceptable for LFI assessments, as static pressure has an insignificant effect on FIV mechanisms. Tests will be conducted for varying bolt tightness and SGIFR radial eccentricities, including cases in which the SGIFR contacts the SG tube wall. Each test duration will ensure at least 1×10^6 cycles of vibration. This testing will establish that the SGIFR is not susceptible to LFI at normal plant operating conditions. In addition, because the CVAP FIV assessments included the SGIFRs, they are part of the applicant’s initial startup testing inspection plan in accordance with TR-0918-60894, Table 7-1, “Pre- and post-initial startup testing inspection locations.” The NRC staff finds the SGIFR testing scope and instrumentation to be acceptable for forward secondary coolant flow as they will capture the presence of any unexpectedly high FIV. However, the staff cannot make a safety finding on the long-term integrity of the SGIFRs under possible reverse secondary coolant flow (either subcooled or two-phase) due to DWO instabilities. As discussed above in Section 3.9.2.4.3.11, a future COL applicant referencing the NuScale design will be responsible for providing an analysis of the potential for DWO secondary-side instabilities in the SGs for staff review and approval.

3.9.2.4.4.2 Flow-Induced Vibration Testing of Helical Coil Steam Generators in Test Facility 3

The susceptibility of the HCSG to VS and FEI will be evaluated prior to the initial plant startup in the SIET TF-3 test facility. As with the SGIFR validation testing, evaluating the HCSG in a separate facility will allow 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 will allow the applicant to more conclusively establish the ranges of operating conditions under which VS and FEI will and will not occur. However, the TF-3 flow validation testing results will not be available during the DC review.

The applicant specified in Section 5.1, “TF-3 Validation Test” of TR-0918-60894 requirements for the test facility, instrumentation, data acquisition, and operation, along with a list of planned tests. The NRC staff performed an onsite audit of the TF-3 facility (in its current state). During the audit, the TF-1 and TF-2 facilities used for previous benchmark testing were also examined. A full audit report, dated January 8, 2020, may be found at ADAMS Accession No. ML19343C756.

During the audit, the NRC staff evaluated the applicant’s plans for attempting to mimic thermal expansion effects in TF-3. In the actual plant, thermal expansion, as well as other forces, is expected to tighten the fits between the HCSG tubing, the support tabs, and the support structures. However, the compression system installed in TF-3 does not fully emulate prototypic conditions, and some tube-to-support connections were loose during the audit. Several tube-to-support connections will likely remain loose throughout TF-3 testing. Therefore, the TF-3 HCSG, though prototypic in all other aspects, will likely be more susceptible to VS and FEI due to the looser connections, and the possibility of inactive supports (tubes that do not contact neighboring tabs). However, this causes the natural frequency of the HCSG tubes to be lower and the TF-3 to be conservative. The applicant performed engineering calculations of the expected thermal, as well as gravitational, fluid dynamic, internal and external pressure, and Coriolis forces acting on the HCSG tubing. The net forces are expected to ensure vertical and/or radial contact between the tubing and adjacent supports. Although the calculations do not conclusively confirm that all tubing to support and support connections will be tight in the actual plant, the connections will be no looser than those in the TF-3 validation testing facility, which the NRC staff finds to be reasonable and acceptable.

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. The procedures are reasonable and include uncertainty in modeling and analysis (used to estimate the actual in-plant margins against VS and FEI) and measurement methods. The applicant computed these total uncertainties for both VS and FEI. The applicant also estimated bias corrections that relate critical flow velocities for VS and FEI in TF-3 to those in the actual plant. Allowable ranges of flow velocities are specified for both mechanisms in TF-3. The lower allowable limits for critical FEI and VS flow velocity are equivalent to [[ ]] and [[ ]] margins 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 HCSG tubing will also be 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. In addition, the provisions in DCA Part 2, Tier 1, Section 2.1.1, and Part 2, Tier 2, Table 14.2-72, "Steam Generator Flow-Induced Vibration Test #72," regarding TF-3 provide reasonable assurance that the TF-3 validation testing will be performed and the acceptance criteria will be met.

3.9.2.4.4.3 Initial Startup Testing

Planned initial startup testing of the prototype NPM for FIV mechanisms of RVIs is limited to (1) performing a flow test of the CNTS MS piping to confirm the lack of significant AR and (2) identifying and localizing any unexpectedly strong FIV effects in RVIs and the HCSG. Instrumentation options have been specified for the following RVI components in TR-0918-60894, Revision 1: upper and lower riser, CRD shafts, ICIGTs, and the HCSG assembly.

TR-0918-60894 provides specifications for testing CNTS MS line piping branch connections (DHRS steam piping tees, MS drain valve branch, and MSIV upstream and downstream bypass lines) to assess any significant AR effects. 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 recent extended power uprates. Both MS lines will be instrumented. This combination of instrumentation is sufficient to determine whether significant ARs are present.

A preliminary instrumentation list is provided for RVIs and the HCSG assembly, along with guidance for acceptable use in operating nuclear power plants. The NRC staff finds that the instrumentation types selected are appropriate for the targeted components and FIV mechanisms of interest and therefore should withstand the temperatures and pressures within an NPM throughout the initial startup testing. While not yet finalized, the breadth of instrumentation should provide sufficient redundancy in the event of a few failures. Sensitivities and dynamic ranges are appropriate and typical of past in-plant measurement programs. Installation procedures and cable routing are also consistent with current best practices for nuclear power plants, with locations chosen to avoid strong cross-flows which might cause cabling failures. Specifically, the following instrumentation is planned:

- Strain gauges are to be mounted to the outermost HCSG tubing at high, middle, and low elevations.

- Strain gauges and accelerometers are to be mounted to the most susceptible ICIGTs.

- Strain gauges and accelerometers are to be mounted to sleeves and support plates adjacent to the CRD shafts.

- Accelerometers are to be mounted near the upper and lower riser slip joint.

Based on these configuration plans, this instrumentation should capture any unexpectedly strong FIV mechanism for these components and is, therefore, acceptable to the NRC staff. The instrumentation may be removed after the initial startup testing in accordance with ASME BPV Code NB-4435, “Welding of Nonstructural Attachments and Their Removal.”

Provisions for testing procedures and data acquisition have been provided and will be finalized 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. 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. The minimum required test time for RVIs and the HCSG will be less than 2.5 days. AR testing will include varying flow rates to identify both the second-order and the first-order shear layer instabilities and requires less testing time since the expected AR frequencies are higher. Although flow rates that lead to the second-order ARs should be identifiable (because they occur at flow velocities lower than the maximum design speed), it is expected that the first-order shear layer ARs will not occur because those corresponding flow speeds are expected to be slightly above the maximum design flow velocity.

Pretest predictions have been provided for CNTS MS branch line testing and include uncertainty analyses consistent with the procedures used for other FIV mechanisms, which the NRC staff finds to be acceptable. Pretest predictions for the initial startup testing of RVIs and the HCSG simply assume that all vibration responses will be due solely to TB, and that instabilities like VS, FEI, F/G, and LFI will not occur. Trending of vibration and strain responses with plant power/flow velocity will be used to identify any instabilities that might appear. In the event that unexpectedly high vibration occurs and leads to intermittent impacts between adjacent structures, “crest factors” will be examined, which relate the ratio of peak to RMS levels. To capture crest factors appropriately, the applicant will acquire data at much higher sampling frequencies than those associated with low-order structural resonances. The NRC staff finds the pretest prediction and measurement methodologies to be acceptable because they are consistent with those for operating plant startup testing and the guidance in RG 1.20.

3.9.2.4.4.4 Inspections

In accordance with TR-0918-60894, 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 NG-5111, “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 IWB-2500, “Examination and Pressure Test Requirements,” Table IWB-2500-1 (B-N-1, B-N-2, B-N-3), “Examination Categories B-N-1, Interior of Reactor Vessel; B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels; B-N-3, 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 DCA Part 2, Tier 2, Section 5.2.4.1. 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 under Service Level D Conditions

This section contains four subsections. The first subsection contains the NRC staff's evaluation of the NPM seismic analysis. The second subsection contains the NRC staff's evaluation of the NPM short-term transient analysis. The third subsection is the RVI components stress analysis under the Service Level D Faulted Condition. And, the fourth subsection contains the NRC staff's evaluation of RPV primary stress under the Service Level D Faulted Condition.

3.9.2.4.5.1 Seismic Analysis of NuScale Power Module

TR-0916-51502 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 RVIs, lower RVIs, 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 forces, moments, and ISRS were used as input loads for the component-level stress analysis. This subsection evaluates the analysis methodology, structural modeling of the NPM component, and analysis results. The fuel seismic assessment is addressed in Chapter 4, "Reactor," of this SER.

3.9.2.4.5.1.1 Analysis Methodology

The NPM seismic analysis methodology consists of the following steps:

(1) Create a 3-D NPM ANSYS model. The model consists of six submodels (i.e., containment with the surrounding pool water, reactor vessel, lower RVIs, upper RVIs, CRDM, and TSS). The six submodels are connected by coupling nodes, constraint equations, contact elements, and fluid coupling to form the 3-D NPM model.

(2) Create a simplified NPM ANSYS beam model that is dynamically equivalent (dry and wet submerged conditions) to the 3-D NPM model. Modal, harmonic, and transient analyses are performed to tune the beam model to match the dynamic response of the 3-D NPM model.

(3) Convert the simplified NPM ANSYS beam model to the simplified NPM SAP2000 beam model for the system for analysis of soil-structure interaction (SASSI) analysis.

(4) Perform an SSI analysis with a SASSI model, including an NPM SAP2000 beam model in each of the 12 reactor bays along with the SAP2000 RXB model. The results of the SSI analysis indicate that NPM 1 and NPM 6 have the highest seismic response among the 12 NPMs.

(5) Create an ANSYS NPM entire pool model. In the NPM entire pool model, the 3-D NPM model is modified to include the entire volume of the reactor pool. The NPM entire pool model includes only NPM 1 or NPM 6.

(6) Perform a time history analysis of the NPM entire pool model by applying the calculated acceleration time histories from the SSI analysis as boundary conditions at the pool floor and walls, as well as at the NPM supports. For the 11 NPMs not included in the NPM entire pool model, the calculated NPM centerline acceleration time histories from the SSI analysis are applied to the surface of the fluid surrounding the NPM to simulate the effects of the missing NPMs. The NPM entire pool model is analyzed for 12 cases with the following conditions:

- one certified seismic design response spectra (CSDRS) compatible 0.5g horizontal ground acceleration and one CSDRS compatible 0.4g vertical ground acceleration
- one soil type
- two NPM module locations (NPM 1 and NPM 6)
- two RXB concrete conditions (cracked and uncracked)
- three NPM stiffnesses for the uncracked concrete condition and three NPM stiffnesses for the cracked concrete condition (nominal NPM stiffness, NPM stiffness adjustment = 1/1.3 = 77 percent of NPM nominal stiffness, and 130 percent of nominal stiffness)

(7) Generate seismic loads within the NPM from the 12 analysis cases.

(8) Perform a component-level stress analysis using the maximum seismic loads in step 7.

The NRC staff considers the methodology of the NPM seismic analysis to be acceptable because it is consistent with common engineering practice that an SSI analysis is performed first, and the output motions from the SSI analysis are then used as input motions for the reactor systems analysis. The NPM seismic analysis applies to the NPM situated in the operating bays and in the reactor flange tool (RFT). Refueling transition operations, such as the NPM suspended by the building crane, including the time in the containment flange tool, are not seismically evaluated, based on the short time durations in these configurations and because the RXB crane is always connected to the CNV when the NPM is in the containment flange tool to prevent the NPM from toppling. For a 12-module NuScale power plant, the time in refueling transition configurations is less than 10 days per year (assuming a 2-year fuel cycle per NPM and refueling of six modules per year). Although the time in refueling transition is not included within the scope of the NPM seismic analysis, the NRC staff determined that this is acceptable due to the short time duration of refueling transition operations. For these reasons, the NRC staff finds that the scope of the NPM seismic analysis is adequate.

3.9.2.4.5.1.2 NuScale Power Module Modeling

In the structural modeling of the NPM, a full 360-degree 3-D NPM ANSYS model was created for the NPM seismic analysis. The NRC staff finds that this is appropriate because the NPM does not have symmetric conditions for a half model or a quarter model. The 3-D NPM ANSYS model consists of six submodels representing the CNV and surrounding pool water in a single bay, RPV, lower RVI, upper RVI, CRDM, and TSS. In each submodel, the ANSYS eight-node shell element and eight-node solid element are used for modeling the geometry. A 360-degree shell cross section is generally represented by 16 shell elements in the circumferential direction. The mass of the shell and solid element was adjusted to account for the mass of bolting, piping fluid, and cables that the submodels excluded. The pool water is modeled by eight-node acoustic fluid elements (FLUID30). The six submodels are connected by coupling nodes, constraint equations, contact elements, and fluid coupling to form the 3-D NPM model.

The CNV submodel consists of the CNV. The RPV submodel consists of the RPV shell, top head, upper and lower supports, FW and MS plenums, and CRDM support structure. At the bottom of the RPV, the nodes associated with the RPV alignment feature and the nodes associated with the CNV alignment feature were coupled in the horizontal direction, but not in the vertical direction, to allow thermal expansion. The RPV upper support was coupled to the CNV ledges in the vertical and circumferential directions, but not in the radial direction, to allow thermal expansion.

The lower RVI submodel consists of the lower riser shell, the CSA (i.e., core barrel, lower and upper core plates, and reflector), fuel assemblies, and CRD shaft supports. Seismic inertia loads of the lower RVI were transferred to the RPV through several connections. In the horizontal direction, the upper support blocks of the lower RVI were coupled to the RPV by no-separation contact elements between the four pairs of mating surfaces. The lower core plate tabs of the lower RVI were coupled to the core support block assembly of the RPV. A simplified beam model represented the 37 fuel assemblies with mass and stiffness tuned to match the fuel assembly frequencies provided by the fuel vendor. The simplified fuel model was connected to the upper and lower core plates of the lower RVI. The reflector of the lower RVI was connected to the core barrel by no-separation contact elements.

The upper RVI submodel consists of the upper riser shell and five CRD shaft supports. Half of the SG mass was applied to the upper RVI, and the other half was applied to the RPV. Radial coupling was provided between the upper riser and the RPV for transferring the SG seismic loads. A bellows was installed between the upper riser shell and the upper riser conic section. The bellows allows for vertical thermal growth while limiting relative horizontal displacements between the upper riser and the lower riser. The upper riser was coupled to the lower riser in the horizontal direction only. The fluid in the annular region between the risers and the RPV was accounted for by fluid coupling between the risers and the RPV. Fluid coupling nodes on the inner and outer surfaces of the annulus were specified at 23 elevations along the annulus.

The CRDM submodel consists of 16 CRDM assemblies that were modeled by pipe, beam, and mass elements. The tops of the CRDMs were coupled laterally to the CNV top head, and the mid-heights of the CRDMs were coupled laterally to the CRDM support frame on top of the RPV. The mass of the CRD shafts was distributed to the CRDM and CRD shaft supports in the RVI, based on tributary length. The TSS submodel consists of the TSS top platform support frame and legs, lifting lugs, and MS piping. The NRC staff's evaluation of the structural modeling of the NPM components is described below.

SRP Section 3.9.2, Revision 3, states that the number of elements is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. In DCA Part 2, Tier 2, Section 4.1, "Seismic Model Methodology," the applicant performed a mesh sensitivity study of the NPM model. The modal analysis was carried out for a refined NPM model with a tenfold increase in the number of elements. A comparison of the modal response (frequency and associated mass participation) with the original coarse NPM model revealed that the difference was within 10 percent. The NRC staff finds the number of elements used in the modeling to be acceptable because the number of elements met the provision in SRP Section 3.9.2.

In TR-0916-51502, Revision 2, the lower RVI submodel modeled the fluid gap between the core barrel and reflector blocks using six Fourier nodes, one for each reflector block. The NRC staff finds the use of the Fourier node method in modeling the fluid gap between the core barrel and the reflectors to be acceptable because the Fourier node method can be used to simulate the dynamic response of two concentric cylinders with fluid gap.

TR-0916-51502, Revision 2, Section 3.1.1, "Detailed 3D Model of the NuScale Power Module in ANSYS," states that the NPM seismic analysis with the reactor pool considers acoustic wave energy dissipation by the RXB floor. Because the RXB is not included in the NPM seismic model, the wave energy dissipation by the RXB is simulated using an acoustic-absorption coefficient applied at the pool bottom surface. The applicant performed detailed fluid-structural interaction (FSI) analysis and had determined that an absorption coefficient of 0.7 is to be the best match between the model without RXB and the model with RXB. For conservatism, the applicant used an absorption coefficient of 0.4. On December 19, 2018, the NRC staff audited the detailed FSI analysis that yields the acoustic-absorbing coefficient. The NRC staff evaluated the applicant's seismic analysis methods, models, and results used to develop the acoustic attenuation coefficient (see the audit report dated March 27, 2019 (ADAMS Accession No. ML19072A136)). The NRC staff finds that the absorption coefficient of 0.4 applied to the bottom of the pool model to simulate wave energy dissipation by the RXB floor is conservative and reasonable based on the relative acoustic impedances of water and concrete and on the applicant's analyses of a detailed model of the overall building.

In the NPM design, the bottom of the upper riser shell is attached to a cone that connects the upper riser to the lower riser. There is a bellows between the upper riser shell and the upper riser conic section. The bellows allows for vertical thermal growth while limiting relative horizontal displacements between the upper riser and the lower riser. The bellows assembly consists of the overlapping upper and lower lateral restraints (i.e., sliding surfaces) and the bellows vertical expansion structure (i.e., convolutions). The applicant supplemented its DCA by a letter dated July 19, 2018 (ADAMS Accession Nos. ML18201A269 (proprietary) and ML18201A268 (nonproprietary)), in which it states that the lateral restraints provide structural support for the CRD shafts and are classified as a seismic Category I component subjected to stress evaluation under the ASME Service Level D condition. The bellows convolutions prevent bypass flow between the cold and hot legs with no structural support function during normal operation and are classified as seismic Category II components. The applicant also stated that the failure of bellows convolutions will not affect natural circulation of the reactor because of the narrow fluid path of the lateral restraints and low differential pressure in the primary loop. The applicant further supplemented its DCA by a letter dated October 15, 2018 (ADAMS Accession Nos. ML18288A271 (proprietary) and ML18288A270 (nonproprietary)), in which it states that the differential pressure between the hot leg and cold leg at the bellows is small (less than 7 kPa (1 psi)). These conditions result in an insignificant amount of bypass flow from failure of the nonseismic, convoluted portion of the bellows. The NRC staff determined that the bellows convolutions can be classified as seismic Category II components because of the narrow fluid path of the lateral restraints and low differential pressure and because they have no structural support function.

TR-0916-51502, Revision 2, Section 4.1.9.2, "Steam Generator Mass," states that half of the SG horizontal mass is applied to the riser and that the other half is applied to the inner surfaces of the RPV. The vertical mass is applied only to the upper SG supports to represent the floating vertical connection at the bottom cantilever interface. The NRC staff noted that the SGs are large components and that the 3-D NPM model did not include them. The NRC staff also noted that the applicant ignored the SG stiffness without assessing its effect on the results. The applicant supplemented its DCA by letter dated June 20, 2018 (ADAMS Accession No. ML18171A409), in which the applicant provided a calculation of the bending stiffness of the RPV and the aggregated bending stiffness of 84 SG tube support columns. The calculation showed that the bending stiffness of the tube support columns is four orders of magnitude lower than the RPV. The applicant concluded that neglecting the SG horizontal stiffness in the NPM model is justified. For the vertical SG stiffness, the NRC staff verified that the following criteria in SRP Section 3.7.2 for decoupling a subsystem from the primary system were met:

i. If $R_m < 0.01$, decoupling can be done for any R_f.

ii. If $0.01 < R_m < 0.1$, decoupling can be done if $0.8 > R_f > 1.25$.

iii. If $R_m > 0.1$, a subsystem model should be included in the primary system model.

Where:

R_m = the ratio of the total mass of the supported subsystem to the total mass of the supporting system.

R_f = the ratio of the fundamental frequency of the supported subsystem to the dominant frequency of the supporting system.

The NRC staff calculated the SG vertical frequency, R_m, and R_f based on the mass of the SGs and the aggregated vertical stiffness of the 168 SG tube support columns. The R_m and R_f values meet the decoupling criteria in (ii). The NRC staff concludes that the SG vertical stiffness can also be excluded in the NPM model. Therefore, the NRC staff finds that the SGs can be excluded in the 3-D NPM model.

3.9.2.4.5.1.3 Simplified NuScale Power Module Model

TR-0916-51502 states that two simplified NPM beam models were developed. The ANSYS model was created first and tuned to match the dynamic characteristics of the 3-D NPM model under dry and wet submerged conditions. The tuned ANSYS model was then converted to the simplified SAP2000 NPM model for use in the SSI analysis along with the SAP2000 RXB structure model. The simplified NPM models consist of the beam, spring, and mass elements. The CNV and RPV are represented by distinct beams that are coupled together using stiff spring elements. Torsional mass moment of inertia was assigned in the mass elements to account for torsional effects. A set of tuned point masses and springs represents the dynamic characteristics of upper and lower RVIs. The dynamic characteristics of the simplified models were tuned to match the 3-D NPM model under dry and wet submerged conditions. Tuning was accomplished by changing the elastic modulus material properties and by adding spring-mass systems to capture missing modes. After repeated tuning, the dominant natural frequencies and associated modal mass of the simplified models were mostly within a 10-percent difference compared to the 3-D NPM model in three directions. Static, harmonic, and transient analyses were also performed. The results of the two simplified models agree well with the results of the 3-D NPM model. Based on the comparison, the NRC staff determined that the applicant properly created the simplified SAP2000 NPM model for use in the SSI analysis. The SSI model contains 12 SAP2000 NPM models, one in each reactor bay. The results of the SSI analysis indicated that NPM 1 and NPM 6 have the highest seismic response among the 12 NPMs.

3.9.2.4.5.1.4 NuScale Power Module Entire Pool Model and NuScale Power Module Seismic Loads

Based on the results of the SSI analysis, the forces at the CNV support skirt and lug supports of the NPM at operating bays 1 and 6 bound those of the other NPM locations. NPMs at operating bays 1 and 6 were selected for the NPM seismic analysis to bound the seismic response of all NPMs. Two ANSYS NPM entire pool models were created for generating NPM seismic loads for the component-level stress analysis. Model 1 has an NPM at operating bay 1, and Model 2 has an NPM at operating bay 6. In both models, the 3-D NPM model was modified to include the entire volume of the reactor pool. A contact element was used between the RXB floor and the CNV skirt to capture any uplift of the NPM. A nonlinear time history analysis was performed by applying the calculated acceleration time histories from the SSI analysis as boundary conditions at the pool floor and walls, and at the NPM supports. For the 11 NPMs not included in the entire pool model, the calculated NPM centerline acceleration time histories from the SSI analysis were applied to the surface of the fluid surrounding the NPM to simulate the effects of the missing NPMs. For each NPM entire pool model, the following six conditions were considered:

(1) RXB uncracked concrete condition with nominal NPM stiffness, 77 percent of NPM nominal stiffness, and 130 percent of NPM nominal stiffness

(2) RXB cracked concrete condition with nominal NPM stiffness, 77 percent of NPM nominal stiffness, and 130 percent of NPM nominal stiffness

A total of 12 cases were performed for generating NPM seismic loads for the two NPM entire pool models.

Based on the results of the 12 analysis cases, the following seismic loads were generated at various locations within the NPM for the component-level stress analysis:

- displacement and acceleration time histories and broadened ISRS at 33 locations within the NPM
- maximum seismic forces and moments at 83 interfaces between the NPM components
- maximum seismic forces and moments within 22 NPM component sections
- maximum seismic forces at four NPM support locations

The paragraphs below discuss the NRC staff's evaluation of the NPM entire pool seismic analysis and system damping.

TR-0916-51502, Revision 2, Section 8.0, "Three-Dimensional Seismic Model Analysis," states that the 3-D NPM in the entire pool model was analyzed using the CSDRS compatible Capitola time histories. Outputs of the analysis include the ISRS, time history data, relative displacements, and forces and moments within the NPM. The analysis did not consider the high-frequency certified seismic design response spectrum (CSDRS-HF) compatible time histories (i.e., the Lucerne time histories). The applicant included COL Item 3.9-12 in DCA

Part 2, Tier 2, Section 3.9.1, for the COL applicant to address CSDRS-HF input at hard rock sites. The NRC staff finds COL Item 3.9-12 to be acceptable because it adequately addresses the NPM component design under CSDRS-HF inputs.

TR-0916-51502, Revision 2, Section 7.4, “Uncertainties in the NuScale Power Module Subsystem Model,” states that uncertainty in the input and assumptions used in the 3-D NPM model was accounted for by considering multiple analyses using ±30-percent variations of the stiffness properties of the model. The applicant performed 12 analysis runs in the NPM seismic analysis, including 77-percent and 130-percent nominal NPM stiffness. For each location, direction, and damping value, the response spectra were calculated for the 12 seismic analysis runs. Using each set of 12 response spectra, an envelope spectrum was constructed by finding the maximum of the 12 response values at each spectral frequency point. The envelope of the 12 spectra was then broadened by ±15 percent to produce the design ISRS. The maximum NPM uplift occurred in the case of the NPM in operating bay 6, with a cracked concrete condition and with an increased NPM stiffness. The NRC staff finds the seismic analysis method to be acceptable because it appropriately considers the variation of the NPM stiffness and the potential resonance between the NPM and RXB natural frequencies.

In TR-0916-51502, Section 4.1.3.3, “Lower Reactor Vessel Internals Boundary Conditions,” the applicant included ANSYS contact elements between reflector block and lower core plate to simulate uplift of the reflector blocks. The analysis reveals that the uplift distance of the reflector blocks is small and not sufficient to close the vertical gap above the reflector block and below the upper core plate. The impact force between the block and lower core plate is calculated for fuel design. The NRC staff finds the analysis method and results to be acceptable because the analysis appropriately considers the uplift of the reflector blocks from the lower core plate during a seismic event.

3.9.2.4.5.1.5 System Damping

TR-0916-51502, Section 8.1, “Transient Analysis,” states that the NPM component seismic analysis uses 4-percent damping. RG 1.61, Revision 1, Table 6, “Damping Values for Mechanical and Electrical Components,” recommends 3-percent SSE damping for pressure vessels and major pressure boundary components. The NRC staff finds that using 4-percent 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.

3.9.2.4.5.1.6 NuScale Power Module in the Reactor Flange Tool

TR-0916-51502, Revision 2, documents the seismic analysis of the NPM in the RFT for refueling. The NPM RFT model encompasses a subset of the NPM model, including the lower RPV, the lower RVIs, the core support structure with fuel, and a representation of the RFT. The RPV submodel was modified to incorporate the refueling ledge at the bottom of the RPV. The mass of the displaced pool water was accounted for by increasing the density of the RPV, RFT, and the lower RVIs. Acceleration time histories on top of the base mat, at the RFT location, were applied to the bottom of the RFT model. The analysis considered cracked and uncracked concrete in the RXB with normal and adjusted NPM stiffness. Uplift of the RPV from the RFT was considered using nonlinear contact element between the RPV and RFT. Time history analysis was carried out with 4-percent system damping. The output consists of ISRS at the top of the lower core plate and at the bottom of the upper core plate. The NRC staff finds the analysis method and results to be acceptable because they appropriately considered the mass of the pool water, the variation in the NPM stiffness, and damping.

3.9.2.4.5.2 Short-Term Transient Analysis of the NuScale Power Module

TR-1016-51669, Revision 1, documents the NPM short-term transient analysis. MSPB, FWPB, and DBPB cause short-term transient events that result in an asymmetric cavity pressurization load between the CNV and RPV and blowdown load within the RPV. The DBPB also includes the loads from the spurious actuation of the RVV or RRV and from a CVCS pipe break. The technical report contains the analytical methods, benchmarking for validating the analysis methods, and the resulting asymmetric cavity pressurization load and blowdown load. These short-term transient structural loads were combined with the SSE load for evaluating stress on the NPM internals under the Service Level D Condition. The thermal-hydraulic code NRELAP5 and the ANSYS model calculate the short-term transient loads. NRELAP5 generates thermal-hydraulic boundary condition inputs for the ANSYS model, which calculates the short-term transient structural loads within the NPM, including forces, moments, and differential pressure loads.

A 360-degree ANSYS model was developed to determine the asymmetric cavity pressurization load between the CNV and the RPV and the blowdown load within the RPV. There are three RVVs with nozzle diameters of 12.38 cm (4.875 in.) at the RPV head and two RRVs with nozzle diameters of 5.72 cm (2.25 in.) at the upper RPV shell. The six DBPB cases analyzed involved the inadvertent opening of each of the three RVVs, each of the two RRVs, and a CVCS pipe break. The CVCS has an outer diameter of 6.033 cm (2.375 in.). The flow from the RVV or RRV entering containment generates asymmetric loads for the CNV and RPV and blowdown loads within the RPV. In the ANSYS model, mass was adjusted by using a combination of point masses, distributed mass elements, and adjusted densities to account for the missing mass of the components that are excluded in the model. Fluid volumes were created between the RPV and the CNV as well as within the RPV. The ANSYS acoustic fluid element was used to model the fluid. The CNV, RPV, lower RVI, upper RVI, RVV, RRV, and CVCS injection line are represented by ANSYS solid elements. The fluid acceleration and thrust force calculated by the NRELAP5 analysis were applied to the elements at the break location. ANSYS time history structural analysis was carried out. For each case analyzed, forces and moments at key structural cross sections and differential pressure loads were calculated. The NPM design used the bounding forces and moments in the six cases. This section of the SER addresses the ANSYS modeling in TR-1016-51669, and SER Section 15.0.2 addresses the evaluation of the RELAP5 modeling.

TR-1016-51669, Section 3.2.3.2, "Flow Acceleration at Break Locations," states that flow acceleration was applied as a body force to the acoustic fluid element nodes on the pipe break face as input loads of the ANSYS model. The applicant supplemented its DCA by a letter dated October 29, 2018 (ADAMS Accession Nos. ML18302A296 (proprietary) and ML18302A295 (nonproprietary)), in which it states that the flow acceleration boundary generated in the NRELAP5 analysis is saved in a text file for each break location. ANSYS commands are used to read acceleration data from the text file and apply the acceleration time history to the acoustic element nodes on the break face. The same methods and ANSYS commands are used for the Heissdampf reactor benchmarking cases and for the simulated breaches in NuScale pressure boundaries. The simulated Heissdampf reactor boundary conditions and dynamic responses are in agreement with the measured results, as shown in Appendices C, F, and G to TR-1016-51669. The applicant further stated that agreement with the experimental results and the consistent method of applying the fluid loads between the benchmarking and design analysis calculations demonstrate that the input loading of the flow acceleration boundary condition is appropriate. The NRC staff finds the modeling approach to be acceptable because the ANSYS commands appropriately capture the input loading in the ANSYS model.

The applicant provided bounding values of the calculated forces and moments of the six cases of RPV valve openings and a CVCS pipe break in TR-1016-51669. In Table 6-4, "Maximum forces and moments at component interfaces," bounding values of maximum forces and moments at 83 NPM component interfaces for all break and valve opening conditions are provided. In Table 6-6, "Maximum forces and moments on containment vessel, reactor pressure vessel, riser, and core barrel assembly," bounding values of maximum forces and moments for 22 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. The applicant stated that the highest forces and moments result from one RVV opening case due to the high mass flow rate and high fluid accelerations generated in this valve opening event. The maximum forces and moments on RPV, CNV, and RVI due to the CVCS injection line break are bounded by the case of one RVV 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 Evaluation of a Reactor Vessel Internals Service Level D Faulted Condition

TR-0916-51502 describes the following major RVI components:

- CSA (core barrel, lower core plate, reflector, core support block assembly, upper core plate, and upper core support)
- lower riser assembly (lower riser section, lower riser transition)
- URA (upper riser section, upper riser transition, upper riser hanger ring, upper riser hanger brace, hanger threaded structural fasters, and lateral restraints of the bellows)
- CRAGT assembly (CRAGT, CRAGT support, CRA card, CRD shaft, and CRD shaft support)
- SG assembly (SG tubes, SG tube support columns, upper SG supports, and lower SG supports)

The applicant supplemented its DCA by letter dated August 2, 2019 (ADAMS Accession No. ML19215A004 (proprietary) and ML19215A003 (nonproprietary)), in which the applicant provided stress evaluation of the above-mentioned RVI components (including associated welds) under Service Level D Condition. The information includes component analysis methodology, load combination, component structure modeling, input loads, major assumptions, acceptance criteria, fluid modeling, boundary conditions, mass distribution, damping values, gap considerations, dominant modes and frequencies, and stress results. The applicant concluded that the stresses of the major RVI components satisfy the structural requirements of the ASME BPV Code for Service Level D loads.

The applicant stated in its letter that, depending on whether seismic forces and moments are calculated for a component under analysis, one of two methodologies is used in the RVI component stress analysis. In the first method, seismic forces and moments are applied at the cross sections of the components. For a component with simple geometry, such as the core barrel, stresses are calculated using closed form solution. In the case of a component with complicated geometry, such as the upper core plate, lower core plate, or lower riser section and transition, forces and moments are applied to the component's FEM. Stresses are calculated by static analysis. The second method is used when seismic forces and moments are not calculated for the component under analysis. In this case, ISRS are defined at the component supports. The ISRS are the bounding ISRS from the 12 seismic analysis cases as mentioned above. Stresses are calculated from the response spectrum method. Examples of components that were evaluated using the second method include CRAGT assembly, URA, and SG assembly. The NRC staff finds that the analysis methodologies for the RVI components are appropriate because the analysis methodologies are consistent with standard practice. The 4-percent damping used in the response spectrum method is consistent with the damping value used in the NPM system seismic analysis and is, therefore, acceptable. Acceptance criteria are based on comparison of the stress results with the allowable stress limits in ASME BPV Code Section III, Appendix F, paragraph F-1331, "Criteria for Components," for components, welds, and threaded structural fasters. For stresses in welds, the quality factor is used by multiplying the allowable stress limit by the quality factor in accordance with ASME BPV Code, Table NG-3352-1, "Permissible Welded Joints and Design Factors."

In the structural modeling, the CRAGT assembly model consists of CRAGT, CRA cards, CRD shaft alignment cone, and CRA lower flange. Three ANSYS response spectrum analyses are performed considering the control rods in a variety of positions, from the rod completely inserted into the fuel to fully withdrawn. To account for added mass of water inside the CRAGT assembly, the mass density of the CRAGT is adjusted by adding the mass of water contained inside the CRAGT assembly. The NRC staff considers the approach to be acceptable because the added mass is properly considered in the FEM. The stress results indicate that the stresses of CRAGT assembly and associated welds meet the ASME BPV Code allowable stress limits under Service Level D loads.

The URA model consists of upper CRD shaft supports, upper riser transition, upper riser section, upper riser hanger ring, upper riser hanger braces, and upper riser bellows, as well as 16 CRD shafts. To account for added mass of water inside the URA, the mass density of the URA is adjusted by adding the mass of water contained inside the URA assembly. Response spectrum analysis is carried out. The CRD shaft has a radial gap of 3.18 mm (0.125 in.) between the CRD shaft and the CRD shaft grid support. In the analysis, the radial gap is not considered. The CRD shafts are coupled to the CRD shaft supports and constrained in the horizontal translational degree of freedom. To investigate the gap effects, the applicant performed additional analysis using two models (linear and nonlinear). The linear model does not consider the CRD shaft radial gap. The nonlinear model includes nonlinear contact elements representing the CRD shaft radial gap. Transient analyses are performed using the time histories calculated from the 3-D NPM model. The analysis results show that the stresses of the linear model are much higher than those of the nonlinear model due to resonance vibration occurring in the linear model. The nonlinear contact changes the major frequencies of the CRD shaft to be away from the transient load frequencies, and resonance does not occur. The NRC staff finds not considering the CRD shaft radial gap in the analysis to be acceptable because the results are conservative. The stress results of the analysis indicate that the stresses of URA assembly and associated welds meet the ASME BPV Code allowable stress limits under Service Level D loads.

The components included in the SG stress evaluation include SG tubes, SG tube support columns, upper SG supports and associated welds, and lower SG supports and associated welds. A 3-D ANSYS model of the SG assembly, the portion of the RPV, and upper riser is developed. The SG tubes are modeled by ANSYS BEAM189 element (three-node high-order beam element). The tube supports are modeled using 2-node BEAM188 elements. The model consists of 21 SG tube columns and 21x 8 SG tube support assemblies. Tube columns 1, 11, and 21 are explicitly modeled while the other tube columns are represented by two ANSYS Matrix50 super elements, one for tube columns 2–10 and the other for columns 12–20. The RPV and upper riser are also modeled by a super element. The three super elements and the three tube columns (1, 11, and 21) are combined to form the SG assembly model. The NRC staff finds that the use of a super element in the analysis is appropriate due to the large number of degrees of freedom in the SG assembly. The ANSYS super element is based on substructure technique. It is commonly used in modeling of complex structures. For fluid modeling and mass distribution, the applicant adjusted the SG tube density to account for secondary coolant inside the SG tubes and primary coolant outside the tubes. For the secondary coolant, average densities of the liquid and steam based on the thermal-hydraulic model outputs are calculated for 11 regions over the height of the SG. Primary coolant densities are also calculated for the same 11 regions. The mass of primary coolant displaced by the SG tubes is applied to the effective tube densities. The mass of primary coolant displaced by the tube supports is applied to the effective densities of the tube supports. Mass elements are used to apply the mass of fluid contained inside the upper riser and upper RPV region near the baffle plate to the appropriate surfaces. Hydrodynamic coupling between the RPV and upper riser is accomplished using fluid elements. The NRC staff finds the SG modeling and stress analysis to be acceptable for the following reasons. The fluid modeling and mass distribution in the SG model properly account for the component mass and added mass effect. Load combination includes dead weight, pressure, DBPB, and seismic. The loads of MSPB and FWPB are smaller than DBPB and, therefore, are not considered in the analysis. Acceptance criteria are based on comparison of the stress results with the allowable stress limits in ASME BPV Code Section III, Appendix F. The applicant provided the maximum membrane stress intensity, membrane plus bending stress intensity, and shear stress in the SG assembly components and associated welds. The stress results indicate that the stresses of the SG assembly and associated welds meet the ASME BPV Code allowable stress limits under Service Level D loads.

In summary, the NRC staff finds that the RVI stress analyses are properly performed. The design of the RVI component meets the structural requirements of the ASME BPV Code Section III for Service Level D loads. ITAAC 1 listed in DCA Part 2, Tier 1, Table 2.8-2, will ensure that equipment listed in DCA Part 2, Tier 1, Table 2.8-1, including RVIs, is designed to withstand design-basis seismic loads. This ITAAC is evaluated in Chapter 14.3 of this SER.

3.9.2.4.5.4 Stress Evaluation of Reactor Pressure Vessel Service Level D Faulted Condition

In the report on the 2017 audit (ADAMS Accession No. ML18023A091), the NRC staff describes how the applicant evaluated the RPV primary stress in the Service Level D faulted condition. The stress report referenced in the audit report contains analysis assumptions, methodology, results, conclusions, and software documentation. The evaluation used the following load combination:

P + DW + EXT +/- SRSS (SSE + DBPB)

P, DW, EXT, SSE, and DBPB are defined as operating pressure, deadweight, external mechanical loads, SSE, and DBPB, respectively. The external mechanical load consists of RPV and CNV support reactions, RVI and CNV interface loads, scram loads, fuel assembly weights, and nozzle loads. The DBPB load includes loads from the spurious actuation of RVVs, RSVs, and RRVs and from a CVCS pipe break. The NRC staff observed that the load combination does not include the blowdown loads of an MSPB and FWPB. The applicant has since updated the RPV primary stress analysis to include the loads at the MS and FW nozzles and plenums. The calculation demonstrates that the inclusion of the MSPB and FWPB loads does not affect the conclusions of the analysis. The NRC staff considers the blowdown loads to be acceptable because the applicant has updated the RPV primary stress analysis to include the loads of the MSPB and FWPB.

The RPV was evaluated in accordance with the stress limits specified in ASME BPV Code, Section III, Appendix F, for the Service Level D faulted condition. Two ANSYS FEMs were developed, one for the vessel top head (including a section of the PZR shell) and one for the vessel shell. The top head model is a 360-degree model that consists of ANSYS solid elements. Vertical and circumferential displacements are fixed at the bottom of the model. Using the specified nozzle loads and seismic interface loads, a linear elastic analysis of the RPV top head was carried out to evaluate stress in the Service Level D faulted condition. Thirty-eight stress classification lines were defined at cross sections near the nozzle, the ligament between nozzles, the crown, and the knuckle of the top head. Using the ANSYS postprocess, stress linearization was performed to obtain membrane and bending stress intensities in these locations. The membrane stress intensities were classified as primary general membrane (P_m) and primary local membrane (P_L), in accordance with ASME BPV Code, Section III, Table NB-3217-1, "Classification of Stress Intensity in Vessels for Some Typical Cases." The calculated stress intensities were compared with the ASME BPV Code allowable stress limits of P_m, P_L, and $P_L + P_b$ (P_b is "primary bending"). The analysis results indicate that the highest stress intensities occurred in the ligaments between the inner CRDM nozzles. The stresses were within the ASME BPV Code's allowable stress limits.

In addition, the vessel model is a 360-degree solid element model that consists of the RPV below the top head. The model is constrained at the RPV upper support with vertical and tangential displacements fixed. Using the specified nozzle loads and interface loads, a linear elastic analysis was carried out. Forty-one stress classification lines were defined at cross sections near nozzles, ligaments between nozzles, the bottom head crown, the bottom head knuckle, and geometry discontinuity junctions. The analysis results indicate that the highest stress intensities occurred at RCS injection and discharge nozzles. The stresses are within the ASME BPV Code's allowable stress limits. In summary, the NRC staff finds the stress analysis of Service Level D to be acceptable because it demonstrates that the design of the RPV meets the structural requirements of the ASME BPV Code for Service Level D loads.

3.9.2.4.6 Correlations of Reactor Vessel Internal Vibration Tests with the Analytical Results

Some benchmarking test data have been compared to analytic results. In particular, the forced response of the HCSG TF-2 was predicted using the same bounding models used by the applicant for its design analyses. The resulting predicted response generally exceeds, sometimes significantly, TF-2 measurements, providing confidence in the conservatism of the applicant's design methods. Based on this, the NRC staff finds the TF-2 benchmarking to be acceptable.

3.9.2.5 Combined License Information Items

DCA Part 2, Tier 2, Table 1.8-2, lists COL information item numbers and descriptions related to dynamic testing and analysis of SSCs from DCA Part 2, Tier 2, Section 3.9.2.

Table 3.9.2.-1: NuScale COL Items for DCA Part 2, Tier 2, Section 3.9.2

COL Item No.	Description	DCA Part 2, Tier 2, Section
COL 3.9-1	A COL applicant that references the NuScale Power Plant design certification will provide the applicable test procedures prior to the start of testing and will submit the test and inspection results from the comprehensive vibration assessment program for the NuScale Power Module, in accordance with Regulatory Guide 1.20.	3.9
COL 3.9-10	A COL applicant that references the NuScale Power Plant design certification will verify that evaluations are performed during the detailed design of the MS lines utilizing acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology contained in "NuScale Comprehensive Vibration Assessment Program Technical Report," TR-0716-50439 is acceptable for this purpose. The COL applicant will update Section 3.9.2.1.1.3 to describe the results of this evaluation.	3.9
COL 3.9-12	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. 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 CSDRS, the standard design of NPM components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.	3.9
COL 3.9-13	A COL applicant that references the NuScale Power Plant design certification will complete an assessment of piping systems inside the reactor building to determine the portions of piping to be tested for vibration and thermal expansion. The piping systems within the scope of this testing include ASME BPVC, 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 COL applicant may select the portions of piping in the NuScale 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

3.9.2.6 Conclusion

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 for service conditions without DWO instability behavior. Important FIV mechanisms have sufficient margin to provide reasonable assurance against their occurrence during normal and faulted NPM operation. TF-3 testing will confirm the safety of the HCSG for FIV. The adequacy of the SGIFR final design to operate without LFI for forward secondary coolant flow conditions will be confirmed by validation testing prior to the initial plant startup. NPM initial startup testing will capture any unexpectedly high FIV, including ARs in the CNTS steam piping. Acceptance criteria are established to accept only benign TB loading. 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, provide further confidence in the safety of the RVIs and HCSG. 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 provided no significant DWO instability occurs. 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 comprehensive vibration analysis program 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 cannot make a safety finding regarding the long-term structural integrity of the HCSG and IFRs due to possible repeated oscillatory and cavitation loads caused by secondary coolant flow DWO instabilities. NuScale has not submitted sufficient information to rule out the possibility of DWO instabilities near full-power operating conditions. This aspect of the structural integrity of the HCSG tubes and SGIFRs due to DWO instability has not been reviewed by the NRC staff and, therefore, has not been resolved. This specific aspect of the structural integrity of the HCSG tubes and IFRs due to DWO instabilities will need to be addressed by a COL applicant referencing the NuScale design as described in Section 3.9.2.4.3.11.

The NRC staff concludes that the 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 LOCA in combination with the SSE, 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 2, addresses component design and provides the NPM SSC design descriptions and design commitments. In DCA Part 2, Tier 1, Tables 2.1-1 and 2.1-2 describe the NPM ASME BPV Code Class 1, 2, and 3 piping systems and mechanical components.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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: DCA Part 2, Tier 1, Table 2.1-4, contains ITAAC for the required ASME BPV Code Class 1, 2, and 3 as-built piping system and component design reports. Section 14.3 of this SER discusses the NuScale ITAAC.

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

Technical Reports: The staff reviewed the following NuScale TRs, which are incorporated by reference in accordance with DCA Part 2, Tier 2, Section 1.6 and Table 1.6-2, and used information in these reports to make the safety findings:

- TR-0916-51502-P, Revision 2, "NuScale Power Module Seismic Analysis," issued April 2019
- TR-0716-50439-P, Revision 2, "NuScale Comprehensive Vibration Assessment Program Technical Report," issued July 2019

- TR-1016-51669, Revision 1, "NuScale Power Module Short-Term Transient Analysis," issued July 2019

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 1.26, Revision 4

- 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 assure 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, provide load combinations and supporting information associated with the ASME BPV Code, Section III, Class 1, 2, and 3 components:

- Table 3.9-2, “Pressure, Mechanical, and Thermal Loads,” defines the pressure, mechanical, and thermal loads.
- Table 3.9-3, “Required Load Combinations for Reactor Pressure Vessel American Society of Mechanical Engineers Stress Analysis,” gives the required load combinations and stress limits for the RPV.
- Table 3.8.2-2, “Load Combinations for Containment Vessel and Support ASME Code Stress Analysis,” and Table 3.8.2-3, “Load Combinations for Containment Vessel Bolt ASME Code Stress Analysis,” relate to the CNV.
- Table 3.9-5, “Required Load Combinations for Reactor Vessel Internals American Society of Mechanical Engineers Stress Analysis,” addresses the RVIs.
- Table 3.9-6, “Required Load Combinations for Control Rod Drive Mechanism American Society of Mechanical Engineers Stress Analysis,” covers the CRDMs.
- Table 3.9-7, “Load Combinations for Decay Heat Removal System Condenser,” relates to the DHRS condenser.
- Table 3.9-8, “Load Combinations for NuScale Power Module Top Support Structure,” is for the NPM TSS.
- Table 3.9-9, “Loading Combinations for Decay Heat Removal System Actuation Valves,” addresses the DHRS actuation valves.
- Table 3.9-10, “Loads and Load Combinations for Reactor Safety Valves,” is for the RSVs.

- Tables 3.9-11, “Load Combinations for Emergency Core Cooling System Valves,” through 3.9-14, “Loads and 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.

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.9.1. DCA Part 2, Tier 2, 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 (i.e., normal, upset, emergency, and faulted conditions, respectively); and test conditions. DCA Part 2, Tier 2, Section 3.9.1, provides the design transients and number of events and occurrences for fatigue analyses. ITAAC 1 in DCA Part 2, Tier 1, Table 2.1-4, 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. DCA Part 2, Tier 2, Section 3.12, describes and discusses the loads used for piping analysis for thermal stratification, cycling, and striping (including NRC Bulletin (BL)-79-13, Revision 2, “Cracking in Feedwater System Piping,” dated October 16, 1979; BL-88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” dated June 22, 1988; and BL-88-11, “Pressurizer Surge Line Thermal Stratification,” dated December 20, 1988). In SER Section 3.6, the staff evaluates pipe whip and pipe impingement loads from pipe breaks.

The staff reviewed DCA Part 2, Tier 2, 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, hydrogen detonation, lifting, handling, and transportation.

DCA Part 2, Tier 2, Section 3.9.3.1.1, addresses FWPB and MSPB, which are high-energy pipe breaks to be considered for analyses outside the CNV but not inside the CNV, where LBB is applied. LBB is reviewed and documented in Section 3.6.3 of this report.

In a letter dated October 23, 2017 (ADAMS Accession No. ML17296B367), the applicant stated that 10 CFR Part 50 Appendix J.III.A, “Type A test” that the integrated Leak Test, is not applicable to the NPM, in accordance with the justification provided in TR-1116-51962, Revision 1, “NuScale Containment Leakage Integrity Assurance Technical Report,” issued May 28, 2019 (ADAMS Accession No. ML19149A298). In DCA Part 7, the applicant has proposed an exemption from the integrated leak rate testing design requirements of GDC 52, “Capability for Containment Leakage Rate Testing,” and has requested that the DC rule for the NuScale design include an exemption from the requirements of 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors,” Type A testing. The staff’s evaluation of this exemption request is contained in Chapter 6 of this SER. The staff finds that the applicant’s applicable loads and load combinations are acceptable in accordance with SRP Section 3.9.3, Acceptance Criterion II.1, for the design and service loadings applicable to the design of ASME BPV Code Class 1, 2, and 3 components and supports.

For seismic loads, the RXB system model, with representation of the NPM subsystem, is analyzed for SSI in the frequency domain using the SASSI2010 computer code. As discussed in DCA Part 2, Tier 2, Section 3.7.5, SASSI2010 is used to obtain seismic design loads and in-structure floor response spectra for the seismic Category I buildings accounting for the effects of SSI. Results from the RXB seismic system analysis include in-structure time histories at each NPM support location and the pool walls and floor surrounding the NPM. With these interface location in-structure time histories, the detailed dynamic analysis of the NPM subsystem is performed using ANSYS. The NPM dynamic analysis provides in-structure time histories and ISRS for qualification of equipment supported on the NPM and time histories at core support locations for seismic qualification of fuel assemblies. TR-0916-51502-P, Revision 2, provides the results of the seismic analysis of the NPM. The staff reviewed the TR with respect to the seismic loads, stresses, and assumed damping values.

In a letter dated August 2, 2018 (ADAMS Accession No. ML18214A835), the applicant clarified that the lowest specified damping value for the SSE event is 4 percent for welded steel or bolted steel with friction connections. The composite damping value assigned to the NPM subsystem is 4-percent damping, while RG 1.61, Revision 1, employs 3 percent for SSE for the RPV. The applicant provided additional information in a letter dated October 16, 2018 (ADAMS Accession No. ML18289B091), stating that all composite dampings of the 3-D NPM model are higher than 4 percent, which is considered conservative, to be used consistently for NuScale seismic SSE analyses. The staff recognizes that it found 4-percent SSE damping acceptable for AP1000 plants, as documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004, and its supplements. Also, 4-percent SSE damping for the NPM is consistent with the NRC's position not to be greater than the damping values specified in Table 1 of RG 1.61. Therefore, the staff concludes that the use of 4-percent damping is conservative and acceptable for the NPM SSE seismic design.

In a letter dated October 16, 2018 (ADAMS Accession No. ML18289B091), the applicant indicated that the seismic analysis neglects fluid damping because fluid damping is less significant than the structural damping (see TR-0916-51502-P, Section 6.6.5.3). The reactor pool water has been assigned zero viscosity in the ANSYS models and correspondingly exhibits no damping. On the basis that the NPM seismic model used a composite damping of 4 percent for all the elements without separating each element, the staff finds this acceptable.

TR-0916-51502-P, Section 8.4.2.6, discusses seismic uplift displacements between the NPM skirt and reactor pool floor and between the lower core plate and reflector blocks. TR-0916-51502-P, Table 8-9, gives the results of the maximum uplift displacement. On the basis that the maximum uplift displacement is too small, the staff determined that the structural integrity of the NPM is acceptable.

The SSE event is analyzed for the lower RPV in the RFT during an outage, when the fuel is exposed to the refueling pool environment for up to 70 hours. TR-0916-51502-P provides the seismic analysis of the lower RPV in the RFT. SER Section 3.9.2 evaluates the seismic analysis in TR-0916-51502-P. SER Section 3.8.4 documents the seismic evaluation of the RFT.

TR-0916-51502-P identifies that the contact between the CNV skirt and pool floor is generated using a rigid surface below the NPM in the operating bay. The rigid surface is defined as a square of 15 m (50 ft), coincident with and centered on, the base of the CNV skirt. Nonlinear contact is established through target elements on the floor and contact elements on the bottom surface of the CNV skirt support ring. TR-0916-51502-P also identifies that the lateral seismic accelerations are applied to a remote point that is centered on and scoped to, the bottom of the

CNV skirt support ring. This allows sliding of the contact with the rigid floor surface during SSE, as permitted by the actual configuration (the CNV skirt engages with the passive support ring on the floor). The vertical seismic acceleration is applied to a separate remote point to allow for seismic uplift displacement of the CNV skirt from the rigid floor. Acceleration boundary condition data generation is described in TR-0916-51502-P, Section 5.3. On the basis that the description of the boundary conditions of the NPM seismic model does reflect the actual design boundary conditions, the staff finds this acceptable.

3.9.3.4.2 Design and Installation of Pressure-Relief Devices

The RCS 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 overpressurization 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 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.

DCA Part 2, Tier 2, Section 3.10.2, states that the guidance and requirements of RG 1.100, Revision 3, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” issued September 2009, and IEEE 344-2004 are the source of the methods and procedures used for seismic and dynamic qualification of mechanical and electrical equipment and ASME Standard QME-1-2007, “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 (ADAMS Accession No. 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.

In DCA Part 2, Tier 2, Tables 3.9-10 and 3.9-11 describe the loads considered for fatigue evaluation for the RSVs and ECCS valves, respectively. DCA Part 2, Tier 2, Section 3.9.1.1, discusses the individual transients and the number of cycles included in the design basis. As described in DCA Part 2, Tier 2, 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.

The percentage of critical damping for mechanical components, including pressure boundary valve bodies, is 3 percent for the SSE and 2 percent for the OBE, in compliance with DCA Part 2, Tier 2, Table 3.7.1-6. The staff finds this consistent with RG 1.61 and therefore acceptable.

The SG thermal relief valves are installed in the FW piping and provide overpressure protection during water-solid conditions that may occur during NPM shutdown.

3.9.3.4.2.1 Steam Generator Operability Assurance

The NuScale Power Plant does not rely on pumps to perform any safety-related functions. DCA Part 2, Tier 2, Section 3.9.6, lists the active, safety-related valves.

Active valves are subject to factory tests to demonstrate operability before installation, followed by postinstallation testing in the plant. DCA Part 2, Tier 2, Section 3.9.6, describes the tests performed as part of the preservice testing (PST) and inservice testing (IST) programs. The PST and IST requirements are contained in the ASME OM Code, Division 1, Section IST.

DCA Part 2, Tier 2, Section 3.9.6, describes the functional and operability design and qualification provisions and IST programs for safety-related valves. DCA Part 2, Tier 2, Section 3.11, discusses EQ of safety-related valves. The seismic qualification of safety-related valves is performed in accordance with ASME Standard QME-1-2007, as endorsed by RG 1.100, Revision 3, and discussed in DCA Part 2, Tier 2, Section 3.10.

Each NPM consists of two independent helical coil steam generators (HCSGs). They are physically integral within the upper power module component. Each HCSG contains numerous tubes, such that a large heat transfer surface area can fit into a small height and volume in the upper RPV shell section running between the FW and steam plenums. The staff reviewed how the HCSGs meet the operating ability requirements in NUREG-1367, "Functional Capability of Piping Systems," issued November 1992.

In a letter dated October 23, 2017 (ADAMS Accession No. ML17296B367), the applicant stated that SG tubing meets the criteria provided for design by analysis according to ASME BPV Code, Section III, Subsection NB-3200, and is not required to be designed using ASME BPV Code equations for piping from Subsection NB-3600. NUREG-1367 provides criteria that must be met for piping if the Level D ASME BPV Code equations from Subsection NB-3600 may not be sufficient to ensure that functionality is not impaired (e.g., because of large displacement). Although NUREG-1367 does not specifically address SG tubing designed under ASME BPV Code, Section III, Subsection NB-3200, the helical coil tubing is expected to satisfy the functional operability conditions in NUREG-1367 as follows:

- The HCSG is designed for reversing dynamic loads as required by the design specification. Loads considered include those from seismic and valve actuation.

- For seismic loading, moments are calculated using elastic response spectrum analysis as required by the design specification. ISRS used in the analysis are broadened by

15 percent, as stated in DCA Part 2, Section 3.7.2.5.1, and TR-0916-51502-P, Revision 2, Section 8.4.2.5. Damping shall not exceed 5 percent.

- Stresses in the HCSG tubing from dead weight are less than 0.25 Sy (Sy is yield strength of material).

- For the HCSG tubing, Do/t = 0.625/0.05 = 12.5 does not exceed 50 (Do is pipe outside diameter, and t is wall thickness).

HCSG tubing is subject to external pressure, thus not meeting the fifth condition stated in NUREG-1367. As required by the design specification, tubes shall be designed for external pressure in accordance with the rules for determining allowable external pressure in ASME BPV Code Case N-759-2, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Class 1, 2, and 3 Section III, Division 1 Supp 11," as endorsed by RG 1.84, Revision 36, "Design, Fabrication, and Materials Code Case Acceptability," ASME Section III, issued August 2014.

The staff finds that the design meets the conditions in Section 9 of NUREG-1367 as described above with the exception of the external pressure, which is designed in compliance with NRC-accepted ASME BPV Code Case N-759-2 and is therefore acceptable.

3.9.3.4.2.2 ASME BPV Code Design for Stress/Fatigue Analysis

Using the guidance in SRP Section 3.9.3, Appendix A.7.A.(iv), the staff conducted an audit to review the ASME BPV Code-required design documents, such as design specifications, design reports, load capacity data sheets, or other related material or portions thereof, in order to establish that the design criteria, the analytical methods, and functional capability satisfy the guidance in the appendix. As part of the audit, the staff reviewed the RPV design documents. The Phase 2 regulatory audit (ADAMS Accession No. ML19018A140) is complete. The staff noted a lack of stresses and fatigue evaluations for internals and critical representative RPV components. The staff performed the Phase 4 regulatory audit on the design for fatigue and preliminary stress analysis to address the stress/fatigue analysis. During the Phase 4 audit, the staff reviewed the method used by the applicant in determining the environmentally assisted fatigue (EAF) correction factor. The applicant used the average temperature of the transient to evaluate the EAF correction factor (Fen). The staff noted that the average temperature that should be used in the calculations should produce results that are consistent with the results that would be obtained using the modified rate approach described in Section 4.2.14 of NUREG/CR-6909, Revision 0, issued February 2007. The staff noted that NuScale's determination of average temperature of the transient did not yield comparable results to the detailed integrated approach as noted in Revision 0 of NUREG/CR-6909. The applicant provided a comparison, for one specific location, of the Fen using the average temperature to the Fen using the average temperature methods of NUREG/CR-6909, Revision 1, issued May 2018, and reducing conservatism by applying the poison ratio to only the thermal stress. This comparison showed that the added conservatism included in the initial calculation compensated for the nonconservatism in the selection of the average temperature, for this specific location. NuScale also stated that the average temperature method without considering Fen threshold temperature is applied to all fatigue analyses. All the fatigue analyses using the correct method will be addressed in the final design. The staff found the calculations in various stages of completion (some calculations were final with assumptions that needed to be verified, and some were not final calculations), but the staff reviewed sufficient information to determine that the applicant could reasonably complete the calculations in accordance with the ASME

BPV Code, Section III. Additionally, any deficiencies were either identified in the corrective action program or were planned to be addressed when the calculations were to be made final documents. For detailed information, see the Phase 4 audit report, dated January 13, 2020 (ADAMS Accession No. ML19340A971).

Based on the review and audits, including the issues documented in the applicant's corrective action program, the staff concludes that the applicant will use the correct methodology to meet ASME BPV Code, Section III, design requirements at the time of final design completion. Additionally, there are ITAAC (e.g., DCA Part 2, Tier 1, Table 2.1-4, item 1) that will confirm that the as-built final design analyses meet the regulations and ASME BPV Code requirements. Therefore, the staff considers the applicant's approach to completing ASME BPV Code design documentation acceptable.

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 DCA Part 2, Tier 2, 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. In DCA Part 2, Tier 2, Section 3.9.3.1.1 and Table 3.9-2 define applicable loads. Dynamic loads are combined using 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.

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Table 3.9-5, specifies 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 linear-type and plate-and-shell-type supports, respectively. The ASME BPV Code Class 1 supports consider high-cycle 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 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 DCA Part 2, Tier 2, Table 3.9-8, load combination and allowable stress limits. SER Section 3.12 evaluates ASME BPV Code Class 1, 2, and 3 piping supports.

DCA Part 2, Tier 2, 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 supports and therefore are also designed to ASME BPV Code, Section III, Subsection NG.

The staff reviewed how the applicant addressed the effects of the friction and wearing between the tubes and tab supports when the tubes are subject to the fluid-induced vibration from the vertical cross flow for the 60-year plant life. TR-0716-50439 discusses the FIV mechanisms that are evaluated for the SG tubes. SER Section 3.9.2 evaluates TR-0716-50439, including the SG tubes and the effects of the friction and wearing between the tubes and tab supports.

As discussed in a letter dated October 23, 2017 (ADAMS Accession No. ML17296B366), a fretting wear assessment is performed using the analytically determined RMS response for the wear couple of the SG tube and tube support. Fretting wear represents a combination of impact and sliding wear and occurs where there are small amplitude and impact force vibrations, which are characteristic of TB. The COF between the tube and the tube support is used to calculate the fretting contact force. The maximum wear depth is calculated over the 60-year design life of the plant using an experimental formulation. In a letter dated March 8, 2018 (ADAMS Accession No. ML18067A520), the applicant stated that a zero-to-peak RMS vibration for sinusoidal whirling of the tube is used.[1] This approximation is equal to the square root of two times the calculated RMS vibration value.

The applicant also stated that although design analysis is performed to estimate wear, these estimates are predictive in nature, and actual wear performance is monitored over the life of the SG design in accordance with Nuclear Energy Institute (NEI) 97-06, Revision 3, “Steam Generator Program Guidelines,” issued January 2011 (ADAMS Accession No. ML111310708). GTS Section 5.5.4, “Steam Generator (SG) Program,” provides the NuScale SG inspection requirements. The applicant provided additional information related to periodic SG tube inspections in DCA Part 2, Tier 2, Section 5.4.1.4. On the basis that the SG tube design uses ASME guidance and tube wear inspection in accordance with NEI 97-06, Revision 3, the staff finds this acceptable.

DCA Part 2, Tier 2, Section 3.9.3.4, states that DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.

3.9.3.5 Combined License Information Items

SER Table 3.9.3-1 lists COL information item numbers and descriptions related to ASME BPV Code Class 1, 2, and 3 components, component supports, and core support structures, from DCA Part 2, Tier 2, Table 1.8-2.

Table 3.9.3-1: NuScale COL Information Item for Section 3.9.3

Item No.	**Description**	**DCA Part 2,Tier 2, Section**

[1] Au-Yang, M.K., “Flow-Induced Vibration of Power and Process Plant Components,” ASME Press, New York, NY, 2001, pages 343 and 361.

COL Item 3.9-2	A COL applicant that references the NuScale Power Plant design certification will develop design specifications and design reports in accordance with the requirements outlined under American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III (Reference 3.9-1). A COL applicant will address any known issues through the reactor vessel internals reliability programs (i.e. Comprehensive Vibration Assessment Program, steam generator programs, etc.) in regards to known aging degradation mechanisms such as those addressed in Section 4.5.2.1.	3.9

3.9.3.6 Conclusion

Based on the above discussion, the staff concludes that the NuScale DCA 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 rods and the related mechanical components that provide the means for mechanical movement. GDC 26, “Reactivity Control System Redundancy and Capability,” and PDC 27, “Combined Reactivity Control Systems Capability,” require that the CRDS provide one of the independent reactivity control systems. The 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 rods shall always be 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 reactivity control elements in the RPV. The review in this section is limited to the CRDM portion of the CRDS.

3.9.4.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, discusses the NPM, which contains the CRDS. DCA Part 2, Tier 1, Section 2.1.1, contains the design description for the CRDS, including the CRDMs.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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 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 the latch mechanism and CRD shaft are safety-related, risk-significant components that ensure positive CRA insertion.

Controlled movement of the CRAs 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 CRA to be inserted by gravity.

Rod position indication is provided by coils located in the sensor coil assembly, which is supported by the rod travel housing.

ITAAC: Section 14.3 of this SER 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 3, “Control Rod Drive Systems,” dated March 2007, 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 Principal Design Criterion 27

GDC 27, "Combined Reactivity Control Systems Capability," requires the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained. The applicant proposed an exemption from GDC 27 and proposed Principal Design Criterion 27, which states the following:

> The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.
>
> Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted.

This exemption request does not address the functionality of the CRDS. The staff evaluates this exemption request in SER Section 15.0.6.

3.9.4.4.2 Descriptive Information

The staff reviewed DCA Part 2, Tier 2, Section 3.9.4, in accordance with SRP Section 3.9.4. DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 4.6, provides the majority of the figures depicting the CRDS.

DCA Part 2, Tier 2, Figure 4.6-1, "Overview of Control Rod Drive Mechanism Locations in Relation to the Reactor Pressure Vessel and Containment Vessel," shows CRDM support structures, and DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.9.4.1, also discusses the CRDM support structures.

DCA Part 2, Tier 2, Section 4.6.2, discusses a failure modes and effects analysis (FMEA) that evaluated failures of the CRDM. The staff reviewed this analysis during an audit of the design and testing programs for the CRDS (ADAMS Accession No. ML17331A357). The staff's review determined that the CRDS is capable of performing its safety-related function following the loss of any active component, as documented in the audit report and SER Section 4.6.

The staff reviewed the description of the method of operations provided in DCA Part 2 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 DCA Part 2, Tier 2, Sections 1.5.1.6 and 3.9.4.4 state that there are two unique features: a remote disconnect mechanism and a longer CRD shaft. The remote disconnect coil is one of the four main coils in the drive coil assembly and is used to remotely connect and disconnect the drive shaft from the CRA, as described in DCA Part 2, Tier 2, Section 3.9.4.1.1. The remote disconnect coil is always deenergized during normal operations and remains in this state during a reactor trip. Therefore, the staff finds that this unique feature will not impact the system’s safety-related function of CRA insertion during a reactor trip

3.9.4.4.3 Codes and Standards

DCA Part 2, Tier 2, Section 3.9.4.2, “Applicable Control Rod Drive System Design Specifications,” describes the classification of the CRDM components, also provided in DCA Part 2, Tier 2, Table 3.2-1, 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.

DCA Part 2, Tier 2, Section 3.9.4, indicates that construction will be in accordance with the same codes used for the design of the components. There is additional discussion of the design, fabrication, inspection, and testing of nonpressure-retaining components; specifically, they do not typically come under the jurisdiction of the ASME BPV Code. Material specification mechanical property requirements are the basis for materials without established stress limits. The CRD shaft was specifically mentioned as a major nonpressure-retaining component, and it will be considered an ASME BPV Code, Subsection NG, component (internal structure). The specific requirements for the CRD shafts are provided in DCA Part 2, Tier 2, Section 3.9.4.1.1. The staff finds these requirements acceptable for the CRD shafts and other nonpressure-retaining components because they satisfy the requirements of GDC 1.

3.9.4.4.4 Load Combinations and Stress Limits

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Table 3.9-6, 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. 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.

3.9.4.4.5 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.” DCA Part 2, Tier 2, Section 3.9.4.4, “Control Rod Drive System Operability Assurance Program,” briefly discusses a prototype testing program for the CRDS, which includes performance testing, stability testing, endurance testing, and production testing. DCA Part 2, Tier 2, Section 4.2.4.2.3, discusses control rod testing. In DCA Part 2, Tier 2, Sections 1.5.1.6 and 1.5.1.7 contain additional discussion of specific testing.

To gain a better understanding of the design and testing methods for the CRDS, the staff initiated an audit of the CRDS testing programs (see ADAMS Accession No. ML17158B428).

The staff reviewed documentation for testing that had already occurred as part of design development and reviewed plans for testing that had not been completed at the time of the audit. Further discussion of this initial audit may be found in the audit summary report (ADAMS Accession No. ML17331A357).

The staff conducted a followup audit of CRDS testing results from September 4, 2018, through October 25, 2018, in accordance with the followup audit plan (ADAMS Accession No. ML18235A509).

During the followup audit, the staff reviewed documentation and results from the CRA drop and CRD shaft alignment testing, which were not available at the time of the initial audit. The staff's review, documented in the followup audit summary report (ADAMS Accession No. ML18325A153), independently compared the testing configuration parameters to the design configuration and found the results to be acceptable. The staff verified that the testing results bounded the performance assumed in the safety analysis. DCA Part 2, Tier 2, Section 1.5.1.7, contains a description of the testing program and the associated results.

A COL item exists for a COL applicant to create an operability assurance program for the CRDS. DCA Part 2, Tier 2, 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 DCA and provide a summary of the 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.

3.9.4.4.6 Boric Acid Accumulation

During the Advisory Committee on Reactor Safeguards subcommittee meeting on April 17, 2019 (ADAMS Accession No. 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 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. In light of the potential high safety significance of this effect, the staff issued RAI 9691, Question 03.09.04-13, which requested assurance from the applicant that the 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. In its response (ADAMS Accession No. ML19200A208), the applicant provided a description of the CRDM configuration. The CRDMs are located above the top of the PZR in the gas phase, and the housings are cooled by heat exchangers connected to the reactor component cooling water (RCCW) system to ensure proper function of the CRDM electromagnetic coils. The applicant stated that the NuScale RCS utilizes dissolved hydrogen in the reactor coolant during power operations in accordance with the EPRI water chemistry guidelines, which require a liquid phase concentration of 25 cubic centimeters per kilogram. The applicant stated that this noncondensable hydrogen gas will accumulate in the CRDM volume. The applicant's response also states that there is no mechanism for volatilized boron contained in saturated steam to be deposited on a cooled metal surface. The applicant believes that once saturated steam condenses on a cooled surface, any volatilized boric acid would dissolve in the liquid condensate, as the concentration of boron would be lower than the solubility limit. It is the applicant's position that this phenomenon is not expected to occur and that the safety function of inserting control rods would not be impeded by boric acid deposition. The staff finds that the applicant has considered this phenomenon in the design process for the CRDMs, as evidenced by its RAI response, and the applicant does not consider this phenomenon to adversely impact the ability of the CRDMs to drop the control rods.

3.9.4.4.7 Control Rod Drive Housing Integrity

NuScale DCA Part 2, Tier 2, Section 3.5.1.2, states that a CRDM housing failure, sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core, is noncredible. Section 3.5.1.2 also notes that the CRDM housing is a Class 1 appurtenance per ASME BPV Code, Section III. In its letter dated July 15, 2019 (ADAMS Accession No. ML19196A368), in response to NRC staff questions on the potential for a control rod ejection accident, NuScale indicated that the CRDM nozzles are an integral part of the reactor pressure vessel (RPV) closure head forging. NuScale specifies that the CRDM nozzle to Alloy 690 safe-end welds are full penetration butt welds, using Alloy 52/152 weld filler materials for corrosion resistance. As a result, NuScale states that the connection between the CRDM nozzles and the RPV closure head will be structurally robust.

Similar to DC reviews for other reactor designs, the NRC staff evaluated the specific aspects of the NuScale CRDM housing to evaluate the potential for a control rod to be ejected from the reactor core. Based on its review, the NRC staff does not consider a gross failure of a NuScale CRDM housing sufficient to create a missile from a piece of the housing, or to allow a control rod to be ejected rapidly from the core, to be credible. For example, control rod missile generation would be prohibited by a combination of (1) the design of the CRDM nozzles as an integral part of the RPV head, (2) the likelihood that a postulated failure of a CRDM nozzle would be preceded by leakage flow in a lateral orientation such that a catastrophic failure of the nozzle would not occur, (3) the length of each control rod, and (4) the ASME BPV Code Class 1 design requirements. CRDM housings are ASME BPV Code Class 1 components, which are subject to stringent requirements for material characteristics, inspections, and quality control during fabrication, erection, and operation. These Class 1 components are also subject to preservice and inservice inspections upon installation in the nuclear power plant, per Section XI of the ASME BPV Code, as discussed in NuScale DCA Part 2, Tier 2, Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing." Therefore, the NRC staff finds that a CRDM housing failure, sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core, is noncredible for the NuScale nuclear power plant. Section 15.4.8 of this SER documents the NRC staff review of NuScale DCA Part 2, Tier 2, Section 15.4.8, "Spectrum of Rod Ejection Accidents," which presents the safety analyses for a postulated control rod ejection accident, even though this event is not expected to occur during the life of the plant. In summary, the potential for a control rod ejection accident at a NuScale nuclear power plant is a nonmechanistic assumption for the purpose of evaluating reactivity with respect to 10 CFR Part 50, Appendix A, GDC 28, “Reactivity Limits.” Sections 15.4.8 and 15.0.6.4.3 of this SER provide additional discussion of the NRC staff review.

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 DCA Part 2, Tier 2, Table 1.8-2.

Table 3.9.4-1: NuScale COL Information Item for Section 3.9.4

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.9-11	A COL applicant that references the NuScale Power Plant design certification 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, nonpressure 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 PDC 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 DCA Part 2 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 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, deflections, 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)

3.9.5.2 Summary of Application

DCA Part 2, Tier 1: Tier 1 information associated with this section is provided in DCA Part 2, Tier 1, Section 2.1.

DCA Part 2, Tier 2: Tier 2 information associated with this section is provided in DCA Part 2, Tier 2, Section 3.9.5.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.5, states that the design and construction of both the core support structures and the internal structures comply with the ASME BPV Code, Section III, Division 1, Subsection NG.

ITAAC: The ITAAC associated with DCA Part 2, Tier 2, Section 3.9.5, are given in DCA Part 2, Tier 1, Section 2.1.2, "Inspection, Tests, Analyses, and Acceptance Criteria." 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 following TRs, which are incorporated by reference in accordance with DCA Part 2, Tier 2, Section 1.6 and Table 1.6-2, apply to this area of review, and the staff used information in these reports to make the safety findings:

- TR-0716-50439-P, Revision 2, "NuScale Comprehensive Vibration Assessment Program Technical Report," issued July 2019

- TR-0916-51502-P, Revision 2, "NuScale Power Module Seismic Analysis," issued April 2019

In response to RAIs, the applicant provided additional information in a letter dated October 10, 2017 (ADAMS Accession No. ML17284A092). This additional information did not result in changes to the DCA. This information supplements the information in the DCA for this section of the safety evaluation and is the source of information used in the technical evaluation section.

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

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.9-5. Section 3.9.3 of this safety evaluation addresses the method of combining loads for ASME Service Levels A, B, C, and D and test conditions.

The staff reviewed DCA Part 2, Tier 2, Table 3.9-5, 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.

In a letter dated October 10, 2017 (ADAMS Accession No. ML17284A092), the applicant stated 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:

> [M]aximum 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.

DCA Part 2, Tier 2, 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 designed using ASME BPV Code, Section III, Subsection NG, as a guide.

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

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.5, states that the RVI assembly comprises these subassemblies:

- core support assembly (CSA)
- lower riser assembly
- URA
- flow diverter
- PZR spray nozzles

In DCA Part 2, Tier 2, 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 upper riser is bolted to the bottom of the PZR baffle plate and is horizontally restrained by the SG tube supports located in the annulus between the upper riser and the RPV wall. A small leg of piping runs from the CVCS injection nozzle in the RPV and into the upper riser to return CVCS flow to the RCS. The lower riser assembly sits on top of the CSA and is secured by socket head cap screws and alignment dowels. The upper support blocks that are welded to the core barrel restrain the upper and lower riser assemblies in the horizontal direction. The CSA is mounted to the bottom head of the RPV, which provides the primary support to the reactor core.

In a letter dated October 10, 2017 (ADAMS Accession No. ML17284A092), the applicant provided a complete list of RVI components that are both core support structures and internal structures. Upon review of this list, the staff finds that the RVI components are appropriately classified as either core support structures or internal structures based on the location and primary function of each RVI component.

TR-0716-50439-P, Revision 2, provides further detail of the RVI assembly and description of the SG, which wraps around the URA, and PZR, which is located above the URA. 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 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 blocks 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 (NG to NB) between the SG tube supports and the RCPB portion of the RPV is at the weld between the upper tube support bars and RPV shell or integral steam plenum and 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. Section 5.4.2 of this safety evaluation provides more detail for the SG tube supports.

The SG flow restrictors are not pressure boundary components and thus not classified as part of the RVI. DCA Part 2, Tier 2, Chapter 5, Figure 5.4-8, “Steam Generator Flow Restrictor Assembly,” shows that the SG flow restrictors are located inside the feed plenum tubesheets, which form part of the RCPB and are designed to ASME BPV Code, Section III, Subsection NB. DCA Part 2, Tier 2, 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 Section 5.4.2 of this safety evaluation.

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 blocks appropriate because they are structural attachments to the RPV.

3.9.5.4.1.2 Design—Core Support Structure (Upper Riser Assembly)

DCA Part 2, Tier 2, 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 (e.g., below the bottom of the PZR).

The URA includes a hanger ring with welded braces connecting to the upper riser section. The braces are welded to the upper riser section. The attachment of the hanger ring to the bottom of the PZR baffle plate is by threaded fasteners. The upper riser hanger threaded fasteners are 304 stainless steel and are the same configuration as standard socket head cap screws. The components in the URA are classified as internal structures designed to ASME BPV Code, Section III, Subsection NG.

The slip joint is the connection (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 designed to ASME BPV Code, Section III, Subsection NG.

Five CRD shaft supports are welded to the inside of the upper riser shell. The five supports are generically referred to as CRD shaft supports; however, each provides support to both the CRD shafts and ICIGTs. These supports are part of the URA and are classified as internal structures designed to ASME BPV Code, Section III, Subsection NG.

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.

DCA Part 2, Tier 2, Section 3.9.5.1, states that there is a bellows assembly in the lower portion of the upper riser to accommodate vertical thermal expansion.

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 is designed to follow ASME BPV Code, Section III, Subsection NG, as a guide, 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 eight SG lower tube support cantilever beams that are part of the SG tube support structure. A CRD shaft alignment drop test was conducted to determine the displacement limits for the CRD shaft supports. The result of this audit is documented in ADAMS Accession No. ML18325A153.

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 detail about the SG tube support structure is documented in Section 5.4.2 of this safety evaluation. In addition, Section 3.9.2 of this safety evaluation provides more details for the design of the bellows. Specifically, the overlap of the upper and lower lateral restraints of the bellows prevents relative lateral displacement between the upper and lower sections of the bellows assembly. This lateral restraint is necessary in order to provide structural support for the CRD shafts and the ICIGTs. The lateral restraints are classified as seismic Category I, which satisfies GDC 2.

The evaluation of the upper riser holes is addressed in Section 3.9.2 of this safety evaluation.

3.9.5.4.1.3 Design—Core Support Structure (Lower Riser Assembly)

DCA Part 2, Tier 2, Section 3.9.5.1, states that the lower riser assembly channels the reactor coolant flow leaving the reactor core upward toward the central 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 tubes, CRA guide tube support plate, and ICIGT support structure. The lower riser assembly is located immediately above the CSA and is aligned with and supported on the CSA by four upper support blocks.

In the October 10, 2017, letter (ADAMS Accession No. ML17284A092), the applicant stated that the CRA guide tube support plate is a grid structure with circular openings for the CRA guide tubes. Four equally spaced lugs extend to the ring at the top of the lower riser and are welded at these locations. The CRA guide tube support plate is classified as an internal structure and is designed to ASME BPV Code, Section III, Subsection NG.

The CRD shaft support structure is a grid, which is also called the ICIGT support. This structure supports both the CRD shaft and ICIGT. The CRD shaft support structure is welded at eight locations to the lower riser transition, located at the top of the lower riser assembly. The CRD shaft support structure is classified as an internal structure and is designed to ASME BPV Code, Section III, Subsection NG.

The upper core plate is the base of the lower riser assembly and has square openings that contain fuel pins. The fuel pin is inserted from the bottom with a special nut configuration, which fits in a counter bore, on top of the upper core plate. The special nut, an internally threaded cylinder with wrench flats and a rounded top, is called the fuel pin cap. It functions as the alignment pin for the lower flange on the CRA guide tube. Both the upper core plate and fuel pins are classified as a core support structure and are 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)

DCA Part 2, Tier 2, Section 3.9.5.1, states that the CSA includes the core barrel, upper support blocks, lower core plate, lower fuel pins and nuts, reflector blocks, lower core support lock inserts, and the RPV surveillance specimen capsule holder and capsules. The core barrel is a continuous ring with no welds. The upper support blocks, which are welded to the core barrel, center the core barrel in the lower RPV. One of the upper support blocks engages a core barrel guide feature on the lower RPV to provide circumferential positioning of the core barrel as it is lowered into the lower RPV. The lower core plate, which is welded to the bottom of the core barrel, supports and aligns the bottom end of the fuel assemblies. Locking devices align and secure the lower core plate to the core support blocks located on the RPV bottom head. TR-0716-50439-P briefly describes each of the major components of the CSA.

In the October 10, 2017, letter, the applicant stated that the core barrel is a cylindrical shell with eight cutouts at the top, which facilitate alignment and coupling with the lower riser assembly. Eight tabs extend radially from the perimeter of the upper core plate and fit into the cutouts at the top of the core barrel. The outer diameter of the upper core plate fits inside the core barrel. Four of the eight tabs are used for circumferential alignment only, while four of the tabs provide alignment but also include holes for socket head cap screws and alignment dowels that are used to attach the upper core plate to the upper support blocks. The core barrel is classified as a core support structure and is designed to ASME BPV Code, Section III, Subsection NG.

There are four upper support blocks welded to the core barrel, spaced at 90-degree intervals. One of the four support blocks functions to circumferentially align the core barrel during assembly via a guide feature. The blocks are approximately 0.6 m (2 ft) tall and 25 cm (10 in.) wide and fill the space between the core barrel and the RPV. The 25-cm (10-in.)-width tapers down to a larger radius at the bottom. This taper interfaces with NPM lifting equipment used during module assembly and disassembly. The top of each block includes threaded holes for socket head cap screws and holes for alignment dowels to couple the CSA to the lower riser. The upper support blocks, which transfer horizontal loads to the pressure vessel wall, are classified as core support structures and are designed to ASME BPV Code, Section III, Subsection NG. There is a single guide feature on the RPV wall, which assures that the CSA is properly oriented within the RPV. The guide feature consists of two rectangular bars placed approximately 20 cm (8 in.) apart and bent at the top to generate a lead-in when the CSA is lowered into the RPV. One of the four upper support blocks has notches on the upper portion of the taper so that it can slide into the alignment feature. Because the upper support block may possibly apply a load during a seismic event, the guide feature hardware is classified as a core support structure and is designed to ASME BPV Code, Section III, Subsection NG. The guide feature is welded to the vessel wall and is part of the vessel.

The lower core plate is a circular plate with a grid of square cutouts located at the base of the CSA. 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 52 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 nut is mounted in a counter bore and has a cup on the perimeter for locking. The fuel pins, including the nuts, are classified as core support structures and are designed to ASME BPV Code, Section III, Subsection NG.

Four core support blocks are welded to the bottom head of the RPV. These welds are jurisdictionally part of the RPV, and are designed to ASME BPV Code, Section III, Subsection NB. The core support blocks perform a core support function and are designed to

ASME BPV Code, Section III, Subsection NG. The core support blocks directly support the core under all service level conditions, and the CSA is secured to the core support blocks at all times. Therefore, the core barrel is not suspended from the RPV head flange. Section 3.9.2 of this safety evaluation provides more details about the core support blocks and their attachment mechanism to the lower core plate.

The reflector blocks are composed of a stack of blocks. Each block is a circular plate with a stepped cutout that matches the perimeter of the fuel. The stack of reflector blocks is not fastened or physically connected to the core barrel. Each block contains cooling channels, which are labeled as flow holes. The blocks are located with respect to each other by alignment pins. Likewise, the bottom reflector block is aligned with the lower core plate by alignment pins. These pins perform only an alignment function but may be loaded during a seismic event. Therefore, the pins are classified as core support structures. The reflector blocks themselves are also classified as core support structures. Both alignment pins and reflector blocks are designed to ASME BPV Code, Section III, Subsection NG.

There are four surveillance specimen capsule holders, and the base support is welded to the core barrel. The welds are part of the core barrel. Therefore, they are classified as core support structures and are designed to ASME BPV Code, Section III, Subsection NG. The surveillance specimen capsule holders themselves are classified as internal structures and are designed to ASME BPV Code, Section III, Subsection III, Subsection NG.

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 safety evaluation. Specifically, letters dated July 26, 2018, and November 1, 2018 (ADAMS Accession Nos. ML18207A526 andML18305B315), provide more detail for the mechanism to secure the core support blocks to the lower core plate, while letters dated January 31, 2019 and March 13, 2019 (ADAMS Accession Nos. ML19031C984 and ML19072A151), provide more detail for the relative movement of the reflector blocks.

Based on the information provided above and information provided in Section 3.9.2 of this safety evaluation, 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)

DCA Part 2, Tier 2, Section 3.9.5.1, states that there are 16 CRA guide tubes that are attached to the upper core plate and 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-0716-50439-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 consists of a hollow cylinder with slots for the CRA cards that are welded to the cylinder. A CRA alignment cone, with an internal taper, is welded at the top of the cylinder. A CRA lower flange, containing tabs with alignment holes, is welded to the bottom of the cylinder. The holes in the lower flange fit over the fuel pin caps and the flange sets on the upper core plate. The top of the CRA guide tube assembly fits into a counter bore, with a slip fit, in the lower side of the CRA guide tube support plate. The CRA guide tube is classified as an internal structure and is designed to ASME BPV Code, Section III, Subsection NG. 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)

DCA Part 2, Tier 2, Section 3.9.5.1, states that an ICIGT support structure is located inside the lower riser to support and align ICIGTs with their respective fuel assemblies. DCA Part 2, Tier 2, Figure 3.9-3, shows a typical ICIGT.

In the October 10, 2017, letter, the applicant stated that there are 12 ICIGTs. 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 fits in the socket in the lower side of the ICIGT support at the top of the lower riser assembly. The tube is a slip/clearance fit in the ICIGT support at this location to allow for thermal expansion. The ICIGTs do not make contact with the CRA guide tube support plate. The bottom end of these guide tube segments is welded to a short cruciform shape at the bottom, below the upper core plate, for centering in the square openings in the upper core plate. The cruciform shape at the bottom of the tube is then welded to the square opening in the upper core plate. The cruciform shapes are for alignment with the fourth segment of the ICIGTs. The fourth segment is part of the fuel assemblies. The upper three ICIGT segments are classified as internal structures and are designed to ASME BPV Code, Section III, Subsection NG.

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.

3.9.5.4.2.3 Design—Reactor Vessel Internals Other than Core Support Structures (Flow Diverter)

DCA Part 2, Tier 2, Section 3.9.5.1, states that a flow diverter is attached to the RPV bottom head under the CSA. This flow diverter smooths the turning of the reactor coolant flow from the downward flow outside the core barrel to upward flow through the fuel assemblies. The flow diverter reduces flow turbulence and recirculation and minimizes flow-related pressure loss.

In the October 10, 2017, letter, the applicant stated that the flow diverter is a thin disc with a raised bubble shape in its center. It is welded to the interior center of the bottom head of the RPV. The outer perimeter of the flow diverter does not reach the core support blocks. The top is below the lower core plate. Therefore, the flow diverter does not interfere with the core support blocks and does not carry loads from the CSA. The weld at which the flow diverter is welded to the bottom head of the RPV is part of the RPV. The flow diverter is classified as an internal structure and is designed to ASME BPV Code, Section III, Subsection NG.

The staff finds that the design code of the flow diverter 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.

The staff performed an audit to support the CRD shaft alignment drop test to determine the displacement limits for the CRD shaft supports. The result of this audit is documented in the audit report (ADAMS Accession No. ML18325A153). Section 3.9.4.4.5 of this safety evaluation details the result of this audit. In summary, the staff reviewed the results from the CRA drop test and CRD shaft alignment test and found the results to be acceptable. Based on the information provided and the result of the audit, 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. DCA Part 2, Tier 2, Table 3.9-5, includes 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 safety evaluation.

3.9.5.4.5 Flow-Induced Vibration

Section 3.9.2 of this safety evaluation 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

In the October 10, 2017, letter, the applicant stated that 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 DCA Part 2, Tier 2, 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 safety evaluation.

3.9.5.4.6.2 Other Design Parameters—Reactor Vessel Internals Gap Fit

In the October 10, 2017, letter, the applicant stated that the RVIs are analyzed to different load combinations, including both Service Levels A and B. Service Level A and B events include plant heatup and cooldown. RVIs are 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 are 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 DCA Part 2, Tier 1, Table 2.1-4 (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

DCA Part 2, Tier 2, 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 blocks at the bottom of the RPV and the upper support blocks that are attached to the upper portion of the core barrel. The vertical loads are transferred from the core barrel to the RPV head through the core support blocks.

In the October 10, 2017, letter, the applicant stated that during normal operation conditions (Level A condition), there are no lateral loads transmitted between the core barrel and the RPV. During some Service Level B, C, and D conditions such as seismic and blowdown events, lateral loads are transmitted through one or more of the upper support blocks. The core support assemblies located at the bottom of the RPV are welded to the RPV. The core support block top plate provides support for the socket head cap screw and alignment dowels that hold down the core support. Section 3.9.2 of this safety evaluation 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

DCA Part 2, Tier 2, 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.

In the October 10, 2017, letter, the applicant stated that 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, using the lower riser assembly lifting lugs, and is stored in a designated stand in the refueling pool. While the lower riser assembly is in the stand, it is supported by load-bearing 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 the information provided by the applicant acceptable because it clarifies the design measures taken to prevent damage of the fuel pins during refueling.

3.9.5.4.6.5 Other Design Parameters—Stress and Fatigue Analysis

The staff conducted the regulatory audits (Phase 1 and 2 audits) of NuScale design specifications, and the staff documented the Phase 1 and 2 audits in "Summary Audit Report of Design Specifications," dated January 25, 2018 (ADAMS Accession No. 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 (ADAMS Accession No. ML19018A140). The staff finds the RVI design information described in DCA Part 2, Tier 2, was adequately translated into the design specification documentation. In September 2019, the staff conducted a regulatory audit for the stress and fatigue analysis of the RVIs, and the staff documented the summary in an audit summary report (ADAMS Accession No. ML19340A971). The staff found the calculations in various stages of completion (some calculations were final with assumptions that needed to be verified, and some were not final) but was able to conclude that the calculations were being performed in accordance with ASME BPV Code, Section III, and 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 documents. Based on the result of this audit and the existence of ITAAC that will ensure compliance with ASME BPV Code requirements (e.g., DCA Part 2, Tier 1, Table 2.1-4, item 3), 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 DCA Part 2, Tier 2, Table 1.8-2.

Table 3.9.5-1: NuScale COL Information Item for Section 3.9.5

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.9-3	A COL applicant that references the NuScale Power Plant design certification will provide a summary of reactor core support structure 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

3.9.5.6 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 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 DCA.

3.9.6.2 Summary of Application

NuScale submitted Revision 3 to DCA Part 2 on August 22, 2019 (ADAMS Accession No. ML19241A315). 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.

DCA Part 2, Tier 1: DCA Part 2, Tier 1, specifies ITAAC for as-built components to confirm that their design requirements have been satisfied in the NuScale Power Plant. DCA Part 2, Tier 1, does not specify Tier 1 requirements specific to the IST program for pumps, valves, and dynamic restraints in the NuScale Power Plant.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.6, references the 2012 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 are not constructed to the ASME BPV Code but are relied on in the NuScale safety analyses. NuScale stated that it considered the NRC guidance in NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants," issued October 2013 (ADAMS Accession No. ML13295A020), in developing its IST program. In DCA Part 2, NuScale included a request to apply Appendix IV, "Preservice and Inservice Testing of Active Pneumatically Operated Valve Assemblies in Nuclear Reactor Power Plants," to the 2017 Edition of the ASME OM Code as an alternative to the stroke-time testing requirements for air-operated valves (AOVs) and hydraulic-operated valves (HOVs) in the 2012 Edition of the ASME OM Code. DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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-1-2007, as endorsed in RG 1.100, Revision 3, with clarifications as described in DCD Part 2, Tier 2, Section 3.10.2, "Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation." DCA Part 2, Tier 2, Section 3.10.2, indicates that ASME QME-1-2007 is used with the exceptions noted in RG 1.100, Revision 3, for the qualification of active mechanical equipment.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.6.2, "Inservice Testing of Pumps," indicates that the NuScale Power Plant design does not contain pumps, whether safety related or not safety related, that are within the ASME OM Code scope. Therefore, the NuScale IST plan does not address pumps.

DCA Part 2, Tier 2, 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 Light-Water Reactor Nuclear Power Plants." DCA Part 2 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. DCA Part 2, Tier 2, Table 3.9-16, “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. DCA Part 2, Tier 2, Section 3.9.6.3, indicates that the NuScale IST plan also includes augmented testing of valves that provide a backup that is not safety related to a safety-related function. DCA Part 2, Tier 2, Table 3.9-17, “Valve Augmented Requirements,” summarizes the augmented testing provisions for valves in the CVCS, CFWS, CNTS, and MSS. DCA Part 2, Tier 2, 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).

DCA Part 2, Tier 2, 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 open or close to reach its safety 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.

DCA Part 2, Tier 2, 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 POV exercise tests, POV skid-mounted components, POV performance assessment testing, and preservice performance assessment testing and QME-1, (4) check valve tests, and (5) pressure-relief device tests.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 3.9.6.4, “Relief Requests and Alternative Authorization 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).

DCA Part 2, Tier 2, 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. Section 3.9.6.4.1 notes that the NuScale TS do not have a Mode defined as “cold shutdown” as used in the 2012 Edition of the ASME OM Code.

DCA Part 2, Tier 2, Section 3.9.6.4.2, “ASME OM Code Version Alternate Authorization,” proposes use of paragraph ISTA-2000, “Definitions,” and Appendix IV to the ASME OM Code, 2017 Edition, in addition to the 2012 Edition of the ASME OM Code in implementing the NuScale IST plan. Section 3.9.6.4.2 specifies that the provisions in Appendix IV to the ASME OM Code, 2017 Edition, were used to develop the POV inservice performance assessment testing described in Section 3.9.6 and the IST tables.

DCA Part 2, Tier 2, Section 3.9.6.4.3, “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, Appendix IV. 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.

DCA Part 2, Tier 2, Section 3.9.6.5, “Augmented Valve Testing Program,” specifies that components not required by ASME OM Code, paragraph ISTA-1100, but with augmented quality requirements similar to ISTA-1100, are included in an augmented IST program. Section 3.9.6.5 notes that these components either provide a backup that is not safety related to a safety-related function or are valves that are not safety related and provide an augmented quality function. 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 endorsed by 10 CFR 50.55a(f), or where the NRC has granted relief 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 3.9-17.

Other sections of DCA Part 2, Tier 2, also specify provisions for various safety-related valves in the NuScale Power Plant design. For example, 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. Section 3.9.3.3 references ASME Standard QME-1-2007 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 reactor vent valves (RVVs). Section 6.2.4, “Containment Isolation System,” describes the CNTS, including the design and operation of the containment isolation valves (CIVs). 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 reactor recirculation valves (RRVs).

ITAAC: DCA Part 2, Tier 1, 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 DCA 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 DCA does not include technical reports for DCA Part 2, Tier 2, 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 52.47, "Contents of Applications; Technical Information," as it relates to, for example, (a) the design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design, (b) information to allow audit of certain procurement specifications and construction and installation specifications, and (c) the application contents that must contain an FSAR that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the SSCs and of the facility as a whole

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 (ADAMS Accession No. ML020700641), and associated Staff Requirements Memorandum, dated September 11, 2002 (ADAMS Accession No. ML022540755)

- SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses, and Acceptance Criteria," dated February 26, 2004 (ADAMS Accession No. ML040230079), and associated Staff Requirements Memorandum, dated May 14, 2004 (ADAMS Accession No. 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 (ADAMS Accession No. ML052770257), and associated Staff Requirements Memorandum, dated February 22, 2006 (ADAMS Accession No. 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 DCA. In addition to design aspects, the staff evaluated DCA Part 2, Tier 2, Section 3.9.6, and its associated sections to determine whether the DCA Part 2 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. As part of its review, the staff evaluated whether the description of the functional design, qualification, and IST programs in DCA Part 2 is acceptable for incorporation by reference in a COL application, together with the plant-specific aspects of those programs.

In its review of a DCA, the staff evaluates whether the application provides assurance that the IST provisions of the ASME OM Code referenced in DCA Part 2 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 DCA, the staff evaluated the description of the IST program in DCA Part 2 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.

DCA Part 2, Tier 2, 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 DC. As part of its review of the DCA, the NRC staff issued several RAIs to NuScale to resolve staff questions on the information provided in the original DCA submittal. In response to the RAIs, NuScale revised the DCA to clarify specific information with respect to functional design, qualification, and IST programs for pumps, valves, and dynamic restraints. The staff reviewed this information to verify that that PST and IST programs were fully described for reference by a COL applicant, consistent with the stated intent of the applicant.

In a letter dated November 2, 2017 (ADAMS Accession No. ML17306A803), the applicant stated that DCA Part 2, Tier 2, Section 3.9.6, fully describes the PST and IST programs for reference by a COL applicant, consistent with the Commission policy for operational programs for new reactors. Based on its review described in this safety evaluation, the staff confirmed that DCA Part 2, Tier 2, Section 3.9.6, fully describes the PST and IST programs for the NuScale Power Plant, consistent with the Commission policy for operational programs for new reactors.

3.9.6.4.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

DCA Part 2, Tier 2, Section 3.9.6, describes the functional design and qualification provisions and IST program for the NuScale Power Plant. DCA Part 2, Tier 2, Section 3.9.6.1, specifies that the functional design and qualification of safety-related valves is performed in accordance with ASME Standard QME-1-2007, as endorsed in RG 1.100, Revision 3, with clarifications as described in DCA Part 2, Tier 2, Section 3.10.2. DCA Part 2, Tier 2, Section 3.10.2, indicates that ASME QME-1-2007 is used with the exceptions noted in RG 1.100, Revision 3, for the qualification of active mechanical equipment.

In response to performance issues with POVs at operating nuclear power plants, the design and qualification process for demonstrating the capability of POVs to perform their safety functions has been specified in previous DCAs as either Tier 1 or Tier 2* information to provide assurance that the NRC will have an opportunity to review in advance any planned modifications to the valve qualification process. DCA Part 2, Tier 2, Section 3.9.6, specifies that safety-related valves will satisfy the qualification provisions of ASME Standard QME-1-2007, as endorsed in RG 1.100, Revision 3, with clarifications as described in DCA Part 2, Tier 2, Section 3.10.2.

Section 3.10.2 indicates that ASME QME-1-2007 is used with the exceptions noted in RG 1.100, Revision 3, for the qualification of active mechanical equipment. Based on the safety significance of the proper performance of POVs, the staff finds the process to demonstrate the functional capability of safety-related POVs in the NuScale Power Plant to be appropriate as a Tier 1 requirement.

In a letter (ADAMS Accession No. ML17213A540) dated August 1, 2017, the applicant stated that it has committed to use the ASME Standard QME-1 to qualify all safety-related valves in DCA Part 2, Tier 2, Sections 3.9.6 and 3.10, and Table 14.3-1, "Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference," and the valve design specifications. The applicant considered that both DCA Part 2 and design specifications provide reasonable assurance that all safety-related valves will be qualified in accordance with ASME Standard QME-1 as stated in RG 1.100. In letters dated December 27, 2017 (ADAMS Accession No. ML17361A136) and May 24, 2018 (ADAMS Accession No. ML18144A918), the applicant stated that the reference to the Qualification Report for safety-related valves in ITAAC 6 in DCA Part 2, Tier 1, Table 2.8-2, agrees with the definitions in ASME QME-1-2007. The applicant also stated that the discussion related to ITAAC 02.08.06 in DCA Part 2, Tier 2, Table 14.3-1, indicates that the functional qualification of safety-related valves is performed in accordance with ASME QME-1-2007 (or later edition), as accepted in RG 1.100, Revision 3 (or later revision). The staff finds that the applicant has clarified the Tier 1 requirement that safety-related valves will be qualified in accordance with the ASME Standard QME-1-2007 as accepted in RG 1.100, Revision 3, or a more recent edition of the standard accepted by the NRC.

The NRC regulations in 10 CFR 52.47 provide that the NRC will require, before certification of a design, that information normally contained in certain procurement specifications and construction and installation specifications be complete and available for audit if the information is necessary for the Commission to make its safety determination.

In a letter dated August 1, 2017 (ADAMS Accession No. ML17213A540), the applicant stated that the design specifications for the ECCS valves were available for audit. The applicant indicated that the design specifications include requirements for design, analysis, materials of construction, fabrication, inspection and examination, testing, preparation, shipment, delivery, owner and supplier responsibilities, and environmental control during fabrication. The applicant stated that this information is sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC and procurement specifications and construction and installation specifications by an applicant, as required by 10 CFR 52.47.

The staff conducted an initial audit of the NuScale component design specifications from June 1, 2017, to August 29, 2017. A report dated January 25, 2018 (ADAMS Accession No. ML18018A234), documents the staff's audit results and followup items. The staff conducted a followup audit of the NuScale design specifications from May 14, 2018, through October 11, 2018. The staff issued a followup audit report on February 11, 2019 (ADAMS Accession No. ML19018A140), with specific open and confirmatory items.

From July 30, 2019, to September 25, 2019, the staff conducted a close-out audit of the revised NuScale design specifications to verify that the remaining items from the followup audit report had been resolved. On October 21, 2019, the staff issued a close-out audit report describing the completion of the followup items for the NuScale design specifications (ADAMS Accession No. ML19291A887). The staff found that the applicant had revised the NuScale design specifications in an acceptable manner to resolve the confirmatory items specified in the followup audit report for the NuScale component design specifications. For the reasons stated above, the staff considers the design specifications to address the design requirements included in GDC 1, 2, 4, 14, 15, 37, 40, 43, 46, and 54 in Appendix A to 10 CFR Part 50.

Based on its audit review, the NRC staff finds, as required by the NRC regulations in 10 CFR 52.47, that information normally contained in procurement specifications and construction and installation specifications is sufficiently complete for the Commission to make a safety determination regarding the NuScale DCA. The staff also finds that the NuScale component design specifications are consistent with the provisions in the NuScale DCA for the design of the specific components sampled in the audit.

3.9.6.4.2 Inservice Testing Program for Pumps

DCA Part 2, Tier 2, 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 DCA Part 2 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).

DCA Part 2, Tier 2, Section 3.9.6, references the 2012 Edition of the ASME OM Code for the description of the IST program in support of the NuScale DCA. The staff finds the reference to the 2012 Edition of the ASME OM Code in the DCA Part 2 description of the IST program for the NuScale DC to be acceptable where implemented as incorporated by reference in 10 CFR 50.55a. 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 120-month 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. DCA Part 2, Tier 2, 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 DCA 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. NuScale DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Table 14.2-100, "Ramp Change in Load Demand Test #100," and Table 14.2-108, "NuScale Power Module Vibration Test #108," reference the development of SSC vibration acceptance criteria during initial power operations. 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 DCA Part 2 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 DCA. In particular, DCA Part 2, Tier 2, Section 3.9.6, indicates that POVs other than MOVs in the NuScale Power Plant include AOVs, 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 (ADAMS Accession No. 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 (ADAMS Accession No. 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 (ADAMS Accession No. 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" (ADAMS Accession No. ML003686003), to discuss the application of lessons learned from valve operating experience and research programs on POVs.

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. As discussed below, NuScale DCA Part 2, Tier 2, Section 3.9.6, includes a request to apply Appendix IV to the ASME OM Code (2017 Edition) to address periodic verification of the design-basis capability of AOVs and HOVs in the NuScale Power Plant.

For the IST program applicable to AOVs and HOVs, DCA Part 2, Tier 2, 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) as an alternative to the 2012 Edition of the ASME OM Code incorporated by reference in 10 CFR 50.55a. As an update to quarterly stroke-time testing of AOVs in the 2012 Edition of the ASME OM Code, 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. The staff describes its review and authorization of the alternative request submitted by NuScale in accordance with 10 CFR 50.55a(z) to implement ASME OM Code (2017 Edition), Appendix IV, for IST program activities for AOVs and HOVs later in this SER section.

For the IST program applicable to SOVs, DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, Table 3.9-15, “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 DCA Part 2, Tier 2, Table 3.9-16, also indicates that each ECCS valve system includes a trip and reset SOV. Subsection ISTC, “Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants,” in the ASME OM Code (2012 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 DCA Part 2, Tier 2, Section 3.9.6.3.2, describes the IST program for check valves in the NuScale Power Plant. Section 3.9.6.3.2 indicates that there are four check valves for each NPM in the NuScale IST plan. Section 3.9.6.3.2 indicates that the check valves will be grouped when applying the ASME OM Code, Appendix II. DCA Part 2, Tier 2, Table 3.9-16 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. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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. 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. 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 DCA 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 SER Section 3.9.6.4.6.2.

3.9.6.4.3.6 Inservice Testing Program for Safety and Relief Valves

Paragraph (5), "Pressure Relief Device Tests," in DCA Part 2, Tier 2, 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 DCA is consistent with the NRC regulatory requirements for safety and relief valves. The staff review of the RSV design is described later in SER Section 3.9.6.4.6.3.

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 DCA Part 2 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 DCA Part 2 did not address squib valves.

3.9.6.4.3.9 Rupture Disks

NuScale DCA Part 2, Tier 2, Table 3.9-16, specifies that the NuScale Power Plant includes two rupture disks as ASME OM Code, Category D, valves. Note 17 to Table 3.9-16 states that the rupture disks are passive, redundant, nonreclosing pressure-relief devices that provide RXB overpressure protection with IST provisions in accordance with ASME OM Code (2012 Edition), Appendix I, paragraph I-1360, "Test Frequency, Classes 2 and 3 Nonreclosing Pressure Relief Devices." Although the staff considers the ASME OM Code provisions to apply to piping systems, the applicant is allowed to use the Code provisions for other applications (e.g., building overpressure protection). The staff has no objection to this conservative 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 DCA Part 2, the staff finds that the NuScale IST program tables incorporate the provisions of the ASME OM Code (2012 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

DCA Part 2, Tier 2, 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 DCA Part 2 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 DCA, the applicant submitted one request for relief from the ASME OM Code provisions, one request for an alternative to the edition of the ASME OM Code to be applied in fully describing the IST program as part of the DC for the NuScale nuclear power plant, and one request for an alternative to the specific provisions of the ASME OM Code for the IAB valve in the ECCS valve system. The NRC staff review of these requests is described below.

3.9.6.4.5.1 Cold Shutdown Definition Relief Request

NuScale DCA Part 2, Tier 2, Section 3.9.6.4.1, requests relief from the ASME OM Code in accordance with 10 CFR 50.55a(f)(5) with respect to the use of the term "cold shutdown" in the ASME OM Code (2012 Edition). In particular, 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 the NuScale TS 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). DCA Part 2, Tier 2, 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 less than [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 (2012 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 reactor coolant temperatures less than [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 outage" relief request in DCA Part 2, Tier 2, 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 ASME OM Code Version Alternative Authorization

NuScale DCA Part 2, Tier 2, Section 3.9.6.4.2, includes a request by the applicant in accordance with 10 CFR 50.55a(z) to apply Appendix IV of the ASME OM Code (2017 Edition) as an alternative to the IST provisions in the ASME OM Code (2012 Edition). Section 3.9.6.4.2 states that Appendix IV is proposed as an alternative to the ASME OM Code (2012 Edition) to develop inservice performance assessment testing as described in Section 3.9.6.3.2 and Table 3.9-16. Section 3.9.6.4.2 also states that Appendix IV provides an acceptable level of quality and safety by using established ASME OM Code requirements to demonstrate that valves 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 POVs in the 2012 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 POVs with periodic diagnostic testing of high-safety significant POVs, in addition to the current ASME OM Code requirements for quarterly POV stroke-time testing. Therefore, the staff concludes that the Appendix IV alternative request proposed in DCA Part 2, Tier 2, Section 3.9.6.4.2, for the POVs 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).

3.9.6.4.5.3 Inadvertent Actuation Block Valve Test Frequency Alternative Authorization

NuScale DCA Part 2, Tier 2, Section 3.9.6.4.3, 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. As discussed above, the NRC accepts the request by the applicant to implement the provisions of Appendix IV to the ASME OM Code (2017 Edition) as an alternative to the provisions in ASME OM Code (2012 Edition). 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 under 10 CFR 50.55a(z) to the ASME OM Code requirements, Section 3.9.6.4.3 includes a request that (1) the preservice testing for the IAB valves in 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 DCA. For preservice testing of the RVV and RRV IAB valves, Section 3.9.6.4.3 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, 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, Section 3.9.6.4.3 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 RVV IAB valve and one RRV 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 RVV IAB valve and one RRV 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) this testing will be performed as an alternative to ASME OM Code (2017 Edition), Appendix IV, paragraph IV-3420, "Stroke Testing."

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 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. Section 3.9.6.4.3 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 Table 3.9-16 indicates that Section 3.9.6.4.3 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 IAB valves from the ASME OM Code requirements to determine whether the requirements in 10 CFR 50.55a(z) are satisfied. As noted in Section 3.9.6.4.3, the applicant groups the ECCS valves on a plantwide (multimodule) basis to optimize testing, examination, and maintenance activities. When 12 NPMs are installed in a NuScale nuclear power plant, six 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 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 DCA Part 2, Tier 2, 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).

3.9.6.4.6 Specific Valve Review

Several sections of DCA Part 2, Tier 2, 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 DCA Part 2 sections as discussed in the following paragraphs.

3.9.6.4.6.1 Emergency Core Cooling System Valves

DCA Part 2, Tier 2, 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 reactor pressure vessel (RPV). The ECCS valves include three RVVs at the top of the RPV and two RRVs on the side of the RPV above the active fuel level. DCA Part 2, Tier 2, 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 NuScale design includes a first-of-a-kind (FOAK) combination of individual valve components for each ECCS valve. In particular, DCA Part 2, Tier 2, Section 6.3, indicates that each ECCS valve consists of four distinct valve components connected by several feet of tubing that contains borated reactor coolant used as the valve hydraulic fluid. The four components are the following:

(1) the main valve which is held closed by hydraulic force from reactor coolant pressure in the main valve control chamber and is opened by reactor coolant pressure (with a small spring to hold the main valve open) when the main valve control chamber is vented by tubing through the IAB valve and trip valve into the CNV

(2) the solenoid-operated trip valve located outside the CNV in the cooling pool that is normally closed and is deenergized to open to vent borated reactor coolant from the main valve control chamber (provided that the IAB valve allows passage of reactor coolant)

(3) the solenoid-operated reset valve located outside the CNV in the cooling pool that is normally closed and is energized to open to pressurize the main valve control chamber with borated reactor coolant from an outside source (provided that the IAB valve allows passage of reactor coolant) to close the main valve against spring force

(4) the IAB valve located between the main valve and SOVs, which is a spring-loaded differential pressure valve that functions to block venting of the main valve control chamber when the RPV to CNV differential pressure is above a predetermined threshold and then uses spring force to overcome the differential pressure when reduced between the RPV and CNV to retract the IAB valve disc to allow reactor coolant to be vented from the main valve control chamber to allow the main valve to open, or to be supplied to the main valve control chamber to close the main valve prior to plant startup, through the applicable SOV

The 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. The NRC regulations in 10 CFR 52.47 specify that a DCA must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. Under 10 CFR 52.47(c)(2), a DC 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 a DCA that uses simplified, inherent, passive, or other innovative means to accomplish the 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.

As part of the DCA review of the FOAK design of the NuScale ECCS valves, the staff conducted a detailed review and multiple audits for this aspect of the NuScale design. In a letter dated August 1, 2017 (ADAMS Accession No. ML17213A540), the applicant provided information related to the availability of ECCS valve design documentation to support staff audit activities.

The staff performed an initial audit of the NuScale ECCS valve design to determine whether 10 CFR 52.47(c)(2) and 10 CFR 50.43(e) had been satisfied. The audit review topics included the following:

- Determine whether the ECCS design drawings and other design documents support the FOAK valve design as reasonable to perform the safety functions specified in the NuScale DCA.

- Determine whether the ECCS valves (and the valve subcomponents) will perform their safety functions in a timely manner over their full range of operational conditions.

- Determine whether the ECCS valves will not inadvertently open when the differential pressure between the RPV and CNV exceeds the specified conditions.

- Determine whether the Failure Modes and Effects Analysis (FMEA) for the ECCS valve design addresses potential failure mechanisms to provide reasonable assurance that the valve design analysis and testing will demonstrate the capability and reliability of the ECCS valves.

- Determine whether the IAB valve in the ECCS can be assumed to be a passive device with a reliability consistent with the Commission policy on passive components with respect to the single-failure criterion.

- Determine whether the ECCS valve will fully open reliably during operation of the main valve and pressure release from the main chamber through the IAB valve.

- Determine whether the plans for valve design testing will demonstrate the capability and reliability of the ECCS valve to support the assumptions in the NuScale DCA.

- Determine whether the qualification plans are sufficient to provide reasonable assurance that a COL holder for the NuScale design will demonstrate the qualification of the ECCS valves to perform their safety functions over the full range of operational conditions up through design-basis conditions.

On February 26, 2018, the staff issued a report (ADAMS Accession No. ML18052A079) describing its findings from the initial audit of the NuScale ECCS valve design documentation. The followup items specified in the initial ECCS valve audit report are listed below:

- the capability of the main valve to open fully in a timely manner for design-basis conditions when required

- the capability of the main valve to not partially or fully open prematurely

- the capability of the IAB valve to close and seal the vent line in a timely manner at the initial opening of the trip valve to prevent the main valve from opening partially or fully until the differential pressure between the RPV and CNV has reduced to the specified conditions

- the capability of the IAB valve to open in a timely manner when the differential pressure between the RPV and CNV has reduced to the specified conditions to allow the main valve to open fully to initiate emergency core cooling within the time specified in accident analyses

- the capability of the trip valve and line size, orifices, fittings, and installed configuration to vent the trip line adequately in a timely manner to allow the differential pressure between the RPV and CNV to close and seal the IAB valve against the force of the IAB spring to prevent the main valve from opening partially or fully (with consideration of hot borated water flashing to steam and boron deposits) until the differential pressure between the RPV and CNV has reduced to the specified conditions

- the capability of the trip valve and line size, fittings, and installed configuration to vent the trip line adequately in a timely manner after the IAB valve has opened to vent the main valve control chamber (with consideration of hot borated water flashing to steam and boron deposits) to allow the main valve to fully open within its stroke-time requirements

As a followup to the initial audit, the staff conducted an audit of the FMEA and supporting documents for the design of the NuScale ECCS valves. In addition to reviewing the design documents, the staff conducted an onsite audit review at the Target Rock facility in Farmingdale, NY. During the onsite audit, the staff reviewed reports, calculations, analyses, and drawings related to the ECCS valve design and discussed the ECCS valve design with NuScale and Target Rock personnel. The staff conducted the onsite audit in conjunction with an NRC vendor inspection evaluating the design and test control, and other QA activities, by Target Rock for the NuScale ECCS valves. The staff describes the results of the vendor inspection at Target Rock in a report dated July 2, 2018 (ADAMS Accession No. ML18183A384).

On August 14, 2018, the staff issued a report describing its findings from the audit of the FMEA and other design documentation for the ECCS valves (ADAMS Accession No. ML18219B634). Based on the audit, the staff determined that NuScale had not provided sufficient information to demonstrate the safety features of the ECCS valves as required by 10 CFR 52.47(c)(2) and 10 CFR 50.43(e). For example, the proof of concept testing of the initial conceptual design of ECCS valves performed for NuScale was not sufficient to demonstrate the design performance of the ECCS valve systems for the NuScale Power Plant in accordance with 10 CFR 52.47(c) and 10 CFR 50.43(e). In addition, NuScale needed to address the technical aspects in resolving the safety questions regarding a partially open failure mode for an ECCS valve. NuScale also needed to address the technical aspects in resolving the safety questions related to its assumption that the IAB valve may be categorized as a passive component with respect to the single-failure criterion or to consider the single failure of the IAB in the accident analysis. NuScale also needed to update the FMEA and other ECCS valve design documents to resolve the audit findings. Therefore, the staff audit report concluded that NuScale had not demonstrated the capability and reliability of the ECCS valves to perform their safety functions to support the NuScale DCA. In an enclosure to the audit report, the staff provided a detailed list of the remaining items to be addressed to demonstrate the design of the ECCS valves.

On September 21, 2018, NuScale submitted its response (ADAMS Accession No. ML18264A312) to the ECCS valve audit report, including plans to conduct ECCS Valve Design Demonstration Testing, to resolve the findings from the NRC audit of the ECCS valve design.

On December 14, 2018, the applicant submitted a letter (ADAMS Accession No. ML18351A145) describing its position with respect to application of the single-failure criterion to the IAB valve used in the ECCS valve system in the accident analyses for the NuScale Power Plant. In Commission Paper SECY-19-0036, “Application of the Single Failure Criterion to NuScale

Power LLC's Inadvertent Actuation Block Valves," dated April 11, 2019 (ADAMS Accession No. ML19060A162), the NRC staff requested that the Commission provide guidance on the appropriate approach to address the single-failure criterion with respect to the IAB valve in the NuScale accident analyses. On July 2, 2019, the Commission issued SRM-SECY-19-0036 (ADAMS Accession No. ML19183A408) directing that the NRC staff should review Chapter 15, "Transient and Accident Analyses," of the NuScale DCA Part 2, Tier 2, without assuming a single active failure of the IAB valve to close. Additional discussion of the staff's implementation of SRM-SECY-19-0036 is contained in Section 15.0.5 of this SER. The NRC staff is following the Commission direction on the performance of the IAB valve in its review of the NuScale accident analyses documented in Chapter 15 of this SER. With respect to the IST program, the ECCS valve system is categorized as an active function as specified in DCA Part 2, Tier 2, Table 3.9-16.

From March 20, 2019, to October 9, 2019, the NRC staff conducted an audit of the ECCS Valve Design Demonstration Testing performed by NuScale at the Target Rock facility to satisfy 10 CFR 52.47(c)(2) and 10 CFR 50.43(e) for the FOAK design of the ECCS valve system. The staff also reviewed the revisions to ECCS valve design documentation, including the FMEA and other documents, to address the followup items specified in the FMEA audit report. As part of its audit review, the staff observed sample test runs of the ECCS Valve Design Demonstration Testing at Target Rock in June and July 2019. During the week of June 3, 2019, the staff conducted a vendor inspection of the NuScale activities at Target Rock for the ECCS Valve Design Demonstration Testing. The staff described the results of the vendor inspection in a report dated July 18, 2019 (ADAMS Accession No. ML19197A241). Based on the vendor inspection and onsite audit review, the staff considers the QA activities for the NuScale performance of the ECCS Valve Design Demonstration Testing to be acceptable because they satisfy the regulatory requirements in 10 CFR Part 50, Appendix B.

As part of the audit, the staff reviewed the final test report from the ECCS Valve Design Demonstration Testing, the revised FMEA report, and other ECCS valve design documents to address the previous audit findings. During the ECCS Valve Design Demonstration Testing, NuScale determined that design changes to the ECCS valve system (including the main valve and IAB valve) were necessary to demonstrate the functionality of the ECCS valve system to perform its design functions. In addition, NuScale determined that the DCA needs to be updated to reflect the ECCS valve performance attributes based on the results of the ECCS Valve Design Demonstration Testing. In addition to describing the testing results, the final test report for the ECCS Valve Design Demonstration Testing indicates that the lessons learned from the testing program need to be considered in the ASME QME-1 qualification program for the ECCS valve system. The revised FMEA report applies an updated methodology to identify potential failure modes for the ECCS valve system, and references the ASME QME-1 qualification program and the IST program for avoiding specific identified failure modes. On September 27, 2019, the staff conducted a review at Target Rock of the vendor calculations evaluating the potential for partial opening of the main valve in the ECCS valve system.

On December 19, 2019, the staff issued a report on the audit of ECCS Valve Design Demonstration Testing and the followup items from the FMEA audit (ADAMS Accession No. ML19340A019). In the audit report, the staff indicates the remaining items to be resolved to complete its review of the design of the ECCS valve system for the NuScale DCA. Based on its audit review, the staff found that the lessons learned from the ECCS Valve Design Demonstration Testing need to be considered in developing the ASME QME-1 qualification program and the IST program for the ECCS valve system. In addition, the staff found that the

ECCS performance attributes specified in the NuScale DCA were not consistent with the results of the ECCS Valve Design Demonstration Testing.

In describing its response to the ECCS valve audit, the applicant submitted letters dated August 21, 2019 (ADAMS Accession No. ML19233A203), October 24, 2019 (ADAMS Accession No. ML19297H199), and November 13, 2019 (ADAMS Accession No. ML19317E531). In a letter dated April 7, 2020 (ADAMS Accession No. ML20098G237), the applicant described the status of the follow-up items from the FMEA audit. For example, the applicant had updated the ECCS valve performance attributes and the IST program description for the ECCS valve system (including the subcomponent valves) in DCA Part 2, Tier 2, to reflect the lessons learned from the ECCS Valve Design Demonstration Testing. In addition, the applicant had initiated engineering change processes to update ECCS valve design documents to reflect the modifications to the design and performance characteristics of the ECCS valve system based on the results of the ECCS Valve Design Demonstration Testing. A COL applicant will address the lessons learned from the ECCS Valve Design Demonstration Testing in developing the ASME QME-1 qualification testing program for the ECCS valve system (including its subcomponent valves). Based on the above discussion, the staff finds that the ECCS valve system design addresses the design requirements in GDC 1, 4, 14, 15, 37, and 46 in Appendix A to 10 CFR Part 50.

In an audit report dated July 28, 2020 (ADAMS Accession No. ML20160A224) addressing potential boron redistribution in the NuScale reactor, the NRC staff describes its review of the applicant's demonstration of the performance of the ECCS main valve spring to open the main valve at very low differential pressure conditions. As described in the audit report, the NRC staff finds that the testing of the ECCS main valve spring at the Target Rock facility satisfies 10 CFR 50.43(e) in demonstrating this design feature of the ECCS main valve spring in support of the NuScale DC application. Further, the design specifications for the ECCS valve system will address the operating parameters for the ECCS main valve spring such that the ASME QME-1 qualification program will include a demonstration of the ECCS main valve spring performance. In addition, the IST program includes provisions for providing reasonable assurance of the performance of the ECCS main valve spring as part of the reactor shutdown process or as an individual IST activity.

Based on the above, the staff confirmed that the ECCS Valve Design Demonstration Testing performed by NuScale satisfied 10 CFR 52.47(c)(2) and 10 CFR 50.43(e) to demonstrate the safety feature of the ECCS valve system described in the NuScale DCA.

3.9.6.4.6.2 Containment Isolation Valves

DCA Part 2, Tier 2, 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 FW isolation check valve housed in the same valve body. DCA Part 2, Tier 2, 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.

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 DCA Part 2 and the design documents for the CIVs as required by 10 CFR 52.47 to support the NuScale DCA. As part of its review, the staff conducted an audit of the CIV design documentation from September 4 to October 31, 2018. From its audit 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 DCA of the NuScale SMR. In addition, DCA Part 2 and the design specifications for the PSCIVs and SSCIVs require that these valves will be qualified in accordance with ASME Standard QME-1-2007, which is endorsed in RG 1.100, Revision 3, to provide reasonable assurance of the capability of the PSCIVs and SSCIVs to perform their safety functions. Based on the audit, the staff determined that NuScale has provided sufficient information in the CIV design documentation to be used in the NuScale Power Plant as required by 10 CFR 52.47 to support the NuScale DC, with the exception of the followup items identified in the audit report dated December 7, 2018 (ADAMS Accession No. ML18331A042). The followup items in the CIV audit report included, for example, updates to (1) DCA Part 2, Tier 2, Section 6.2, to clarify several aspects of the design description of the CIVs and (2) the PSCIV and SSCIV design specifications to incorporate the reactor module loading specification.

On January 31, 2019, NuScale provided its closure plan for the CIV audit followup items, including updates to DCA Part 2 and the design documentation (ADAMS Accession No. ML19031C973). In a letter dated June 28, 2019, NuScale notified the NRC of the status of the followup items from the initial regulatory audit of the CIV design documents (ADAMS Accession No. ML19179A183).

From July 30, 2019, to September 25, 2019, the NRC staff conducted an audit of the resolution of the followup items from the initial CIV design audit. On November 26, 2019, the staff issued a report on its audit of the CIV design and discussed the resolution of the followup items (ADAMS Accession No. ML19319C232). As discussed in the audit report, NuScale has revised DCA Part 2, Tier 2, Section 6.2.4, and the PSCIV and SSCIV design specifications to resolve the followup items from the initial CIV design audit. Based on its review, the staff finds that the description in the DCA of the design and qualification provisions for the PSCIVs and SSCIVs provides reasonable assurance of their capability to perform the applicable safety functions. In addition, the staff finds that the PSCIV and SSCIV design specifications provide design and qualification provisions consistent with the DCA. Therefore, the staff finds the design and qualification provisions for the PSCIVs and SSCIVs in the DCA and their design specifications to be acceptable for the NRC review of the NuScale DCA because the CIV design addresses the design requirements in GDC 1, 2, 4, 14, 40, 43, 46, and 54 in Appendix A to 10 CFR Part 50.

In addition to the audit, the NRC staff reviewed the description in the NuScale DCA of the design, qualification, and testing of the CIVs. For example, DCA Part 2, Tier 2, Section 3.9.6.1, specifies that the functional design and qualification of safety-related valves are performed in accordance with ASME Standard QME-1-2007 as accepted in RG 1.100, Revision 3. DCA Part 2, Tier 2, 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 of the ASME OM Code. DCA Part 2, Tier 2, Table 3.9-16, specifies that CIVs are categorized as active valves in the IST program and indicates the IST program provisions in accordance with the ASME OM Code and the 10 CFR 50.55a(z) alternative authorized in this SER section. Based on its review, the staff finds the description in the NuScale DCA for the design, qualification, and testing of the CIVs to be acceptable for the NRC review of the NuScale DCA.

3.9.6.4.6.3 Reactor Safety Valves

DCA Part 2, Tier 2, 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 PZR 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. The staff reviewed the description of the RSVs in DCA Part 2 and prepared RAIs to obtain additional information on the RSV design, operation, and testing. In addition, the staff conducted an audit of the RSV design documentation. The following paragraphs describe the staff review of the RSV design, operation, and testing.

DCA Part 2, Tier 2, Section 5.2.2.4.1, “Reactor Safety Valves,” specifies that the RSVs are safety-related, Seismic Category I, Quality Group A, components. DCA Part 2, Tier 2, 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.” DCA Part 2, Tier 2, 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.”

As part of its review, the staff conducted an audit of the RSV design documentation from September 4 to October 31, 2018. From its audit 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 DCA. The staff found that DCA Part 2 and the design specifications indicate 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-1-2007, which is endorsed by RG 1.100, Revision 3, to provide reasonable assurance of the capability of the RSVs to perform their safety functions. Based on the initial audit, the staff determined that NuScale has provided sufficient information in the RSV design documentation to be used in the NuScale Power Plant as required by 10 CFR 52.47 to support the NuScale DCA, with the exception of the followup items identified in the audit report dated December 7, 2018 (ADAMS Accession No. ML18331A042). The followup items from the RSV audit included, for example, updates to (1) the RSV design specification to address valve size information, inservice examination provisions, pressure and temperature specifications, and reactor module nozzle loads and (2) the RSV diagram to specify the valve opening time and applicable ASME BPV Code edition.

On January 31, 2019 (ADAMS Accession No. ML19031C973), NuScale provided its closure plan for the RSV audit followup items, including updates to DCA Part 2 and the design documentation. In a letter dated June 28, 2019, NuScale notified the NRC of the status of the followup items from the initial regulatory audit of the RSV design documents (ADAMS Accession No. ML19179A183).

From July 30, 2019, to September 25, 2019, the NRC staff conducted an audit of the resolution of the followup items from the initial RSV design audit. On November 26, 2019, the staff issued a report on its audit of the RSV design and discussed the resolution of the followup items (ADAMS Accession No. ML19319C232). As discussed in the audit report, NuScale has revised the RSV design specification and RSV diagram to resolve the followup items from the initial RSV design audit. Based on its review, the staff finds that the description in the DCA of the design and qualification provisions for the RSVs provides reasonable assurance of their capability to perform the applicable safety functions. In addition, the staff finds that the RSV design specification provides design and qualification provisions consistent with the DCA. Therefore, the staff finds that the design and qualification provisions for the RSVs in the DCA and their design specification to be acceptable for the NRC review of the NuScale DCA because the RSV design addresses the design requirements in GDC 1, 4, 14, and 15 in Appendix A to 10 CFR Part 50.

In addition to the audit, the staff reviewed specific provisions provided in the NuScale DCA for the design and qualification of the RSVs. For example, DCA Part 2, Tier 2, Section 5.2.2.2.2, includes COL Item 5.2-2 specifying that a COL applicant that references the NuScale Power Plant DC 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 5.2.2.9, “System Reliability,” indicates that the RSVs are considered passive devices with respect to accident analyses. This SER 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 DCA Part 2, Tier 2, Chapter 15, “Transient and Accident Analyses.” With respect to its review of the NuScale IST program, the staff notes that DCA Part 2, Tier 2, Table 3.9-15, 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. DCA Part 2, Tier 2, Table 3.9-16, 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.

In a letter dated October 4, 2017 (ADAMS Accession No. ML17277B826), the applicant stated that the RSVs will be functionally qualified in accordance with ASME QME-1 as addressed in DCA Part 2, Tier 2, Section 3.10, with the specific details addressed in the QME-1 Qualification Specification and QME-1 Qualification Program, and documented in the QME-1 Qualification

Report. This statement conforms to the provision in NuScale DCA Part 2, Tier 2, Section 3.9.6.1, that the functional design and qualification of safety-related valves is performed in accordance with ASME Standard QME-1 as endorsed in RG 1.100, as described in DCA Part 2, Tier 2, Section 3.10.2. The NRC has accepted the implementation of ASME Standard QME-1-2007 in RG 1.100, Revision 3. The staff finds the provisions specified in the NuScale DCA for the functional qualification of the RSVs to be consistent with the NRC regulatory guidance and therefore to be acceptable.

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 (ADAMS Accession No. 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 (ADAMS Accession No. ML003708068); and SECY-95-132, with their applicable SRM, dated July 21, 1993; June 30, 1994; and June 28, 1995 (ADAMS Accession Nos. ML003708056, ML003708098, and ML003708019, respectively), and an NRC staff public memorandum dated July 25, 1995 (ADAMS Accession No. 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.

In NuScale DCA Part 2, Tier 2, Section 3.9.6.5 and Table 3.9-17 describe an augmented testing program for specific valves. In particular, Section 3.9.6.5 specifies that components not required by ASME OM Code, paragraph ISTA-1100, but with augmented quality requirements similar to those in paragraph ISTA-1100, are included in an augmented IST program. Section 3.9.6.5 notes that these components either provide a backup that is not safety related to a safety-related function or are valves that are not safety related that provide an augmented quality function. 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 granted relief in accordance with 10 CFR 50.55a(f), commensurate with the augmented requirements for those components. Section 3.9.6.5 indicates that DCA Part 2, Tier 2, Table 3.9-17, 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.

For reliance on specific components that are not safety related, NuScale submitted letters dated July 9, 2018 (ADAMS Accession No. ML18190A509) and November 20, 2018 (ADAMS

Accession No. ML18324A889), describing its augmented program for valves that are not safety related but are relied on to mitigate a DBE. The applicant also revised DCA Part 2 to specify that these valves (1) will have a proven design that demonstrates reliable operation based on operating experience in comparable systems or will be tested to prove the design under expected operating conditions, and (2) will be tested to the intent of the ASME OM Code as endorsed in 10 CFR 50.55a, or tested in accordance with any granted relief. The applicant also updated DCA Part 2, Tier 2, Table 3.2-1, to reference Section 3.9.6.5 and Table 3.9-17, in addition to Section 15.0.0.6.6, "Treatment of Nonsafety-Related Systems," for valves that are not safety related but are relied on to mitigate a DBE.

The staff reviewed the DCA to verify that the provisions for specific components indicated by NuScale that are not safety related but are relied on to mitigate a DBE were adequate. For example, DCA Part 2, Tier 2, Section 10.3, "Main Steam System," for secondary MSIVs and secondary MS isolation bypass valves, and Section 10.4, "Other Features of Steam and Power Conversion System," for FW regulation valves and secondary FW check valves indicate that these valves will be either a commercially available valve that uses a proven design and demonstrates reliable operation based on operating experience, or a design with no previous operating experience that may be proven through testing to demonstrate that the valve performs as expected at operating conditions. These sections also specify that the applicable valves will be periodically tested in accordance with the Augmented Valve Testing Program described in DCA Part 2, Tier 2, Section 3.9.6.5, with the testing requirements specified in DCA Part 2, Tier 2, Table 3.9-17. In addition, DCA Part 2, Tier 2, Table 3.2-1, in Note 6 references the provisions in DCA Part 2, Tier 2, Section 3.9.6.5, Section 15.0.0.6.6, and Table 3.9-17 for these specific valves. Consistent with NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff," issued November 1976 (ADAMS Accession No. ML13267A423), the NRC staff finds that DCA provisions provide reasonable justification for the capability of the specific valves that are not safety related to perform their intended functions as indicated in the accident analyses. Therefore, the staff finds that the applicant has identified an acceptable augmented testing program.

3.9.6.4.8 Initial Test Programs

NuScale DCA Part 2, Tier 2, 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. For example, the ITP tests include the following:

- DCA Part 2, Tier 2, Table 14.2-1, "Spent Fuel Pool Cooling System Test #1"
- Table 14.2-2, "Pool Cleanup System Test #2"
- Table 14.2-3, "Reactor Pool Cooling System Test #3"
- Table 14.2-6, "Pool Leak Detection System Test #6"
- Table 14.2-7, "Reactor Component Cooling Water System Test #7"
- Table 14.2-8, "Chilled Water System Test #8"
- Table 14.2-10, "Circulating Water System Test #10"
- Table 14.2-11, "Site Cooling Water System Test #11"
- Table 14.2-13, "Utility Water System Test #13"
- Table 14.2-14, "Demineralized Water System Test #14"
- Table 14.2-15, "Nitrogen Distribution System Test #15"

- Table 14.2-16, “Service Air System Test #16”
- Table 14.2-17, “Instrument Air System Test #17”
- Table 14.2-23, “Radioactive Waste Drain System Test #23”
- Table 14.2-27, “Main Steam System Test #27”
- Table 14.2-28, “Feedwater System Test #28”
- Table 14.2-29, “Feedwater Treatment System Test #29”
- Table 14.2-30, “Condensate Polishing System Test #30”
- Table 14.2-31, “Feedwater Heater Vents and Drains System Test #31”
- Table 14.2-32, “Condenser Air Removal System Test #32”
- Table 14.2-34, “Turbine Oil Storage System Test #34”
- Table 14.2-37, “Solid Radioactive Waste System Test #37”
- Table 14.2-39, “Boron Addition System Test #39”
- Table 14.2-53, “Process Sampling System Test #53”

The NRC staff has evaluated the adequacy of the ITP tests in Chapter 14 of this SER. 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 DCA Part 2, Tier 2, 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	DCA Part 2, Tier 2, Section
COL Item 3.9-4	A COL applicant that references the NuScale Power Plant design certification will submit a Preservice Testing program for valves as required by 10 CFR 50.55a.	3.9.6
COL Item 3.9-5	A COL applicant that references the NuScale Power Plant design certification will establish an Inservice Testing program in accordance with American Society of Mechanical Engineers Operation and Maintenance Code and 10 CFR 50.55a.	3.9.6
COL Item 3.9-6	A COL applicant that references the NuScale design certification will identify any site-specific valves, implementation milestones, and the applicable American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code (and ASME OM Code Cases) for the preservice and inservice testing programs. These programs are to be consistent with the requirements in the latest edition and addenda of the OM Code incorporated by reference in 10 CFR 50.55a in accordance with the time period specified in 10 CFR 50.55a before the scheduled initial fuel load (or the optional Cases listed in Regulatory Guide 1.192 incorporated by reference in 10 CFR 50.55a).	3.9.6.3
COL Item 3.9-8	A COL applicant that references the NuScale Power Plant design certification 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 3.9-9	A COL applicant that references the NuScale Power Plant design certification 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 DC 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 DCA 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 52.47 of the NRC regulations. The staff concludes that the NuScale DCA provides assurance that the functional design, qualification, and IST provisions for valves in the NuScale Power Plant referenced in DCA Part 2 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 120-month 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, information associated with this section is found in DCA Part 2, Tier 1, Sections 2.8, “Equipment Qualification,” and 3.14, “Environmental Qualification—Common Equipment.”

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.10, addresses the acceptance criteria, code 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 highly reliable electrical equipment.

The qualification of electrical equipment is performed according to the Institute of Electrical and Electronics Engineers (IEEE) Std. 344-2004. The qualification of active mechanical equipment is conducted according to ASME QME-1-2007. 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, DCA Part 2 describes the documentation of the equipment qualification records.

ITAAC: DCA Part 2, Tier 1, Section 2, Table 2.8-1, lists the equipment addressed by the ITAAC of Table 2.8-2. 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 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, Revision 1
- RG 1.92, Revision 3
- RG 1.100, Revision 3

3.10.4 Technical Evaluation

The staff performed its review of DCA Part 2, Tier 2, 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 3, issued March 2007. The SRP contains six acceptance criteria. The following subsections discuss the staff's review of the consistency of DCA Part 2 with these criteria.

The seismic qualification methodology described in DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.10.2.1, “Qualification by Testing,” 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 in DCA Part 2, Tier 2, Section 3.10.2.1, is consistent with the criterion. Moreover, the applicant will use testing or analysis to qualify equipment as stated in DCA Part 2, Tier 2, Section 3.10.2.1. The staff finds that this is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.i.

ii. SRP Section 3.10, Section 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-1-2007 as described in RG 1.100 for the seismic qualification of active mechanical equipment. Following ASME QME-1-2007 as described in RG 1.100 is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.ii.

iii. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.10.1.2, “Performance Requirements for Seismic Qualification,” states, “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 that DCA Part 2, Tier 2, Section 3.10.1.2, is 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. DCA Part 2, Tier 2, 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, the malfunction of which can be caused by 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."

DCA Part 2, Tier 2, Section 3.10.2.1, is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.v, and, for this reason, is acceptable.

vi. DCA Part 2, Tier 2, 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."

DCA Part 2, 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. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.10.2.1 acceptable because it is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.vii.

viii. DCA Part 2, Tier 2, 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 that DCA Part 2, Tier 2, Section 3.10.3, is acceptable because it is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.viii.

ix. DCA Part 2, Tier 2, Section 3.10.2.1, states, "the loads include forces imposed by piping onto the equipment." The staff finds that DCA Part 2, Tier 2, Section 3.10.2.1, is 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. 344-2004, 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 unnecessary.

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." DCA Part 2, Tier 2, Section 3.10.2.1, states that "IEEE Std. 344-2004 applied when qualification is by test. The guidance of IEEE Std. 344-2004 allows this option." The staff finds that following IEEE Std. 344-2004 as stated in DCA Part 2, Tier 2, 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." RG 1.61 is referenced in the seismic topical report incorporated by reference into DCA Part 2, Section 3.11.2.1. Section 3.7.1 of this SER documents the staff's review of the damping values and finds that Acceptance Criterion II.1.A.xiii is satisfied. DCA Part 2, Tier 2, Section 3.7.1.2, states that the damping values used for the analysis of the seismic Category I and II SSCs are based on RG 1.61, Revision 1, and provides both SSE and OBE damping values in DCA Part 2, Tier 2, Table 3.7.1-6. The staff determined that the applicant has used values of critical damping that are consistent with those in RG 1.61. The staff finds this acceptable for use in any subsequent dynamic analysis for the NuScale design.

xiv. DCA Part 2, Tier 2, 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 344-2004 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-1-2007 as described in RG 1.100 are also used for the seismic qualification of active mechanical equipment, as stated in DCA Part 2, Tier 2, Section 3.10.1.1. Subsection QR-7312, "Dynamic Loading," of ASME QME-1-2007 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-1-2007 and IEEE Std. 344 will be used, and they contain analysis approaches consistent with criterion 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. DCA Part 2, Tier 2, 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." DCA Part 2, Tier 2, Section 3.10.3, states, "The qualification of supports for electrical equipment and instrumentation, which includes 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 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.

iv. 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. DCA Part 2, Tier 2, 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 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)." DCA Part 2, Tier 2, Section 3.10.1.1, notes that the requirements of IEEE Std. 344-2004 endorsed by RG 1.100, Revision 3, will be implemented. The use of these updated standards is acceptable to the staff as they have been endorsed in the later revision of SRP Section 3.10 (Revision 4, issued December 2016). Therefore, the applicant has satisfied 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 DCA Part 2 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 DCA Part 2 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 DCA Part 2, Tier 2, Section 3.10.1.2, the qualification of instrumentation is addressed in DCA Part 2, Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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)

In DCA Part 2, Tier 2, Section 3.10.4, “Test and Analysis Results and Experience Database,” COL Item 3.10-2, the applicant stated that a COL applicant that references the NuScale Power Plant DC will develop the equipment qualification database and ensure that equipment qualification record files (EQRFs) are created for the SSCs that require seismic qualification. Section 3.10.4 states that the experience database containing plant EQRF data is maintained for the life of the plant.

The results of seismic qualification testing and analysis, according to the criteria in DCA Part 2, Tier 2, Sections 3.10.1, 3.10.2, and 3.10.3, are included in the corresponding EQRFs. The EQRFs are created and maintained during the equipment selection and procurement phase for the equipment requiring qualification. The EQRFs contain a detailed description of the equipment and its support structures, qualification methodology, test and analysis results. The EQRFs are updated and modified as new tests and analyses are performed. The experience database containing plant EQRF data is maintained for the life of the plant.

The EQRFs should 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
- suitability for inspection
- 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).

The applicant, in DCA Part 2, Tier 2, 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 non-active mechanical components, the structural integrity of which is maintained to perform their safety function.

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-1-2007 is used with the exceptions noted in RG 1.100, Revision 3, for the qualification of active mechanical equipment. The staff conducted an audit to verify that the actuated valves were tested as active equipment in accordance with DCA Part 2, Tier 2 Section 3.10, from November 8, 2017, to January 31, 2018. In the audit, the staff found that seismic testing provisions specified in DCA Part 2 had not been completely and consistently incorporated into the design specifications, as summarized in the audit report dated October 25, 2018 (ADAMS Accession No. ML18173A291). The inconsistencies consisted of not addressing the following in entirety in each design specification:

- application of input motion
- TRS
- 5 OBE+1 SSE or 2 SSE requirement
- the requirement to have seismic testing

The staff found that the design specifications were incomplete and required revision to address the deficiencies indicated in the audit report. NuScale satisfactorily addressed these items, as confirmed by the staff in the followup audit summary, dated December 20, 2019 (ADAMS Accession No. ML19331A397).

3.10.4.6 Equipment Qualification Program Implementation Documentation (SRP 3.10, Acceptance Criterion 6)

An EQRF is developed for each piece of electrical equipment and instrumentation classified as seismic Category I. DCA Part 2, Tier 2, 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 EQRF defines 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.

In accordance with COL Item 3.10-1, a COL applicant that references the NuScale Power Plant DC will develop and maintain a site-specific seismic and dynamic qualification program. Each EQRF contains a section specifying performance requirements. This specification establishes the safety-related functional standards of the 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. DCA 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. The staff found that DCA 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 Tier 2, 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 DCA

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 DCA 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

Table 3.10.5-1 lists COL information item numbers and descriptions related to DCA Part 2, Tier 2, Section 3.10.

Table 3.10.5-1: NuScale COL Information Items for Section 3.10

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.10-1	A COL applicant that references the NuScale Power Plant design certification will develop and maintain a site-specific seismic and dynamic qualification program.	3.10
COL Item 3.10-2	A COL applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure	3.10.4

	equipment qualification record files are created for the SSCs that require seismic qualification.	
COL Item 3.10-3	A COL applicant that references the NuScale Power Plant design certification will submit an implementation program for Nuclear Regulatory Commission approval prior to the installation of the equipment that requires seismic qualification.	3.10.4

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 the equipment must be environmentally qualified to meet its intended design function related to safety.

GDC 4 requires the EQ of mechanical and electrical equipment. The equipment must 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.

DCA Part 2, Tier 2, Section 3.11, 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 DCA Part 2, Tier 2, 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 52.47(a)(13), 10 CFR 50.49, and GDC 4. The staff reviewed 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.

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

DCA Part 2, Tier 1: Sections 2.1, 2.8, and 3.14 of DCA Part 2, Tier 1, contain the requirements for EQ of electrical and mechanical equipment. The Tier 1 requirements in Section 2.1 are related to the protection of safety-related SSCs against dynamic and environmental effects, as specified in GDC 4. Tier 1, Section 2.8, provides EQ of equipment specific to each NPM. Tier 1, Section 3.14, provides EQ of equipment shared by NPMs.

DCA Part 2, Tier 2: The applicant provided a Tier 2 description in Section 3.11, summarized here, as follows.

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, DCA Part 2, Tier 2, Appendix 3C, describes the methodology used by the applicant to environmentally qualify electrical and mechanical equipment. DCA Part 2, Tier 2, Table 3.11-1, “List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments,” provides the list of equipment located in a harsh environment to be environmentally qualified.

In Tier 2, Table 3.11-2, “Environmental Qualification Zones—Reactor Building,” the applicant identified areas of the plant that could be subjected to a harsh environment following an accident. Further, Tier 2, 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.

In DCA Part 2, Tier 2, 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-1-2007. In DCA Part 2, Tier 2, 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”; in DCA Part 2, Tier 2, Section 3.9.6 and Section 3.10, respectively, address these topics.

ITAAC: ITAAC related to EQ are addressed in SER Section 14.3.6.

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

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

Topical Reports:

- TR-0915-17565-NP-A, Revision 4, “Accident Source Term Methodology Topical Report” (ADAMS Accession No. 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, 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, 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, requires that sufficient records be maintained to furnish evidence of activities affecting quality.

- 10 CFR 52.47 states, in part, that the Commission will require, before certification of a design, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

- 10 CFR 52.47(a)(13) requires an application for a standard DC 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 standard DC 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 (ADAMS Accession No. 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 (ADAMS Accession No. ML031480402), Category I guidance may be used if relevant guidance is not provided in RG 1.89.

- RG 1.63, Revision 3, issued February 1987 (ADAMS Accession No. ML003740219), endorses IEEE Std. 317.

- RG 1.73, Revision 1, "Qualification Tests for Safety-Related Actuators in Nuclear Power Plants," issued October 2013 (ADAMS Accession No. 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, Revision 4, issued June 2006 (ADAMS Accession No. ML061580448) provides guidance acceptable to the staff for the EQ of the 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 3, endorses, with exceptions and clarification, ASME Standard QME-1-2007 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 (ADAMS Accession No. 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 (ADAMS Accession No. 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 1, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," issued October 2003 (ADAMS Accession No. ML032740277), 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 (ADAMS Accession No. 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 (ADAMS Accession No. 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-1-2007 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 DCA Part 2, Tier 2, Section 3.11, which describes the applicant's approach to conforming with 10 CFR 50.49 for the environmental qualification (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 DCA Part 2, Tier 2, 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) are 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

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 after design-basis events (DBEs) to ensure that certain functions are accomplished

- 10 CFR 50.49(b)(2)—electrical equipment that is not safety related, the failure of which, under the postulated environmental conditions, could prevent satisfactory performance of the safety functions of the safety-related equipment

- 10 CFR 50.49(b)(3)—certain PAM equipment and RG 1.97

The applicant explained in DCA Part 2, Tier 2, 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 accident (DBA) or infrequent event that produces a harsh environment, (2) equipment with a design function related to safety that is relied on for its ability to achieve or maintain a safe-shutdown condition for a DBA or infrequent event that produces a harsh environment, and (3) certain PAM equipment.

The staff reviewed DCA Part 2, Tier 2, 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.

In DCA Part 2, Tier 1, Table 2.8-1, the applicant indicated that some EPAs are not Class 1E, including the I&C equipment, PZR heater power, and CRD power EPAs. The staff confirmed that the EPAs, including those for I&C equipment, PZR heater power, and CRD power, are appropriately categorized. In addition, DCA Part 2, Tier 2, Section 3.11.7 and Note 6 of Table 3.11-1 addresses EQ of items such as cables, connectors, electrical splices, conduit seals, thread sealants, terminal blocks, or lubricants through commodities.

The equipment important to safety that is subject to EQ is divided in two plant areas, as described in DCA Part 2, Tier 2, Appendix 3C, Section 3C.3, "Introduction." These plant areas are the RXB and the CRB. The CRB is considered to have a mild environment. DCA Part 2, Tier 2, Table 3.11-2, identifies the RXB rooms that are subject to a harsh environment.

The service condition environments are divided into two categories: (1) harsh environment and (2) mild environment. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.”

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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.11.1.2. The parameters include pressure, radiation, temperature, chemical spray, humidity, submergence, and electromagnetic and radiofrequency interference in specific plant building and room locations. 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 DCA Part 2, Tier 2, Appendix 3C, Section 3C.4.2, “Aging.” 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 DCA Part 2, Tier 2, Section 3.11.5.1, “Chemical Environments,” which states the following:

> Applicable chemical environments are defined in Appendix 3.C for normal and abnormal operating conditions. The chemical environments from the most limiting design basis event is also considered in the qualification of the equipment and presented in Appendix 3.C.

Appendix 3.C, Table 3C-6, “Normal Operating Environmental Conditions,” Note 4, states, “The boron concentration in the pool areas will be nominally 1800 ppm. EPRI primary water chemistry guidelines show the pH of a pool with 1800 ppm boron concentration to be 4.75.” Appendix 3.C, Table 3C-7, “Design Basis Event Environmental Conditions,” Note 4, states, “the CNV post-accident pH for any postulated accident that results in core damage is 6.9 at 1000 ppm boron concentration and 8.3 at 200 ppm boron concentration. These values remain essentially unchanged between 25C and 200C.” The staff finds that the incorporation of this information provides reasonable assurance that all environmental parameters are considered, in conformance with 10 CFR 50.49(e).

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 DCA Part 2, Tier 2, Appendix 3C, Section 3C.4.3, “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 DCA Part 2, Tier 2, Appendix 3C, Section 3C.4.3, 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). DCA Part 2, Tier 2, Section 3.11.1.2, states the following:

> Service conditions are the actual environmental, physical, mechanical, electrical, and process conditions experienced by equipment during service. Plant operation includes both normal and abnormal operations. Abnormal operation occurs during plant transients, system transients, natural phenomena, or in conjunction with certain equipment or system failures.

In addition, DCA Part 2, Tier 2, Appendix 3C, Table 3C-5, "EQ 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).

DCA Part 2, Tier 2, Appendix 3C, Section 3C.6, "Qualification Methods," 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).

DCA Part 2, Tier 2, 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 52.47(a)(13). The table describes the equipment, its location, EQ environment, operational time, EQ category, and PAM.

Based on the staff's review of the applicant's EQ program described in DCA Part 2, Tier 2, 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 52.47(a)(13) because the DCA 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 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 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. DCA Part 2, Tier 2, Section 3.11, states that electrical equipment identified in DCA Part 2, Tier 2, Table 3.11-1, will be environmentally qualified using the guidance in IEEE Std. 323-1974. DCA Part 2, Tier 2, Section 3.11, states that equipment located in a harsh environment will be qualified in accordance with IEEE Std. 323-1974.

DCA Part 2, Tier 2, Section 3.11.2.1, “Environmental Qualification of Electrical Equipment,” states that PAM equipment will be environmentally qualified in accordance with RG 1.97, Revision 4, which endorses IEEE Std. 497-2002, “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.

Based on the review discussed above, the staff concludes that DCA Part 2 conforms to the RG 1.89 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 52.47(a)(3), an application for a standard DC 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 DCA Part 2, Tier 2, Section 3.11.3, “Qualification Test Results,” 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, 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 DCA Part 2, Tier 2, Section 3.11:

> Components in the scope of this Section that are subject to environmental design and qualification 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.

The qualification approach complies with GDC 4. DCA Part 2, Tier 2, Appendix 3C, 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 not subject to a harsh 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.

DCA Part 2, Tier 2, Appendix 3C, Section 3C.4.2, 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 52.47(a)(19), a DCA 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. The applicant stated that safety-related I&C systems are designed in compliance with Criterion III. The staff finds that the applicant complies with the requirements of Criterion III because the applicant followed the methodology for testing requirements in IEEE Std. 323-1974, which is endorsed by RG 1.89.

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, 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.

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. DCA Part 2, 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 3.C. Section 3C.8 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.

3.11.4.2 Environmental Qualification of Mechanical Equipment

The objective of this review is to determine if DCA Part 2 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 DCA Part 2, Tier 2, 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 specified under Section 3.11.3 above to support the acceptability for reference in a COL application. The staff issued several RAIs to NuScale to resolve staff questions on the information provided in the original DCA Part 2 submittal. In response to the RAIs, NuScale revised DCA Part 2 to clarify specific information with respect to EQ of mechanical equipment. In this section, the staff focuses on the revised DCA Part 2 and its compliance with the applicable NRC regulations and guidance rather than discussing each RAI and NuScale response.

3.11.4.2.1 Identify Safety-Related Mechanical Equipment Located in Harsh or Mild Environment Areas and Operating Times

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.2, describes the safety classification of SSCs.

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.11-1. That table also lists the operating times for mechanical equipment located in harsh environments.

DCA Part 2, Tier 2, Appendix 3C.4, "Qualification Criteria," states that mechanical equipment required to perform a design function related to safety located in mild environments is listed in the associated EQRF.

DCA Part 2 identifies safety-related mechanical equipment located in harsh environments but does not identify safety-related mechanical equipment located in mild environments. The applicant provided supplemental information by letter dated August 17, 2017 (ADAMS Accession No. ML17229B488), stating that the NuScale design does not have any safety-related mechanical equipment located in mild environments that may contain nonmetallic parts. The applicant also stated that if any new safety-related mechanical equipment that may contain nonmetallic parts is located in mild environments, it will be environmentally qualified in accordance with ASME QME-1-2007, as accepted in RG 1.100, Revision 3. The staff finds this information acceptable because it is consistent with the guidance in DSRS Section 3.11 to identify safety-related mechanical equipment located in mild and harsh environments that may contain nonmetallic parts and to specify that such equipment will be qualified in accordance with ASME QME-1-2007, as accepted in RG 1.100, Revision 3. ASME QME-1-2007 includes Appendix QR-B, "Guide for Qualification of Nonmetallic Parts."

3.11.4.2.2 Identify Nonmetallic Subcomponents of Mechanical Equipment

DCA Part 2, Tier 2, Section 3.11.6, "Qualification of Mechanical Equipment," states that DCA Part 2, Tier 2, 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.

For mechanical equipment located in a mild environment, the applicant provided supplemental information as referenced in a letter dated August 17, 2017 (ADAMS Accession No. ML17229B488), stating that the NuScale design does not currently have any safety-related mechanical equipment located in mild environments that may contain nonmetallic parts.

The staff finds that the identification of nonmetallic subcomponents of mechanical equipment in Tier 2, Section 3.11 and Appendix 3C, 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

DCA Part 2, Tier 2, Section 3.11.1.2, states that environmental conditions considered in the design of the NuScale reactor include AOOs and normal, accident, and postaccident environmental conditions. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.11 and Appendix 3C, describe the external environmental conditions. The internal operational service conditions, such as system or component operating temperatures, are identified in the section of DCA Part 2 that applies to the system or component.

The staff finds that the identification of environmental conditions and process parameters for nonmetallic parts of mechanical equipment in DCA Part 2, Tier 2, Section 3.11 and Appendix 3C, is consistent with the guidance in DSRS Section 3.11 and is acceptable.

3.11.4.2.4 Identify Nonmetallic Material Capabilities

DCA Part 2, Tier 2, Section 3.11.6, states that nonmetallic parts are designed to perform their required function during normal, abnormal, and accident conditions including the effects of fluid medium on the environmental conditions. The nonmetallic parts are designed to be capable of performing their intended function for the environmental and service parameters identified in DCA Part 2.

The staff finds that the identification of nonmetallic material capabilities in DCA Part 2, Tier 2, 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

In DCA Part 2, Tier 2, Section 3.11.6 and Appendix 3C.4 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, as further explained in a NuScale letter dated October 17, 2018 (ADAMS Accession No. ML18290A557).

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-1-2007, 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-1-2007, 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.

DCA Part 2, Tier 2, 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.

In DCA Part 2, Tier 2, Section 3.11.6 and Appendix 3C.4 state that mechanical 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 EQRF lists the environmental parameters and the qualification process.

In a letter dated August 17, 2017 (ADAMS Accession No. ML17229B488), the applicant stated that the NuScale design does not have any safety-related mechanical equipment that may contain nonmetallic parts located in mild environments and stated that if any new safety-related mechanical equipment that may contain nonmetallic parts is located in mild environments, it will be environmentally qualified in accordance with ASME QME-1-2007, as accepted in RG 1.100, Revision 3. The staff finds the applicant's information acceptable because qualification of nonmetallic parts of active mechanical equipment in accordance with ASME QME-1-2007, Appendix QR-B, is consistent with the guidance in DSRS Section 3.11 and meets the regulatory requirements in 10 CFR Part 50, Appendix A, GDC 1, 2, and 4, and 10 CFR Part 50, Appendix B, Criterion III and XI.

3.11.4.2.6 Design Specification Audit

DCA Part 2, Tier 2, Appendix 3C, Section 3C.5, "Design Specifications," states that the equipment design specification identifies the applicable codes and standards, required operating time, performance requirements, design functions related to safety, operational service conditions, environmental service conditions, accepted methods of qualification, and acceptance criteria. The design specification also provides the basis for establishing the EQ of the specific equipment for the family of equipment.

The NRC regulations in 10 CFR 52.47 state, in part, the following:

> [I]nformation submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

The staff audited the design specifications as discussed in the introduction of 10 CFR 52.47 for the EQ of nonmetallic parts of safety-related mechanical equipment. The staff closed its review of specific audit items and identified some of the audit items as requiring resolution. From July 30, 2019, to September 24, 2019, the staff conducted a closeout audit of the revised NuScale design specifications to verify that the remaining followup items had been resolved. The staff issued a closeout audit report, dated December 20, 2019, describing the completion of the followup items for the NuScale design specifications (ADAMS Accession No. ML19331A397). Based on its audit review, the NRC staff finds, as required by the NRC regulations in 10 CFR 52.47, that information normally contained in procurement specifications and construction and installation specifications are sufficiently complete for the Commission to make a safety determination regarding the NuScale DCA.

3.11.4.3 Environmental Qualification of the Radiation Environment

The staff reviewed DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.11 and Appendix 3C. Guidance for the staff's evaluation appears in Revision 3 of Section 3.11 of the NuScale DSRS, Revision 1; in RG 1.89; and in Appendix I to RG 1.183.

As discussed in DCA Part 2, Section 3.11.5.2, 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 DCA Part 2, Tier 2, Table 3C-2 and Table 3C-3, 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, in that environmental conditions, including all significant types of radiation during normal and accident conditions, are considered. Based on this, the staff finds this approach acceptable.

However, DCA Part 2, Tier 2, Section 3.11.1.2, indicates that only gamma doses are considered in determining the radiation environment. The staff evaluated the stated conditions and determined that Tables 3C-2 and 3C-3 specify that the TID was used in the classification of areas (not only the gamma dose). Additionally, since Sections 3.11.5.2, 3C.5.1, and 3C.5.5 clarify that gamma, neutron, and beta radiation were considered in the TID, the staff determined that this information in DCA Part 2, Tier 2, Section 3.11.1.2, has no negative impact on the classification of equipment or the TID used for qualifying equipment and therefore is 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. DCA Part 2, Tier 2, Chapter 12, "Radiation Protection," provides information about the normal operation source terms used to develop the normal operation dose rate information provided in DCA Part 2, Appendix 3C, Table 3C-6. Chapter 12 of this SER presents the staff's review of these source terms.

DCA Part 2, Tier 2, Appendix C, Table 3C-6, provides the normal operation TID in Zones A through N, which includes all areas of the RXB that have environmentally qualified equipment located in a harsh environment. This includes equipment inside of containment and under the bioshield. The staff reviewed the normal operation TIDs based on the source terms, radiation protection design features, and radiation zoning provided in Chapter 12 of the FSAR and the neutron flux spectra at the inner surface of the CNV head provided by letter dated July 22, 2019 (ADAMS Accession No. ML19203A321). The neutron information provided by the applicant was generally consistent with neutron spectral and fluence data generated by the NRC staff performing calculations using ORIGEN, SCALE and MCNP6.2. Since the information provided by the applicant and the calculations performed by the NRC staff were generally consistent with the neutron TID provided in DCA Part 2, Tier 2, Table 3C-6, the NRC staff concludes that there is reasonable assurance that the neutron radiation environments in the upper RPV, the upper CNV, and under the bioshield have been adequately described by the applicant. The staff reviewed the normal operation TIDs for certain areas in Table 3C-6, and found them to be consistent with the information in Chapter 12 and the neutron flux spectra. As a result, the staff found the normal operation dose rates to be acceptable. The staff notes that the gamma and neutron doses assume the presence of the 1.5-meter (5-ft)-thick equivalent concrete shielding for the walls with major penetrations for the MS and feed lines and ventilation lines outside the nuclear power module. The shielding for these major penetrations will be provided by the COL applicant, as discussed in Chapter 12, Section 12.3, of this SER.

Regarding accident radiation doses to equipment, the accident source term methodology topical report TR-0915-17565, Revision 4 (ML20057G132), provides information on accident source terms and equipment qualification dose methodology for areas inside containment and under the bioshield, in combination with the accident source term information in DCA Part 2, Tier 2, 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.

The staff reviewed the potential accident scenarios in the NuScale FSAR and determined that the only accident scenarios in the NuScale design that could result in core damage were the result of beyond design-basis events. Therefore, the staff has determined that core damage events are not DBAs that need to be evaluated to address 10 CFR 50.49(e)(4) in the NuScale design (as described in SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," dated August 16, 2019 (ADAMS Accession No. ML19107A455)). Similar to 10 CFR 50.49(e)(4), since LOCAs and other DBEs do not result in significant core damage in the NuScale design and 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, the staff has determined that core damage need not be assumed in addressing the radiological equipment qualification aspects of 10 CFR Part 50, Appendix A, GDC 4. 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 52.47(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 (ADAMS Accession No. ML003707849), SECY-93-087 (ADAMS Accession No. ML003760768), and the associated SRM for SECY-93-087 (ADAMS Accession No. ML003708056). The radiological conditions associated with equipment survivability are discussed in DCA Part 2, Tier 2, Section 15 and Section 19.

Since there are no DBAs that result in core damage 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 EQ 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 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 (ADAMS Accession No. ML19290F633) dated October 2019, for TR-0915-17565.

While TR-0915-17565 includes an acceptable methodology for developing the design-basis iodine spike accident source term and dose rates, DCA Part 2, Tier 2, Table 12.2-34, includes the design-basis iodine spike source term used in developing the TIDs in DCA Part 2, Tier 2, Table 3C-8. DCA Part 2, Tier 2, Section 12.2.1.13, specifies that the specific concentration values in Table 12.2-34 are in excess of the values that would be calculated using the methodology in TR-0915-17565. To understand the basis for the values in Table 12.2-34 and the EQ source term methodology in TR-0915-17565, as well as to assess other radiation protection related material, the staff conducted an audit related to the iodine spike source term and EQ doses (see ADAMS Accession No. ML19308A944 for the audit report). During the audit, it was determined that the source term provided in Table 12.2-34 was based on higher reactor coolant radionuclide concentrations than the design-basis reactor coolant concentration in DCA Part 2, Tier 2, Table 11.1-4. In addition, the source term in Table 12.2-34 included other conservatisms, which also made the source term higher than what would be calculated using the methodology in TR-0915-17565. This included assuming a crud burst for corrosion products and increasing the radionuclide inventory of some daughter products to account for uncertainty. These and other conservatisms resulted in the DCA Part 2, Tier 2, Table 12.2-34, source term being significantly higher (i.e., more conservative) than what would be expected using the methodology described in TR-0915-17565. Since the source term was significantly more conservative than methodology approved in TR-0915-17565, the staff found the source term to be acceptable.

In addition, in reviewing the accident TIDs provided in DCA Part 2, Tier 2, Table 3C-8, "Limiting Design Basis Accident EQ Radiation Dose" the staff noted that the source term and the doses inside containment and under the bioshield were significantly higher than those the staff estimated using the Table 12.2-34 source term and the methodology in TR-0915-17565. Since higher doses would result in equipment being designed to withstand higher radiation levels than what may actually be there, the staff found the TID values used for equipment qualification for areas inside containment and under the bioshield, provided in Table 3C-8, to be acceptable. More information on the staff's review of the Table 12.2-34 source term and Table 3C-8 accident TIDs can be found in the staff's audit report. For areas other than inside of containment and under the bioshield, the staff determined that the DBA dose is expected to be low because DBAs do not result in core damage in the NuScale design 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 radiation Zone M 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.

In DCA Part 2, Section 3.11.1.2 and Appendix 3C specify that synergistic effects of environmental conditions (such as the effects of radiation and temperature) are also considered for both normal and accident conditions when these effects are significant. 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.

In addition, the NRC staff looked at the expected radiation environment of components, particularly nonsafety-related components, located within the CNV. The NRC staff evaluated these components because they would not be within the EQ program (10 CFR 50.49) and may not be considered within the scope of GDC 4. The NRC staff concern was that radiation-induced degradation of these components might result in a failure mechanism during an accident, which could impact a critical safety function (e.g., radiation degradation of materials that resulted in unexpected debris generation that then reduced core cooling during an accident). Some examples of the types of components the NRC staff considered are the reactor component cooling water (RCCW) flexible hoses to the control rod drive mechanism (CRDM) motors and magnets, the RCCW thermal relief valves, and power and signal cables for the individual rod position indicator system coils. This type of information was made available to the NRC staff during a Chapter 12 Phase I audit (see "Audit Report—Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design" (ADAMS

Accession No. ML18124A182)) and in a CRDM audit (see “U.S. Nuclear Regulatory Commission Staff Report of Regulatory Audit for NuScale Power, LLC, Design Certification Application Final Safety Analysis Report Sections 3.9.4 and 4.6,” dated December 25, 2017 (ADAMS Accession No. ML17331A357)). As part of the audit, the NRC staff looked at the radiation environment inside of the CNV and the location of nonsafety-related components. In the response to a question dated June 14, 2018 (ADAMS Accession No. ML18165A464), the applicant revised DCA Part 2, Tier 2, Section 3.11.6, to state that there would be no environmentally induced debris inside the CNV that would interfere with the proper functioning of the ECCS. The NRC staff finds this response acceptable because the applicant stated that there will be no nonsafety-related equipment within the CNV that could interfere with the operation of the ECCS.

DCA Part 2, Section 3.11.5.2, specifies that “the radiation doses are continuously monitored during plant life and compared to the calculated doses. If the measured doses are higher than the calculated doses, the EQ Master List will be revised if an affected mild environment becomes harsh.” In addition, as discussed in DCA Part 2, 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. To address the monitoring of equipment located in harsh environments to ensure that the dose requirements are not exceeded, Section 3.11.5.2 includes the following COL item:

> A COL applicant that references the NuScale Power Plant DC will ensure the Environmental Qualification Program cited in COL Item 3.11-1 includes a description of how equipment located in harsh conditions will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh environments will remain qualified if the measured dose is higher than the calculated dose.

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.

DCA Part 2, Appendix 3C, Section 3C.4.2, 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 equipment qualification 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 DCA Part 2, Tier 2, Table 1.8-2.

Table 3.11-1: NuScale COL Information Items for Section 3.11

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.11-1	The COL applicant that references the NuScale Power Plant design certification 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	The COL applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structure, system, and components that require environmental qualification.	3.11.3
COL Item 3.11-3	A COL applicant that references the NuScale Power Plant design certification will implement an equipment qualification operational program that incorporates the aspects 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.	3.11.7
COL Item 3.11-4	A COL applicant that references the NuScale Power Plant design certification will ensure the environmental qualification program cited in COL Item 3.11-1 includes a description of how equipment located in harsh conditions will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh environments will remain qualified if the measured dose is higher than the calculated dose.	3.11.5.2

3.11.6 Conclusion

As described above, the staff has reviewed all of the relevant information applicable to DCA Part 2, Tier 2, 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 52.47(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 also reviewed the COL information items in DCA Part 2, Tier 2, Table 1.8-2.

The staff concludes that the provisions in the NuScale DCA and TR-0915-17565 for the EQ of mechanical and electrical equipment in the NuScale design are acceptable and meet the applicable NRC requirements and are consistent with guidance. This conclusion is based on the applicant having specified provisions in DCA Part 2, Tier 2, 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 nonseismic 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 nonseismic Category I piping with seismic Category I piping. The following sections of this report provide the staff's evaluation of the adequacy of the DCA Part 2, Revision 3, 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Chapter 1, "Introduction," states that a graded approach is used, based on the level of design information, which is proportional to the safety significance of the SSC being addressed, and that the information presented in Tier 1 is consistent with the information presented in Tier 2.

DCA Part 2, Tier 2: In DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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" (ADAMS Accession No. ML14065A067). The paper's graded approach to the piping analysis for DCAs is consistent with the Commission's direction in SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," dated November 8, 1990 (ADAMS Accession No. 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 DC 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.

ITAAC: ITAAC for the NPM appear in Tier 1, Table 2.1-4. ITAAC 1 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: The TR for this area of review is TR-0916-51502-P, Revision 2, "NuScale Power Module Seismic Analysis," issued April 2019 (ADAMS Accession No. ML19093B850), which is incorporated by reference in accordance with DCA Part 2, Tier 2, Section 1.6 and Table 1.6-2.

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 52.47(a)(22), which requires that a DCA include 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 (ADAMS Accession Nos. ML003707889and ML003707892, respectively)
- SECY-93-087 and its associated SRM (ADAMS Accession Nos. ML003708021 and ML003708056 respectively).
- DSRS Section 3.7.2, Revision 0, “Seismic System Analysis,” issued June 2016 (ADAMS Accession No. ML15355A389)
- DSRS Section 3.7.3, Revision 0, “Seismic Subsystem Analysis,” issued June 2016 (ADAMS Accession No. 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 (ADAMS Accession No. 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 (ADAMS Accession No. ML14043A231)
- NRC white paper, “Piping Level of Detail for Design Certification,” dated March 4, 2014 (ADAMS Accession No. 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 a DC applicant as detailed in the white paper and industrial practice and programs.

The staff issued several RAIs to NuScale to resolve staff questions on the information provided in the original DCA Part 2 submittal. In response to the RAIs, NuScale revised DCA Part 2, to clarify specific information with respect to the structural design of piping and pipe supports. In this section, the staff focuses on the revised DCA Part 2 and its compliance with the applicable NRC regulations and guidance rather than discussing each RAI and NuScale response.

In DCA Part 2, Tier 2, 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 reactor coolant pressure boundary (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 decay heat removal system (DHRS) lines. For ASME Class 1 piping detailed design analysis, the applicant has chosen the CVCS RCS discharge line from the RPV nozzle connection to the first anchor. This line is chosen because it is representative of the NuScale ASME Class 1 piping with respect to loadings and fatigue usage and is longer than the other Class 1 lines, with more seismic supports and longer spans between restraints and, therefore, is more challenging for structural piping analysis. For ASME Class 2 detailed piping design analysis, it has chosen the FW and MS lines. The two FW lines are analyzed from the RPV nozzle connection to anchor supports on the outboard side of the reactor bay wall. The MS lines are analyzed from the RPV nozzle connection to anchor supports on the outboard side of the reactor bay wall. The applicant selected the MS and FW lines for Class 2 detailed design pipe stress analysis because they experience bounding loads for the Class 2 systems. The FW and the MS lines are recommended by the staff’s white paper and therefore are acceptable choices. The applicant further stated that the preliminary and detailed piping analyses use all applicable loads mentioned in DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 applicant, in letter No. RAIO-0716-66374, dated July 22, 2019 (ADAMS Accession No. ML19203A342), provided tabulated pipe stress summaries for sections of the MS, FW, and RCS discharge lines. These lines are in the scope for detailed final design piping analysis, discussed in the “graded level of detail approach in piping design” in DCA Part 2, Tier 2, Section 3.12.1, and in Section 3.12.2 of this SER. The calculated pipe stresses and fatigue CUFs (applicable to Class 1 piping), meet allowable limits in ASME BPV Code, Section III and therefore are acceptable. Based on its review, as discussed above, the staff finds that the applicant has successfully completed the detailed as-designed piping analyses for the in-scope piping.

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 DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, 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, 2013 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

DCA Part 2, Tier 2, 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 1 in Tier 1, Table 2.1-4. 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 DCA Part 2, Tier 2, 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. In addition, according to DCA Part 2, Tier 2, COL Item 3.9-2, a COL applicant that references the NuScale Power Plant DC will develop design specifications and design reports in accordance with the requirements outlined under ASME BPV Code, Section III.

Based on its review, the staff finds the DCA Part 2 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 DCA Part 2, Section 3.12.2.2, “Codes and Standards,” 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 DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.3.2.1, “Development of In-structure Response Spectra,” the ISRS are developed using the methods and guidance of RG 1.122. Section 3.12.3.2.1 also indicates that if the response spectra broadening is not determined using RG 1.122 procedures, it will be peak broadened by +/-15 percent. DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, 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. 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

In DCA Part 2, Tier 2, 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 (ADAMS Accession No. 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 DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.3.3, "Independent Support Motion Method," and in TR-0916-51502-P, Revision 2, 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. DCA Part 2, Tier 2, Section 3.12.3.3, 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

DCA Part 2, Tier 2, 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 showed 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.

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.

In DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, 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 Small-Bore Piping Method

DCA Part 2, Tier 2, Chapter 1, shows that for preparation, DCA Part 2 uses SRP guidance. DCA Part 2, Tier 2, Table 1.9-3, “Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS),” shows that DCA Part 2, Tier 2, Section 3.12.3, “Piping Analysis Methods,” is prepared in conformance with SRP Section 3.12, Subsection II.A. SRP Section 3.12, Acceptance Criterion II.A.vi, discusses small-bore piping and recommends (by referring to SRP Section 3.9.2, Acceptance Criterion II.2.A) for its evaluation either the dynamic analysis method or the equivalent static load method. DCA Part 2, Tier 2, Section 3.12, makes no distinction between small-bore piping and large-bore piping. DCA Part 2, Tier 2, Section 3.12.3.7, “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.

According to DCA Part 2, Tier 2, Section 3.12, the design code on record for the NuScale ASME Class 1, 2, and 3 piping is specified as the ASME BPV Code, Section III, 2013 Edition (no addenda). In DCA Part 2, Tier 2, Section 3.12.2.1, the applicant stated that the conditions of use for ASME BPV Code, Section III, are applied in accordance with 10 CFR 50.55a(b)(1) as applicable to the 2013 Edition. The regulation in 10 CFR 50.55a(b)(1)(ii) provides conditions for socket weld leg dimensions. In DCA Part 2, Tier 2, Section 5.4.2.4, “Tests and Inspections,” and Section 6.1.1.2, “Composition and Compatibility of Core Cooling Coolants,” show that socket welds are not used on lines greater than or equal to DN 20 (NPS 3/4) and any socket weld used on piping less than DN 20 (NPS 3/4) conforms to 10 CFR 50.55a(b)(1)(ii) and ASME B16.11 “Forged Fittings, Socket-Welding and Threaded.” The staff finds the DCA Part 2, Tier 2, statements regarding socket welded connections acceptable because they show that socket welded connections in the NuScale design conform to the 10 CFR 50.55a(b)(1)(ii) conditions for weld leg dimensions.

3.12.4.2.7 Nonseismic/Seismic Interaction

DCA Part 2, Tier 2, Section 3.12.3.8, “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 nonseismic piping is attached to seismic Category I piping, the nonseismic 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 nonseismic piping is designed not to cause a failure of the seismic Category I piping. The applicant’s provisions for considering nonseismic 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

DCA Part 2, Tier 2, Section 3.12.3.9, “Seismic Category I Buried Piping,” states that the NuScale design does not include ASME BPV Code buried piping. It also discusses the possibility that a COL applicant may find it necessary to route a buried piping line to add makeup inventory to the reactor pool for long-term support beyond the DBE. The applicant’s position on buried piping is acceptable to the staff since buried piping is not part of the NuScale design, and because DCA Part 2 shows that the COL applicant will develop the methodology for the design and analysis of buried piping if needed.

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

DCA Part 2, Tier 2, Section 3.12.4.1, “Computer Codes,” lists ANSYS and AUTOPIPE for computer programs used in the design of NuScale piping. DCA Part 2 also introduces COL Item 3.12-1, which allows a COL applicant that references the NuScale Power Plant DC 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. DCA Part 2, Tier 2, 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. SER Section 3.9.1 presents further evaluation of the acceptance of computer programs used in analyses and benchmarking methods.

3.12.4.3.2 Decoupling Criteria

In DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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. It also states that ASME Class 1 piping analysis does not use overlapping models and that a limited number of Class 2 or Class 3 piping analyses may use overlapping models if the routing of the connecting B31.1 piping is not yet completed to the next anchor. 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 DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Sections 3.7.1 and 3.7.2.

3.12.4.4.2 Design Transients

DCA Part 2, Tier 2, Section 3.9.1, discusses design transients and operating condition level categories, as defined in ASME BPV Code, Section III. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.12.5.3, "Loadings and Load Combination," 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. In DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.5.3, the applicant stated that because the operating-basis earthquake (OBE) is defined as one-third of the SSE, the OBE is not considered in the design and showed that for earthquakes that exceed the OBE an operating reactor will be shut down for inspections. This portion of DCA Part 2 is in accordance with 10 CFR Part 50, Appendix S, and therefore is acceptable to the staff. This portion of DCA Part 2 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 Tables 3.12-1, 3.12-2, and 3.12-3, "Required Load Combinations for Class 1, 2, & 3 Supports," of DCA Part 2, Tier 2, 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. 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.

The staff reviewed the proposed loads, load combinations, and stress limits given in the DCA Part 2, Tier 2, 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 of DCA Part 2, Tier 2, 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

DCA Part 2, Tier 2, Section 3.12.3.5, "Damping Values," states that the single damping value of 4 percent is used in the 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, and as mentioned above, the OBE is not part of the NuScale design.

In DCA Part 2, Tier 2, Section 3.12.3.5, 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. DCA Part 2, Tier 2, Section 3.12.3.2.2, shows two techniques that are used to determine composite modal damping and their formulations. One is based on mass and the other on stiffness as the weighting function. The staff reviewed the applicant's techniques and formulations for determining composite modal damping values and found them acceptable because they are 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

DCA Part 2, Tier 2, Section 3.12.5.4, "Combination of Modal Responses," states that Section 3.12.3.2 addresses the combination of modal responses. Section 3.12.4.2.2.3 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.3.4, the applicant showed that when the time-history method is used for seismic analysis of NuScale piping, the modal superposition method is utilized and that 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.

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 DCA Part 2, Tier 2, Section 3.12.5.5, "Fatigue Evaluation of ASME BPV 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, $(PCE_{\text{max}}) = 20$; 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 DCA Part 2, Tier 2, Section 3.12.5.5, 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 DCA Part 2, Tier 2, Section 3.12.5.6, "Fatigue Evaluation of ASME BPV 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 52.47(a)(22). SRP Section 3.12 includes criteria related to this bulletin, to the extent that the issue applies to a given design.

In DCA Part 2, Tier 2, Section 3.12.5.7, "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 Materials Reliability Program (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 DCA Part 2, Tier 2, Section 3.12.5.7, 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 (see ADAMS Accession No. 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. It 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. DCA Part 2, Tier 2, 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 52.47(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 (ADAMS Accession No. 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.

The staff reviewed NuScale's piping layout and concurs with the applicant's determination in DCA Part 2, Tier 2, Section 3.12.5.8.3, "Feedwater Line Stratification," 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.

The staff reviewed the piping layout of the DHRS to FW line. The staff noted that the DHRS condensate return piping from the passive condenser penetrates the CNV and is routed to the FW piping. This section of DHRS is not isolable from the FW line. When the DHRS is not in operation, this section of DHRS is full of stagnant water at a lower temperature than the FW and therefore could create the potential for thermal stratification.

In a letter dated February 7, 2018 (ADAMS Accession No. ML18038B623), the applicant confirmed that the DHRS is not isolable from the FW line. As shown in DCA Part 2, Tier 1, Table 2.1-1, the DHRS line is ASME Class 2. The applicant noted that the DHRS line is not connected to the RCS, and therefore, BL-88-08 is not directly applicable to the DHRS. However, the applicant evaluated the DHRS line for BL-88-08 TASCS effects. The applicant's response identified that the CFD analysis performed shows that the temperature fluctuations in the DHRS condensate piping and DHRS containment penetration cause thermal stresses that are below the fatigue endurance limit for the materials of the piping, welds, and CNV. The staff accepts the applicant's response because it provides reasonable assurance by evaluation that although the DHRS condensate line is susceptible to TASCS, fatigue damage is prevented. In addition, the applicant in its response showed that it has incorporated the operating experience from BL-88-08 into the piping design specification requiring that the DHRS condensate piping thermal cyclic loads be addressed in the ASME analysis and design report. The staff finds the applicant's response acceptable because it satisfies 10 CFR 52.47(a)(22), which requires that applicants provide information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

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 DCA Part 2, Tier 2, Section 3.12.5.8.1, "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 DCA Part 2, Tier 2, Section 3.12.5.9, "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

DCA Part 2, Tier 2, Section 3.12.5.10, “Functional Capability,” indicates that the functional capability provisions for ASME Class 1, 2, and 3 piping systems needed to provide an adequate fluid flowpath under Level D service loading conditions are consistent with the guidance of NUREG-1367. The section also shows that it satisfies NUREG-1367, Section 9.1, “Functional Capability Assurance, Present Code Requirements.” Since the applicant committed to satisfying the provisions of NUREG-1367, which is the current staff guidance related to functional capability referenced in SRP Section 3.12, the staff finds this acceptable.

NUREG-1367 was developed to address concerns that the increased Level D stress limits in some of the edition years of the ASME BPV Code were high enough that the functional capability of piping subject to such stresses was questioned. The staff 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, 2013 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

DCA Part 2, Tier 2, Section 3.12.5.11, “Combination of Inertial and Seismic Anchor Motion Effects,” shows that DCA Part 2, Tier 2, Section 3.12.3.2.9, discusses how the SAM effects have been evaluated for piping. Section 3.12.4.2.2.5 of this report discusses the staff’s evaluation of the applicant’s combination of inertial and SAM effects. The staff, per its review shown in Section 3.12.4.2.2.5 of this report, finds the applicant’s analysis acceptable.

3.12.4.4.14 Operating-Basis Earthquake as a Design Load

In DCA Part 2, Tier 2, Section 3.12.5.12, “Operating-Basis Earthquake as a Design Load,” the applicant referred to DCA Part 2, Tier 2, Section 3.7, and 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.

Section 3.12.4.4.7 of this report discusses the applicant’s reasons for eliminating the OBE from the piping design and its rationale for considering the OBE effect in the fatigue evaluation of piping. The staff found these positions acceptable. Section 3.7 of this report documents the staff’s evaluation of DCA Part 2, Tier 2, Section 3.7.

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.

DCA Part 2, Tier 2, Section 3.12.5.13, “Welded Attachments,” shows that for the NuScale ASME Class 1 piping, no welded attachments to the piping are permitted for support or restraint of the piping because of design and service loads, except for other functions not associated with maintaining the structural integrity of the piping pressure boundary, such as pipe whip and rupture restraint. Section 3.12.5.13 also states that welded attachments to Class 2 and 3 piping are permitted for the structural qualification of piping. According to the applicant, pipe welded attachments are considered in accordance with ASME BPV 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

In DCA Part 2, Tier 2, 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 of this material appears in Section 3.12.4.4.4 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 the "Thermal Expansion" portion of DCA Part 2, Tier 2, Section 3.12.5.3, and Section 3.12.5.14, "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. The applicant also stated that if the Class 2 or Class 3 piping is connected to a Class 1 component, thermal analysis of the Class 2 or 3 piping is required so that the effects of piping expansion can be included in the analysis of the Class 1 component. DCA Part 2, Tier 2, 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 finds the applicant's approach to the minimum temperature for thermal analysis acceptable because the applicant has committed to using ASME BPV Code, Section III, Division 1 (2013 Edition), for the structural qualification of piping. The ASME BPV Code states in Subsubparagraph NC/ND-3673.1(b) that piping structural analysis for thermal expansion is not required for Class 2 and 3 piping if the operating temperature of the piping system is at or below 150 degrees F (65 degrees C) and the piping is laid out with inherent flexibility, as provided in Subparagraph NC-3672.7. In addition, as discussed above, the applicant has provided adequate assurance that the evaluation of Class 1 components accounts for the thermal expansion effects of Class 2 or 3 piping.

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 DCA Part 2, Tier 2, Section 3.12.5.15, "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 overpressurization 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

DCA Part 2, Tier 2, Section 3.12.5.16, "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 DCA, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.6.1 and Section 3.12.6.8, "Seismic Self-Weight Excitation," for ASME-classified supports and for seismic Category I supports, the applicant used ASME BPV Code, Section III, Subsection NF. In addition, for Service Level D, the NuScale pipe support design uses the stress limits in the ASME BPV Code, Section III, Nonmandatory Appendix F. SRP Section 3.12, Acceptance Criterion II.D.i, states that the design of ASME Class 1, 2, and 3 piping supports should comply with the design criteria requirements of ASME BPV Code, Section III, Subsection NF. Subsection NF states that Appendix F is used for the stress limit factors for Service Level D. DCA Part 2, Tier 2, 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 linear-type and plate-and-shell-type 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.

In DCA Part 2, Tier 2, Section 3.12.6.1, the applicant also discussed codes for the design of seismic Category II (applicant's term) and nonseismic supports. (For the applicant's definition of seismic Category II, see Section 3.12.4.2.7 above.) The applicant showed that standard supports for seismic Category II piping are designed, manufactured, tested, and installed in accordance with Subsection NF of the ASME BPV Code, which as described in the paragraph above is acceptable to the staff. Standard supports and standard support parts used in nonstandard supports for nonseismic piping in the NuScale B31.1 piping are designed in accordance with the requirements of the ASME BPV Code B31.1, paragraphs 120 and 121. The structural elements of nonstandard supports for seismic Category II and nonseismic piping are designed using guidance from ANSI/AISC N690. Structural elements of supports for nonseismic piping are also designed using guidance from the AISC *Steel Construction Manual*, 14th Edition, 2011. The staff recognizes that the methods in these codes provide reasonable assurance of the structural integrity of Category II and nonseismic pipe supports and have been used in pipe support design in current nuclear plants. Based on precedent, the staff finds that the design criteria in DCA Part 2 for the seismic Category II and nonseismic piping supports are 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 DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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). DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.6.2, some ASME Class 3 supports are connected to concrete building structures or building steel. In this DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.5.3. The staff's evaluation of Section 3.12.5.3 appears above in Section 3.12.4.4.3. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.9-2. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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

The use of baseplates for pipe supports in the NuScale design is expected to be minimal. In DCA Part 2, Tier 2, Section 3.12.6.4, the applicant stated that the NuScale design baseplates are not used for any Class 1 or Class 2 pipe supports because these are supported by the CNV. However, some Class 3 pipe supports may be supported off of the building and may use baseplates. The applicant also stated that, in cases where these designs are needed, concrete anchor bolts are evaluated using ACI 349 with the conditions and limitations given in RG 1.199. The applicant also stated that all aspects of the anchor bolt design, baseplate flexibility, and factors of safety will be addressed as identified in NRC BL-79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts." The staff noted that SRP Section 3.12, Acceptance Criterion II.D.iv, states that the design of the pipe support baseplates and anchor bolts should comply with guidance in NRC BL-79-02.

Based on the review described above, the staff finds the applicant's position on pipe support baseplate and anchor bolt design acceptable, as the applicant's position meets the staff's regulatory guidance and SRP recommendation.

3.12.4.5.5 Use of Energy Absorber and Limit Stops

Because DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.6.8, acceptable because it is consistent with the staff SRP guidance and is also the same method that the staff approved in past DCAs, as documented in NUREG-1793, Section 3.12.6.8.

3.12.4.5.9 Design of Supplementary Steel

In DCA Part 2, Tier 2, Section 3.12.6.9, “Design of Supplementary Steel,” the applicant also provided its position on the design of supplementary steel for pipe supports. This section states that all seismic Category I pipe supports for NuScale are designed to ASME BPV Code, Section III, Subsection NF, and seismic Category II pipe supports, including supplemental steel required to connect the structural support elements to building structures, are designed using ANSI/AISC N690. Supplemental steel for nonseismic pipe supports is designed using the AISC *Steel Construction Manual*, 14th Edition. As stated in Section 3.12.4.5.1, of this SER because ASME BPV Code, Section III, Subsection NF is recommended by SRP Section 3.12 for ASME Class 1, 2, and 3 pipe supports and because the use of ANSI/AISC N690 and the use of the AISC *Steel Construction Manual*, provide reasonable assurance that the structural adequacy of the seismic Category II and nonseismic pipe supports is maintained, the staff finds the applicant's approach to the design of supplementary steel in pipe supports 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 COF.

In DCA Part 2, Tier 2, 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 (ADAMS Accession No. ML14099A519)). Therefore, the staff accepts the COF value of 0.3 for steel-to-steel applications.

DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.12.6.13, "Pipe Deflection Limit," the applicant stated that standard supports, including springs, are generally not used for ASME class piping inside the NuScale reactor module. However, some of the Class 3 supports may use spring supports. The applicant also showed that if standard supports, including springs, or standard support parts are used, the manufacturer's recommendations are followed to determine deflection limits (or travel range limits for springs). Where rods or struts are used, an installation tolerance of 1 degree is applied to the manufacturer's swing angle limit. The applicant also showed that maximum displacements and rotations determined at B31.1 piping flexible joints are verified to be within the manufacturer's recommended limits.

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

DCA Part 2, Tier 2, Section 3.12.6.13, 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 DCA, 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 DCA Part 2, Tier 2, Section 3.12.

Table 3.12-1: NuScale COL Information Items for Section 3.12

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.12-1	A COL applicant that references the NuScale Power Plant design certification may use a piping analysis program other than the programs listed in Section 3.12.4.1; however, the applicant will implement a benchmark program using the models for NuScale Power Plant standard design.	3.12.4.3
COL Item 3.12-2	A COL applicant that references the NuScale Power Plant design certification will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. A COL applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.	3.12.5.1

DCA Part 2, Tier 2, Section 3.12, also mentions that if a COL applicant referencing the NuScale Power Plant DC finds it necessary to route Class 1, 2, and 3 piping not included in the NuScale Power Plant DC 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 DCA Part 2, Tier 2, 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 52.47(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

By application dated August 22, 2019 (ADAMS Accession No. ML19241A315), the applicant submitted the information in DCA Part 2, Tier 1, "Certified Design Descriptions and Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)," and DCA Part 2, Tier 2, 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). While this SER is based on Revision 3 of the DCA, additional communication from the applicant relevant to the material in this section can be found in the following letters:

- August 10, 2017 (ADAMS Accession No. ML17222A218)
- December 5, 2017 (ADAMS Accession No. ML17339A997)
- December 11, 2017 (ADAMS Accession No. ML17345B219)
- December 12, 2017 (partial response) (ADAMS Accession No. ML17346A519)
- December 15, 2017 (ADAMS Accession No. ML17349A815)
- December 18, 2017 (ADAMS Accession No. ML17352B254)
- December 18, 2017 (ADAMS Accession No. ML17352B263)
- January 29, 2018 (partial response) (ADAMS Accession No. ML18029A846)
- February 12, 2018 (ADAMS Accession No. ML18043B167)
- March 21, 2018 (partial response) (ADAMS Accession No. ML18080A176)
- March 21, 2018 (partial response) (ADAMS Accession No. ML18080A177)
- March 27, 2018 (ADAMS Accession No. ML18086B442)
- May 15, 2018 (ADAMS Accession No. ML18135A127)
- July 5, 2018 (ADAMS Accession No. ML18186A678)
- July 12, 2018 (ADAMS Accession No. ML18193B178)
- August 23, 2018 (ADAMS Accession No. ML18235A536)
- September 13, 2018 (ADAMS Accession No. ML18256A300)
- January 22, 2019 (ADAMS Accession No. ML19022A364)

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

DCA Part 2, Tier 1: Information associated with this section is found in DCA Part 2, Tier 1, Section 2.1.

DCA Part 2, Tier 2: The applicant described the use of threaded fasteners associated with Class 1, 2, and 3 pressure-retaining joints in DCA Part 2, Tier 2, Section 3.13, which is summarized in the following discussion. Other sections of the DCA also discuss the use of threaded fasteners.

As described in DCA Part 2, Tier 2, 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 (ADAMS Accession No. ML100160003)
- RG 1.65, Revision 1, “Materials and Inspections for Reactor Vessel Closure Studs,” issued April 2010 (ADAMS Accession No. ML092050716)
- RG 1.84, Revision 36

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 DCA Part 2, Tier 2, Table 5.2-1, “American Society of Mechanical Engineers Code Cases.”

DCA Part 2, Tier 2, Table 1.9-3, states that the DCA conforms to the acceptance criteria in SRP Section 3.13.

3.13.2.1 Design Considerations

DCA Part 2, Tier 2, Section 3.13.1, “Design Considerations,” states that pressure boundary threaded fasteners comply with ASME BPV Code Class 1, 2, and 3 requirements.

DCA Part 2, Tier 2, Section 5.2.3.6, “Threaded Fasteners,” discusses the RCPB threaded fasteners and cites Section 3.13. DCA Part 2, Tier 2, Section 5.2.5.3, “Reactor Pressure Vessel Flange Leak-Off Monitoring,” states that bolted flanges and covers in the RCS are sealed by double concentric O-rings. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 6.2.4.2.1, also discusses the CNV flange connections.

DCA Part 2, Tier 2, Section 5.4.1.5, discusses the pressure-retaining components that are part of the SGs, including bolting material, which are listed in DCA Part 2, Tier 2, Table 5.4-3, “Steam Generator Piping, Piping Supports, and Flow Restrictor Materials.”

DCA Part 2, Tier 2, Section 5.3.1.7, “Reactor Vessel Fasteners,” states where threaded inserts are used on the RPV, as well as their design requirements. DCA Part 2, Tier 2, Section 6.2.1.1.2, “Design Features,” describes where threaded inserts are located on the CNV. DCA Part 2, Tier 2, Table 5.2-4, “Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances,” states that the threaded inserts are dual-certified Type 304/304L stainless steel. DCA Part 2, Tier 2, Section 6.1.1.1, “Material Selection and Fabrication,” and Table 6.1-1, “Material Specifications for ESF Components,” also states that threaded inserts for CNV bolting are fabricated of Type 304/304L stainless steel. The seal weld for all threaded inserts requires a fabrication examination using magnetic particles or liquid penetrant in accordance with the ASME BPV Code, Section III, Division 1, paragraph NB-5271.

In DCA Part 2, Tier 2, 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 CNV upper flange to the clad. The stud welds have a fabrication liquid penetrant exam and no PSI or ISI exams.

NuScale RAI response letters (ADAMS Accession Nos. ML18080A176 and ML18080A177) provide additional detail on the design of the lock plates.

3.13.2.2 Materials Selection

DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.13-1, “ASME BPV Code Section III Criteria for Selection and Testing of Bolted Materials.” DCA Part 2, Tier 2, Table 3.13-1, is based on SRP Table 3.13-1. DCA Part 2, Tier 2, 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 SCC. The threaded fastener materials selected meet the requirements of the following three EPRI reports related to boric acid corrosion:

- EPRI TR-101108, “Boric Acid Corrosion Evaluation (BACE) Program, Phase—Task 1 Report,” issued December 1993
- EPRI NP-5985, “Boric Acid Corrosion of Carbon and Low-Alloy Steel Pressure-Boundary Components in PWRs,” issued August 1988
- EPRI NP-5558-SL, “Boric Acid Application Guidelines for Intergranular Corrosion Inhibition,” issued December 1987

Specifically, the threaded fasteners are fabricated from either SB-637, UNS N07718 (Alloy 718); or SA-564, Grade 630, Condition H1100. 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 DCA Part 2, Tier 2, 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 precipitation-hardening heat treatment. All uses of SA-564, Grade 630, threaded fasteners are heat treated to the H1100 condition.

In DCA Part 2, Tier 2, 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 RG 1.28. The applicant also stated that lubricants will be selected in accordance with the guidance in NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,” issued June 1990 (ADAMS Accession No. 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. "NuScale Power, LLC Response to NRC Request for Additional Information No. 252 (eRAI No. 9183) on the NuScale Design Certification Application," dated December 5, 2017 (ADAMS Accession No. ML17339A997), explains that precipitation-hardened SA-564, Grade 630, undergoes fracture toughness testing in accordance with the ASME BPV Code based on the 10 CFR Part 50, Appendix G, definition of ferritic material.

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 NCA-3862. 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

The applicant stated that ISI will be in accordance with the ASME BPV Code, Section XI, as listed in DCA Part 2, Tier 2, 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."

DCA Part 2, Tier 2, 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. This section of the DCA describes the inspection for the RPV and CNV main flange bolts (greater than 5 cm (2 in.) in diameter) and pressure-retaining bolting that is 5 cm (2 in.) or less in diameter. The only threaded fasteners greater than 5 cm (2 in.) in diameter are the RPV and CNV main flanges.

In DCA Part 2, Tier 2, Table 5.2-6, "Reactor Pressure Vessel Inspection Elements," and Table 6.2-3, "Containment Vessel Inspection Elements," state that the threaded fastener insert welds will receive a VT-1 examination when bolts are removed.

In DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.13, appear in DCA Part 2, Tier 1, Section 2.1. Section 14.3 of this SER 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 DCA Part 2, Tier 2, 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 (ADAMS Accession No. ML19149A298)

- TR-0917-56119-NP, Revision 1, "CNV Ultimate Pressure Integrity," issued June 2019 (ADAMS Accession No. 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.

The staff notes that RG 1.28 is not mentioned in SRP Section 3.13. RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of

Water-Cooled Nuclear Power Plants," was withdrawn (79 FR 38963; July 9, 2014) and replaced with RG 1.28 after the most recent version of SRP Section 3.13 was issued in March 2007.

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 two tables in DCA Part 2, Tier 2, Section 3.13, and verified that they cite the appropriate portions of the 2013 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 design. The staff reviewed these fasteners as part of its evaluation of DCA Part 2, Tier 2, 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	DCA Part 2, Tier 2, Section(s)
CNV Flange	Main Flange Closure Studs	Alloy 718	N/A	3.8.2.1.2 6.1.1.1 Table 6.1-1
RPV Flange	Main Flange Closure Studs	Alloy 718	N/A	5.2.3.6 Table 5.2-4 5.3.1.7
RPV Head	Reactor Safety Valve	Alloy 718	RPV 18–19	5.2.2.2.1 5.2.3.6 Table 5.2-4
RPV Head	I+C Channels A–D	Alloy 718	RPV 39–42	Table 5.2-4
RPV Head	RVV Flange	Alloy 718	N/A	5.1.3.6 5.2.2.5 5.2.3.6
Upper RPV Section	RRV Flange	Alloy 718	N/A	5.1.3.6 5.2.2.5 5.2.3.6
Upper RPV Section	FW Access Port 1–4	Alloy 718	RPV 43–46	5.4.1.2 5.4.1.5 Table 5.2-4 Table 5.4-3
Upper RPV Section	Main Steam Access Port 1–4	Alloy 718	RPV 47–50	5.4.1.2 5.4.1.5 Table 5.2-4 Table 5.4-3
Upper RPV Section	PZR Access Port 1–2	Alloy 718	RPV 21–22	5.2.3.6 Table 5.2-4
Upper CNV	PZR Access Port 1–2	Alloy 718	CNV 31–32	3.8.2.1.4 6.1.1.1 Table 6.1-1
Upper CNV	CNV Manway Cover 1	Alloy 718	CNV 26	3.8.2.1.4 6.1.1.1 Table 6.1-1

Location	Component	Material	Penetration	DCA Part 2, Tier 2, Section(s)
Upper CNV	SG Inspection Port Cover 1–4	Alloy 718	CNV 27–30	3.8.2.1.4 6.1.1.1 Table 6.1-1
CNV Top Head	CRDM Access Opening	Grade 630, Condition H1100	CNV 25	3.8.2.1.4 6.1.1.1 Table 6.1-1
CNV Top Head	CNV Head Manway	Grade 630, Condition H1100	CNV 24	3.8.2.1.4 6.1.1.1 Table 6.1-1
CNV Top Head	Electrical PXR Heater Power	Grade 630, Condition H1100	CNV 15–16	3.8.2.1.6 6.1.1.1 Table 6.1-1
CNV Top Head	CRDM Power	Grade 630, Condition H1100	CNV 37	3.8.2.1.6 6.1.1.1 Table 6.1-1
CNV Top Head	I+C Divisions 1–2	Grade 630, Condition H1100	CNV 8–9	3.8.2.1.6 6.1.1.1 Table 6.1-1
CNV Top Head	I+C Channels A–D	Grade 630, Condition H1100	CNV 17–20	3.8.2.1.6 6.1.1.1 Table 6.1-1
CNV Top Head	RPI Group 1–2	Grade 630, Condition H1100	CNV 38–39	3.8.2.1.6 6.1.1.1 Table 6.1-1

DCA Part 2, Tier 2, Table 6.1-3, “Pressure Retaining Materials for RCPB and ESF Valves,” lists the acceptable materials for the RCPB and engineered safety feature (ESF) valve-threaded fasteners (e.g., the ECCS trip/reset valve and CIV test port cover threaded fasteners). These are not included in the above table.

The staff performed an audit related to the use of threaded inserts in the NuScale design (see the audit report dated December 6, 2017 (ADAMS Accession No. ML17335A105)).

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.

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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, and10 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 because of 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. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 3.13.1.1, cites three EPRI reports related to boric acid corrosion and states that the materials selected meet the applicable requirements in these reports. The materials selected are generally resistant to boric acid corrosion. Therefore, the staff finds that meeting the applicable requirements in these reports is acceptable for the threaded fastener materials.

The applicant stated that the design for threaded fasteners meets the cleaning criteria of RG 1.28. DCA Part 2, Tier 2, 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. Additionally, DCA Part 2, Tier 2, Section 3.13.1.1, states that lubricants containing molybdenum sulfide are prohibited. DCA Part 2, Tier 2, 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 RG 1.28 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.

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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 3.13-1, for ASME BPV Code Class 1, 2 and 3 systems. Since DCA Part 2, Tier 2, 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

In DCA Part 2, Tier 2, Sections 5.2.4, 6.2.1, and 6.6 present additional information on PSI and ISI.

For two unique portions of the NuScale design—the threaded inserts and lock plates—NuScale provided augmented PSI and ISI. In DCA Part 2, Tier 2, Tables 5.2-6 and 6.2-3 list the augmented VT-1 inspection for the threaded insert seal welds when bolts are removed. The lock plate stud welds are not subject to PSI or ISI exams. Since these augmented inspections are capable of detecting unacceptable indications and ensure cladding integrity for the unique threaded insert portion of the NuScale design, the staff finds these augmented fabrication inspections acceptable. 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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.6.2.5. The staff finds these augmented inspections acceptable as documented in Section 3.6.2 of this SER.

Compliance with the requirements of the ASME BPV Code, Section XI, also satisfies the regulatory requirements of 10 CFR 50.55a. In DCA Part 2, Tier 2, Section 3.13.1.4, Section 3.13.2, Table 3.13-1, and Table 3.13-2 state 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. DCA Part 2, Tier 2, 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 Tier 1 and 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 Section 14.3 of the SER.

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 DCA Part 2, Tier 2, Table 1.8-2.

Table 3.13-2: NuScale COL Information Item for Section 3.13

Item No.	Description	DCA Part 2, Tier 2, Section
COL Item 3.13-1	A COL applicant that references the NuScale Power Plant design certification will provide an inservice inspection program for American Society of Mechanical Engineers (ASME) Class 1, 2 and 3 threaded fasteners. The program will identify the applicable edition and addenda of ASME Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.	3.13

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.

The staff reviewed DCA Part 2, Tier 2, Section 3.13.2, which lists COL Item 3.13-1. The staff confirmed the consistency of the wording with DCA Part 2, Tier 2, Table 1.8-2.

3.13.6 Conclusion

Based on its review of the information provided by NuScale, the staff concludes that the NuScale DCA 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.